"12 YEARS LATER... AN UPDATE REPORT ON THE NUCLEAR REACTOR STUDY"

Commissioned by Dr. Bertram Wolfe,
Vice President and Chief Scientist,
Nuclear Energy Operations, 1987

"NUCLEAR REACTOR STUDY"

Conducted by Dr. C.E. Reed,
Senior Vice President,
Corporate Studies and Programs, 1975

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NUCLEAR REACTOR STUDY

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12 YEARS LATER ... AN UPDATE REPORT

COMMISSIONED BY DR. BERTRAM WOLFE

VICE PRESIDENT AND CHIEF SCIENTIST

NUCLEAR ENERGY OPERATIONS

1987

PREAMBLE: THE REED REPORT -- A 1987 PERSPECTIVE

General Electric has decided to release the report on its 1975 Nuclear Reactor Study, the Reed Report. This preamble places the Reed study in perspective and explains why GE has decided to release the report.

The Reed Report is being released in order to overcome several misleading impressions which have been widely circulated. In fact:

o NRC reviews in 1976 and 1978 confirmed that all safety issues mentioned in the Reed Report had been disclosed to NRC previously.

o The Reed study was performed early in the BWR/6 design cycle, and the recommendations were implemented before BWR/6 plants went into operation.

o The Reed Study was not kept secret by GE from government agencies and utilities.

The Reed study was a review of the preliminary design for GE's latest reactor product line, the BWR/6. The study was undertaken because the BWR/6 had attained unprecedented success in the utility market, winning 42 orders within two years. The objective of the review was to assure that the BWR/6 would be a product of the highest quality and that the Nuclear Energy Division had the technical resources to perform successfully the multi-billion dollar contract backlog. GE's Chairman Reginald H. Jones asked one of GE's most respected scientists Dr. Charles Reed, Senior Vice President-Corporate Studies and Programs, to organize a task force composed of the Company's most experienced scientists and engineers to conduct an independent review of the preliminary design for the BWR/6 which was being developed by GE's Nuclear Energy Division.

The Reed Task Force consisted of ten people, only two of whom came from the nuclear division. After a nine-month study, the task force issued a lengthy report making over one hundred findings and recommendations. Recommendations were implemented over the next five years. Because of the time cycle from order to operation in the nuclear industry, there was sufficient time to carry out the recommendations before the first U.S. BWR/6 went into operation ten years later, in 1985.

The performance of BWR/6 plants compares favorably with the best reactor types of this size throughout the world. Increased focus on nuclear fuel development programs has resulted in the achievement of a BWR fuel reliability record of 99.98%. The recommendations of the Reed Task Force and the investment GE made in implementing these recommendations clearly made a significant contribution to the success of the BWR/6.

The Confidentiality Issue

GE has limited the distribution of the Reed Report for the past twelve years for three reasons:

First, the Reed study was a candid review, reporting to corporate management on the technical work and organizational capability of one of GE's divisions. The Report was intended for an internal purpose; advising top management on what actions should be taken and what corporate resources were required by one of GE's decentralized businesses. Such studies are performed frequently; Dr. Reed chaired a dozen similar studies.

Internal self-critical reviews represent an important management technique for a large decentralized organization. The effectiveness of such reviews depends on complete candor between the organization under review and the members of the review group and also in the report submitted to top management. Making internal, self-critical studies public may set a troublesome precedent, because it could have a chilling effect on the candor with which future studies can be conducted.

More is at stake than GE's self interest. The effectiveness with which U.S. corporations manage large decentralized organizations represents one of America's competitive strengths in the world economy. The privacy of internal self-critical studies is as important to GE as the privacy of sources of information is to a newspaper.

Second, when issued, the report contained highly sensitive trade secrets, discussing GE's latest reactor product line and providing a road map outlining technical development programs. Making the report public would have been of great value to our competitors. The competitive significance of the report has diminished with the passage of time.

Third, with the passage of time the Reed Report has become dated. While this has decreased the risk of competitive damage, it has increased the danger of misunderstanding and distortion. A 1975 study of the preliminary design of the BWR/6 can only be understood in the light of subsequent events, including design changes in the BWR/6 plants as actually built, other GE responses to the Reed recommendations, and the enormous increase in technical knowledge and reactor operating experience in the past 12 years. To place the Reed Report in context requires an extensive update report. Such an effort will help the fair-minded reader. It provides no protection against those whose interests are served by using selected excerpts out of context.

GE has recognized from the beginning that its interest in the confidentiality of the Reed Report must be balanced against the legitimate concerns of others. For that reason the Report has been made available on numerous occasions to those with a direct interest. This has included the Nuclear Regulatory Commission (NRC), two Congressional committees, utilities with boiling water reactors (BWRs), nuclear licensing boards, and intervenors in nuclear plant licensing proceedings. Such access was provided based on assurances of confidential treatment.

Decision to Release the Report

On May 27, 1987 lawyers representing the plaintiffs in the Zimmer suit (Cincinnati Gas & Electric v. GE) filed in open court several documents referring to the Reed Report. By so doing, plaintiffs breached the terms of a protective order, which they had signed, under which such documents could only be filed under seal. These documents, which did not include the Reed Report itself, promptly became available to a reporter, whose paper ran a series of misleading articles. These articles in turn resulted in other articles around the country. The largest number of articles appeared in communities where utilites were operating nuclear power plants with GE reactors.

The impression conveyed by many of the articles was that GE had engaged in a "12-year cover-up" of a "secret report" which contained "undisclosed safety problems". The inference was that these problems affected presently operating power plants. The articles ignored or underplayed the real facts, even though these facts had been promptly made available by GE:

- o NRC reviews in 1976 and 1978 confirmed that all safety issues mentioned in the Reed Report had been disclosed to NRC previously.
- o The Reed study was performed early in the BWR/6 design cycle, and the recommendations were implemented before BWR/6 plants went into operation.
- o The Reed Study was not kept secret by GE from government agencies and utilities.

The misleading articles raised concerns in communities where GE BWRs are in operation and were troublesome for utilities owning these plants. The articles also led to requests for the release of the Reed Report from public utility commissions and other public officials in states where GE reactors are in operation.

In light of these developments, GE has decided that the Reed Report should be released, without restriction on its distribution or reproduction. GE continues to believe that the privacy of internal self-critical studies is important. However, an exception is being made in the case of the Reed Report because of the interests of people in the communities involved and our utility customers, and in response to the requests of concerned public officials.

GE is releasing the Report of the Reed Task Force, 146 pages in length; two Appendices, totaling 80 pages; plus ten Sub Task Force Reports, totaling 620 pages. This 846-page package includes all of the documents which the Reed Task Force submitted in July 1975 to GE corporate management in Fairfield, Connecticut and to the management of the Nuclear Energy Division in San Jose, California. These 1975 documents are accompanied by an "Update Report" which describes in detail the actions which have been taken to implement the Reed Task Force recommendations. An executive summary of the Update Report is also included.

How The Reed Study Was Conducted

The Reed study was begun in October 1974, and the report was completed in July 1975. Dr. Reed assembled a task force of nine of GE's most experienced scientists and engineers. Two were from GE's nuclear business while the others were from different laboratories and businesses within GE. The task force held 11 meetings over a period of six months, each meeting lasting for two or more days. The group divided the work into 10 sub-tasks, which involved in-depth technical studies and theoretical evaluations of specific areas such as nuclear fuel, mechanical systems and materials, processes and chemistry. The Reed Task Force and its sub-groups met with scores of engineers and scientists involved in GE's nuclear operations. Before it was completed, more than 100 GE scientists and engineers were involved in the study.

In keeping with GE's long-standing belief that technological progress is made by providing opportunities for criticism and debate, the Reed study group conducted its review in an atmosphere of frankness and candor. A concerted effort was made to obtain information from all levels of the organization, not just senior management. The engineers and scientists directly involved in the development of the BWR/6 were encouraged to express their judgments on controversial design issues. This type of internal critical review is frequently used within General Electric and other companies engaged in evolving scientific and technical fields. The objective is to enable the participants to present critical assessments for management consideration. While this is a useful technique, it must be recognized that findings made by review groups on difficult issues sometimes prove to be exaggerated, or even incorrect. For example, a study similar to the Reed study concluded in 1957 that GE's first commercial boiling water reactor, the Dresden-1 BWR would only reach half of its design output of 175 megawatts. In actuality, Dresden-1 reached 175 megawatts in its initial 1959 operation and later was upgraded to 200 megawatts.

The Reed Report contained a large number of detailed findings and recommendations based upon many interviews and meetings. The recommendations dealt with the overall reactor design, as well as with specific plant components and systems. Recommendations were also made concerning development and test facilities, management and organization. The Report was reviewed with General Electric senior management, which strengthened many existing programs and initiated new activities to respond to the recommendations.

Not A Safety Study

The Reed study was not a safety review. Safety issues were the subject of separate and continuing programs in GE's nuclear division. However, given the breadth of the Reed study and GE's pervasive concern for safety, it was inevitable that a study of this type would touch on issues of potential safety significance. In 1975 GE's nuclear safety and licensing organization reviewed the Report in detail. That review resulted in the identification of 27 safety-related items, ultimately condensed to 25 issues. All of these issues were previously known to the Nuclear Energy Division and were being addressed in the BWR/6 design. Additionally, it was determined that all items were previously known to the NRC and no formal notification to the NRC was required.

Recently both NRC and GE have again reviewed the list of safety-related issues referred to in the Reed Report. Both concluded that all of these issues have been satisfactorily resolved.

<u>The Time Cycle for Nuclear Plants</u>

The BWR/6 represented the latest evolutionary step in GE's line of boiling water reactors, going back to the first BWR plant which went into utility service in 1959. Consistent with practice in the nuclear industry, and other high technology industries where lead times are long, the BWR/6 was introduced and sold on a preliminary design basis. The preliminary design incorporated the newest product concepts, with detailed engineering work to be performed thereafter. Because of rapid advances in technology and the long duration of nuclear power plant projects, it has been standard practice to contract for the latest technology in this fashion and to finalize designs as actual construction of the plant moves forward. In this way, utility customers could obtain the latest technology, affecting both plant safety and performance, and be assured that the design would reflect changing regulatory requirements — a constant and necessary concern in the nuclear industry.

While the BWR/6 plants were being constructed, all nuclear plant schedules were greatly extended by changing regulatory requirements resulting from the accident at Three Mile Island. The implementation of the Reed Task Force recommendations benefitted from these schedule extensions.

Prior Disclosures of Reed Report

The Report was shared with many others who needed to know about it.

- GE in 1975 advised Chairman Anders of the NRC of the scope and purpose of the study and GE's conclusions that no items in the Reed Report required reporting.

- In 1976, two senior NRC technical experts reviewed the Reed Report and concluded that it did not identify new safety issues and that there were no instances of significant safety concerns which had not been previously reported to the NRC.

- In 1976, GE testified on the Reed Report before the Joint Committee on Atomic Energy of the U.S. Congress and a technical member of the Joint Committee staff reviewed the report.

- In 1978, GE met with members of the Subcommittee on Energy and Power of the House Interstate and Foreign Commerce Committee and NRC personnel to review the status of Reed Report items relating to safety issues being reviewed by the NRC. Following up on that meeting, Chairman Hendrie of the NRC advised Chairman Dingell by letter that all issues had either been resolved or were scheduled for resolution as part of ongoing NRC programs.

- Portions of the Reed Report have been made available to intervenors and utilities in various NRC licensing proceedings where they were pertinent to issues in the proceedings.

- The NRC has had a copy of the Reed Report since 1979.

Cost Impact of Reed Report

General Electric spent many hundreds of millions of dollars in the development work leading to the BWR/6. While much of this work was underway before the Reed study took place, the Reed Report resulted in heightened top management attention, and assured that the programs were adequately staffed and funded. Some of the General Electric expenditures were in the following areas:

- o GE constructed a number of new test facilities or modified existing facilities to perform technical analyses and demonstrate the adequacy of designs.
- o As a result of extensive development work, GE improved the initial designs of some components to improve their reliability. An example is the flow control valve, where testing indicated a revised design would be desirable.
- o A major effort involved analytical and test work performed to verify and improve design methods to assure that the final design of systems and components would be satisfactory.

GE paid for all basic design work and equipment necessary to meet its contractual commitments. On issues addressed in the Reed Report utility customers only paid GE for changes in GE's contractual scope of supply, caused either by changing NRC regulations or options selected by customers to achieve better plant performance.

As is customary throughout the nuclear industry, GE and other suppliers were obligated to meet NRC regulatory requirements in their final designs, even if those regulations changed, as they frequently did. Because NRC changes often caused significant changes in the contractual scope of supply, nuclear contracts normally provided that the suppliers would be compensated for the cost of the change. Thus the costs of regulatory changes were borne by the utility owner not the supplier. Also, advances in technology, which exceeded original contract specifications, were made available at extra cost and were often accepted by the utilities.

Summary

The Reed Report, a critical evaluation of the preliminary design of the GE BWR/6 reactor, served to focus management attention on key technical issues to be resolved before the final design could be detailed. This focus and the development programs resulting from it led to a BWR/6 product line which was much improved over earlier designs.

The performance of BWR/6 plants compares favorably with the best reactor types of this size throughout the world. Increased focus on nuclear fuel development programs has resulted in the achievement of a BWR fuel reliability record of 99.98%. The recommendations of the Reed Task Force, and the investment GE made in implementing these recommendations, clearly made a significant contribution to the success of the BWR/6.

ORGANIZATION OF 1987 UPDATE ANALYSIS

The Update Report analyzes the findings and recommendations in the main report of the Reed task force and the 10 sub-task reports. There is a certain amount of replication and overlap among the many findings and recommendations made in the main report and the sub-tasks reports. This analysis groups and discusses the topics addressed in the reports. One topic may treat several recommendations in the reports and in a few cases, more than one topic may address the same recommendation.

For convenience, the appendices provide a cross reference to show where the findings and recommendations in the Reed Report are addressed here.

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EXECUTIVE SUMMARY

OF

1987 UPDATE ANALYSIS

UPDATE REPORT

EXECUTIVE SUMMARY OF 1987 UPDATE ANALYSIS

This section is an executive summary of the items in the main analysis sections of the Update Report. The executive summary paraphrases the finding or recommendation contained in the Reed Report and calls that the Issue. The corresponding Response describes the actions taken by GE and the Results describes what was the outcome relative to the issue raised. The issues are numbered in conformance with the main analysis sections of the Update Report list of issues. At the end of the main analysis sections of the Update Report is a series of appendices that give cross reference lists showing which items in this Update Report correspond to the findings or recommendations in the Reed Report including its sub-task reports.

1.0 Technology and Design Methods Improvements

1.1 Design Methods

Issue - Verification of adequacy of design methods, particularly in coupled nuclear/thermal hydraulic areas.

Response - GE expanded its efforts to acquire data in areas of steady state core power distribution and critical eigenvalue, system response to pressurization transients, and system hydrodynamic stability characteristics.

Result - The design methods proved to be adequate and no derates resulted. The data provided the required basis for qualification and verification of existing analytical models and for development of a new model using more highly refined calculations. NRC approved this model.

1.2 Emergency Core Cooling System (ECCS) Analysis

Issue - New NRC requirements required new ECCS computer programs.

Response - GE's evaluation models for ECCS were formally approved in 1982. Based on experimental data a realistic computer program called SAFER/GESTR was developed which demonstrated significant margin to previously calculated peak cladding temperature during LOCA conditions.

Result - GE successfully developed qualified computer models and BWR/6 plants were never adversely impacted. The impact on earlier designs caused by this regulatory change was alleviated through the development by GE of new, qualified models which NRC approved.

1.3 Seismic Design

Issue - Upgrade and focus organizational efforts to meet new NRC seismic requirements.

Response - Within GE a single organizational group was made responsible for methods, criteria and computer programs and a second group became responsible for application of the technology to plants under construction.

Result - GE and architect engineers successfully worked together to avoid adverse impacts which might otherwise have resulted from this regulatory change. The consolidation of responsibility within GE assured that seismic state-of-the-art technology was properly identified and analyzed and applied to equipment evaluations.

1.4 Flow Induced Vibration

Issue - Complement computer modeling of flow induced vibration with a test program to confirm design margins for reactor internals were adequate.

Response - The High Flow Hydraulic Test Facility was constructed as a full size sector model of the BWR/6. Initial operation of the facility occurred in May 1978.

Result - Testing confirmed BWR/5 and 6 design margins were adequate.

1.5 Plant Radiation

Issue - Buildup of radiation levels in plants and potential for impact on plant maintenance and servicing.

Response - Since the time of the Reed Report, GE and others in the industry have focused considerable effort in developing better tools, plans and procedures to minimize the impact of radiation levels on maintenance operations.

Result - Significant advancements have been made in understanding radiation buildup in operating plants and minimizing its impact on maintenance and servicing of these plants.

1.6 Non-Destructive Testing (NDT)

Issue - Organizational diffusion of NDT efforts in GE.

Response - A consolidated NDT function has been established.

Result - The major NDT function has been strengthened, consolidated, land is managed by an NDT professional.

2.0 Equipment Reliability Assurance

2.1 Core Internals

Issue - Design life/replaceability of reactor vessel internal components.

Response - Evaluation of materials subject to radiation using available experimental data shows that BWR/6 internal components should last their design lifetime.

Result - Notwithstanding the evaluation performed that justifies expected useful life, BWR/6 reactor internals have been designed to be removable.

2.2 Control Blade Life

Issue - Cracks in absorber rods in operating plants suggested a reduced lifetime for control blades.

Response - Review of operating data resulted in developing a control blade life prediction model and control blade lifetime criteria. These criteria are used by operating plants.

Result - A new control rod blade using hafnium has also been developed. This blade has a longer life expectancy and has now received NRC approval.

2.3 Control Blade Tolerances

Issue - Unacceptable frequency of out-of-tolerance finished products requiring excessive scrap and rework.

Response - Improved handling and forming techniques were incorporated in the manufacturing process.

Result - Current experience as measured by quality checks and performance feedback demonstrates that these products meet manufacturing tolerances and other requirements. Excessive scrap and rework were avoided.

2.4 Setpoint Drift

Issue - Normally experienced drift of instrumentation setpoints required a high frequency of abnormal occurrences reports to NRC due to unnecessarily conservative setpoint limits.

Response - Manufacturers' field data and nuclear plant operating data on instrument drift have been statistically analyzed to select realistic setpoint values that assure plant performance within license limits including appropriate margin.

Result - New setpoint values were developed using new statistical selection methods approved by NRC. Operating plant data shows consistency with the new statistical selection methods. Unnecessary abnormal occurrence reports are avoided.

2.5 Control and Instrumentation Equipment

Issue - New design features in electrical, control and instrumentation systems should receive extra reviews, analyses and testing to assure continued high quality and avoidance of excessive downtime.

Response - Independent design reviews and extensive reliability analyses were carried out as well as qualification testing and, where practical, installation of prototype equipment in operating reactors.

Result - Operating data received from the field demonstrates the continuing high quality and reliability of the new systems.

2.6 Flow Control Valve

Issue - Reliability of the Flow Control Valves.

Response - A special task force was organized to review the preliminary design of the flow control valve and to recommend testing. A special GE test facility was built to perform life testing on the valve.

Result - Test results led to changes in the preliminary design. These valves have performed well in operating plants with normal periodic maintenance.

2.7 Safety/Relief Valve (SRV)

Issue - The introduction of a new SRV design employing a direct spring-loaded concept might impact plant availability.

Response - Extensive life cycle, seismic and environmental tests have been performed for the new valve. In addition, in-service experience was followed for two valves to assure leak tightness compatible with BWR operation.

Result - In-service performance of the valve to date demonstrates significantly improved leak tightness, which improves further after the initial fuel cycle. Performance to date has not reduced plant availability. All SRVs are checked and maintained periodically in accordance with applicable codes.

2.8 Main Steam Isolation Valve (MSIV)

Issue - Recommend construction of a test facility to develop ways to improve leak tightness of MSIV.

Response - An MSIV test facility was constructed. Tests and operating data identified parameters that affected leak tightness and design modifications were developed.

Results - Application of the MSIV improvements to thirteen operating plants has proven successful.

2.9 Jet Pumps

Issue - Thorough vibration testing of the new BWR/6 jet pumps to avoid failures from vibration.

Response - All BWR/6 jet pump models were tested in a new test facility. Vibration tests during plant startup are also conducted. In addition, installation procedures have been changed to reduce susceptibility to vibration.

Result - All BWR/6 jet pump data to date indicate jet pump vibration levels meet acceptance criteria.

2.10 Reactor Pressure Vessel (RPV)

Issue - RPV integrity under severe stress conditions such as 1) loss of coolant accident (LOCA) conditions, 2) anticipated transients without scram (ATWS), 3) ability to anneal non-jet pump RPVs due to radiation embrittlement

Response - LOCA and ATWS conditions were evaluated to reconfirm RPV integrity even under extreme conditions. The embrittlement of RPVs is monitored by dosimeters periodically removed and checked.

Results - Analyses reconfirm that reactor pressure vessel integrity is maintained for at least the 40-year design life.

2.11 Vessel Nozzles and Spargers

Issue - Cracking observed in RPV nozzle cladding.

Response - The cause (thermal cycling) of cracking was identified and design changes or repairs made. Cladding was removed to increase thermal cycling tolerance. Operational changes were made to reduce thermal cycling. Thermal cycling detection equipment was installed.

Result - Design changes and repairs were made in most plants. Experience to date shows no additional cracking at these plants. Improved ultrasonic inspection techniques are used to monitor for cracking.

2.12 Radioactive Waste Processing

Issue - Adequate operating feedback to assure that the level of effort being applied in the waste area is effective.

Response - The BWR/6 included features such as maximum recycle of treated water, improved evaporator, redundancy of equipment, a travelling-belt filter and programmable controllers.

Result - GE's current BWR/6 radwaste processing system provides an improved, more efficient design but GE maintains communication and feedback channels with customers on all issues, including radwaste.

2.13 Gaskets, Seals and Packing

Issue - Lifetime of gaskets, seals and packing in equipment such as control rod drives, valves and pumps.

Response - Field surveys of component lifetimes were conducted to assure satisfactory or at least predictable performance. In the case of the reactor water cleanup pump, design modifications, improved maintenance instructions and operating guidelines were issued.

Result - Service Information Letters (SILs) were issued recommending replacement intervals for equipment parts. Operating experience with reactor water cleanup pumps has improved. As of 1986 a new seal design is being offered to plant owners. Recirculation pumps have incorporated a seal purge, extending seal to four years from an earlier two-year interval.

2.14 Recirculation Bypass Valve

Issue - Since details of design of the bypass valve were not known at the time, thorough testing was recommended.

Response - It was concluded during the detailed design phase of the BWR/6 that the bypass system was not required, so it was deleted from the design.

2.15 Relief Valve Augmented Bypass (REVAB)

Issue - Effectiveness of REVAB.

Response - After further study the REVAB system was deleted from the BWR/6 design.

2.16 Control Rod Valve (CRD) Materials

Issue - Cracking in the CRD collet retainer tube.

Response - Extensive testing of cracked collet retainer tubes obtained from the field showed operability beyond six times the design life. Nevertheless in 1976 an improved design was put in service. Destructive test evaluations were made of BWR/6 CRD components. A new high strength material ASTM XM-19 was selected.

Result - The NRC has approved use of the new materials in CRDs. Operation of the new materials has been satisfactory. In the meantime monitoring of old in-service CRD continues to assure that life expectancy is not changing.

2.17 Pressure Control System (PCS)

Issue - PCS reliability.

Response - A prototype PCS test was performed in 1976. At each plant site the PCS is functionally tested. During plant startup the system operation is checked and calibrated with the lead systems engineer.

Result - The system has been performing well in operating BWR/6 plants.

3.0 Fuel and Core Improvements

3.1 Fuel Mechanical Design

Issue - Fuel design and analysis treatment of transients.

Response - Analysis and test programs have been conducted as recommended. In addition field experience continues to be monitored.

Result - The fuel design has proven to be adequate. Transients in operating reactors have not resulted in significant fuel cladding perforations.

3.2 Fuel Failures - Short Term

Issue - Fuel design and core management changes to minimize effect of fuel operating recommendations (PCIOMR) on capacity factor.

Response - Introduced fuel design improvements and operating methods such as control cell core, and improved basis for both fuel operating recommendations and procedures in addition to process computer use. Introduction of barrier fuel has eliminated the need for PCIOMR.

Result - Capacity factor losses due to fuel operating recommendations are significantly reduced through improved design core management and PCIOMR procedures for non-barrier fuel.

3.3 Fuel Failures - Long Term

Issue - Long term reduction in incidence of fuel failures.

Response - Mounted intensive effort to identify mechanism, understand it and minimize it. Introduced new designs including 8x8 lattice and barrier fuel as discussed in 3.2 above.

Result - Reliability of 2.4 million 8x8 fuel rods stands at 99.98%. (All causes.)

3.4 Fuel Manufacturing

Issue - Assure superior quality in manufactured fuel, maintain acceptable cost .

Response - Identified areas for improvement, intensified QA programs, improved manufacturing processes.

Result - Fuel reliability rate on first cycle fuel now exceeds 99.999% where failure cause originates in manufacturing.

3.5 New Fuel Failure Mechanisms

Issue - Surveillance recommended on fuel performance as bundle exposures increased and new mechanisms may appear.

Response - Maintain systematic program of lab tests, test reactor experimentation, surveillance, post irradiation examinations.

Result - Use of data acquired from surveillance and test programs forms the bases of the to overall reliability record of 99.98% on 8x8 fuel.

3.6 Fine Motion Control Rod Drives (FMCRD)

Issue - Recommended investigating potential usefulness of FMCRD in making fuel operating recommendations easier for utility to follow.

Response - Considered as a potential retrofit design change, testing program conducted.

Result - Program terminated. Success of other fuel programs and operating improvements made FMCRD unnecessary. (See Items 3.2 and 3.3.)

3.7 Fuel Test Facilities

Issue - Establish qualified facility including GE Test Reactor (GETR), Radioactive Materials Lab (RML) for continued fuel tests, surveillance.

Response - Decision was to pursue testing in available reactors in Norway and Sweden. RML staff and facilities judged adequate for most examinations. Contract facilities identified to supplement needs.

Result - Using alternative facilities, high quality and timely surveillance and tests of fuel have been conducted routinely.

3.8 Fuel Spacer

Issue - Concern expressed about seismic capability of 8x8 spacer design. Also recommended pursuing spacer design for 9x9 fuel.

Response - Seismic test program conducted to demonstrate seismic capability of 8x8 spacer. NRC has approved 8x8 fuel design and it is in use at operating reactors.

Result - Over 170 tests have demonstrated seismic adequacy of the 8x8 spacer. Success with the 8x8 program obviated the need for a 9x9 bundle.

4.0 Materials and Process Improvements

4.1 Stress Corrosion Cracking

Issue - Susceptibility of BWR materials/components to stress corrosion cracking.

Response - Actions were undertaken to understand the cause and to develop and implement appropriate countermeasures at operating plants and those under construction.

Result - The cause of cracking and suitable countermeasures have been identified. These have been and are continuing to be implemented in both operating plants and those under construction.

4.2 Materials Effort and Control of Applications

Issues - Materials engineering responsibilities are diffused in the organization, test facilities need augmentation, material specifications may need tightening, and there is a need to complete development changes before production.

Response - Fuel and structural materials engineering were consolidated into two components, test facilities were augmented, materials and processes specifications were upgraded to significantly strengthen and control materials requirements. Also, a "test before use" approach was adopted to allow qualification of development changes before implementation.

Result - Over 400 materials and processes specifications have been changed or issued. Improved procedures for control of materials application and implementation are in place. Materials effort was augmented and organizationally focused.

4.3 Corrosion Fatigue

Issue - Studies should determine if potential problems for the BWR in fatigue crack propagation, crevice corrosion, crud contribution from Inconel 600, weld sensitization, cyclic frequency and wave form should be considered in design.

Response - The identified issues were studied and appropriate responses were implemented including corrosion fatigue design rules and development of hydrogen water chemistry to minimize environmental effects.

Result - The potential concerns have been addressed and no open issues remain.

4.4 Weld Bead Straightening

Issue - Possible introduction of undesirable residual stresses by the practice of weld bead drawing or straightening applied to jet pumps.

Response - Weld bead straightening stress and resultant degree of sensitization were measured.

Result - The magnitude of the weld bead straightening stresses and degree of sensitization were found to be within the bounds of conventional weld joints. However, to provide added margin, controls for weld bead straightening were implemented in design documentation.

4.5 Fuel Channels

Issue - Possible control blade binding due to corrosion and creep of fuel channels during operation.

Response - Thicker walled channels were implemented using an improved manufacturing process and channel creep has been monitored. Also, improved corrosion resistance channel heat treatments have been implemented.

Result - Channel corrosion has been reduced and an improved understanding gained of the creep process. No control blade binding has been experienced.

4.6 Manufacturing Process Laboratory

Issue - Completion of process development and quality control laboratories for reactor equipment and fuel at the Wilmington site.

Response - The recommended laboratories were established and staffed with highly professional personnel.

Result - New processes have been developed for manufacturing control as well as for enhancements used for fuel and structural components manufactured by Wilmington.

4.7 Installation of Reactor Internals

Issue - Recommended policy that installation of reactor internals be completed in the shop rather than in the field.

Response - The in-shop installation was done for the Hanford 2 facility with apparent advantages prior to this recommendation. Recommendation to pursue this course was implemented.

Results - At the CBIN facility one additional BWR/5 and 11 BWR/6s were so fabricated.

5.0 Design Margin and Operability Improvements

5.1 Design Thermal Margins

Issues - The BWR/6 design was acknowledged to be in the preliminary design stage. Thus it was believed that design might require more thermal margins and operating flexibility to avoid forced power reductions to meet possible future licensing limits.

Response - During completion of core design, improvements were introduced and calculational models compared against operating data.

Result - Excess margin has been demonstrated for BWR/6 plants, permitting higher power level operation rther than power reductions.

5.2 Power Maneuvering

Issue - Increased load following capability in the BWR/6 and capability for grid frequency control beyond that acceptable to customers was recommended.

Response - A survey of utilities' needs was conducted in 1976-1977 and compared to BWR/6 capabilities.

Result - BWR/6 performance meets or exceeds utility requirements in all areas except local governor control which can be added if desired.

5.3 Fast Scram Drive

Issue - Recommended completion of fast scram drive development program already underway.

Response - Fast scram drives have been subjected to comprehensive tests and are considered fully qualified.

Result - Fast scram drives are operating in all BWR/6 plants and have accumulated 15 reactor years of satisfactory experience.

5.4 Feedwater/Level Contrcl

Issue - Capability of the feedwater Control System to prevent high or low water scram under certain conditions was questioned for plants with Relief Valve Augmented Bypass (REVAB) or 100% bypass.

Response - REVAB is no longer included as an option on BWR/6. Analysis and test have established that plants with 100% bypass are not subject to scram with full load rejection.

6.0 Product Service Improvements

6.1 Refueling Time

Issue - Recommended studies of procedures and tools to reduce refueling outage times, and thus improve availability.

Response - Improvements have been made in the refueling platform, the grapple, fuel transporting, quicker vessel head removal, faster fuel sipping, improved undervessel servicing equipment and construction of an $8.5 million refueling floor mockup facility to proof-test new tools and equipment.

Result - These improvements have streamlined outage tasks; however, total outage times have varied widely depending on many factors such as number of regulatory modifications implemented.

6.2 BWR Service

Issue - Recommended expansion of the existing program to offer product service to BWR owners.

Response - The GE product service program has substantially expanded and includes: AMPS, an advanced maintenance planning service tailored to each plant's specific needs; CIRS, a component information retrieval system providing statistics on operational performance; placement of GE senior engineers at each reactor site to identify and implement specific improvement programs, COPs, contingency operating programs to cover time when systems are out of operation, and many others such as annual conferences on technical problems, spare parts, computers and training all to share experience and new techniques.

Result - Plant availability is the result of many factors. The expanded GE product service program contributes to better performance and better availability.

6.3 Replacement Fuel

Issue - Recommended availability of emergency fuel supply for unscheduled plant shutdowns.

Response - Initially GE provided a contingency pool of fuel bundles for the eventuality cited in the recommendation. It became evident through operating experience that fuel failures did not cause unplanned shutdown so the contingency pool was discontinued.

Result - GE fuel performance in BWRs has been excellent. Normally planned refueling outages replace 20 to 30 percent of the fuel since it has produced its designed energy. Replacement of any failed fuel at this time can be accomplished with minimum effect on plant downtime.

6.4 Failed Fuel Detection

Issue - It was felt that equipment to more accurately identify failed fuel bundles quickly might be needed and a development program was recommended.

Response - An improved wet sipping system has been developed to check a fuel bundle's integrity more quickly. In addition a sampling for a discrete radioisotope concentration gives a more precise indication of a failed bundle. Finally, significantly improved fuel performance has led to a reduced number of fuel failures.

Result - Failed fuel detection is now performed with minimal effect on outage time. In addition, fuel performance has achieved a reliability level of 99.98%, reducing the need to detect failures.

7.0 Regulatory/Licensing

7.1 Anticipated Transients Without Scram (ATWS)

Issue - GE BWR response to new ATWS requirements of the NRC.

Response - A GE study submitted to the NRC in 1976 showed that the BWR scram system possesses high inherent reliability and since the probability of an ATWS was so low, the additional automatic boron injection system required by the NRC is not necessary.

Result - In resolution of the ATWS issue the NRC issued a ruling in 1984 requiring that the BWR have an automatic recirculation pump trip, an alternate rod insertion system and manual initiation of the liquid boron system in case of an ATWS.

7.2 Mark III Dynamic Loads

Issue - High priority was recommended for the program to determine hydrodynamic loads on the Mark III containment due to loss of coolant accident (LOCA) and safety relief valve (SRV) discharge.

Response - LOCA and SRV testing was completed in 1979 and a final loads report was issued in 1980.

Result - The NRC reviewed the Mark III LOCA and SRV hydrodynamic loads report and issued its findings. Overall NRC acceptance of Mark III containment is indicated by approval of GESSAR FDA and Grand Gulf, Perry, Clinton and Riverbend licenses.

7.3 Dynamic Buckling

Issue - If dynamic buckling behavior of the Mark III steel containment could not be satisfactorily modeled analytically then a test model should be used.

Response - GE has complied with the NRC interim criteria for evaluation for free standing steel containment buckling for plants undergoing licensing review.

Results - BWR/6 plants have complied with the NRC interim criteria for evaluation of free standing steel containment buckling. The issue is not applicable to the majority of Mark III containments which are reinforced concrete containments with steel liners.

7.4 Mark III Radiation Levels

Issue - Further study and analyses of the Mark III containment to assess personnel radiation exposure was recommended including possibly a cover for the suppression pool recommended.

Response - GE employed the A/E firm C.F. Braun to analyze radiation exposure. GE issued the results of this study and recommendations to its customers in 1978 for measures to add margin for this issue.

Results - Mark III plants have accumulated 15 reactor years of experience without loss of access to the containment and without increased personnel exposure. A suppression pool cover was not considered practical.

7.5 Plant Arrangement

Issue - Recommended evaluate plant arrangement for increased provision for security, sabotage prevention and handling of mixed oxide fuel in the BWR/6 Mark III containment.

Response - Studies of the specific procedures for personnel access into controlled areas of the plant, industrial security and sabotage protection have been considered by GE and discussed with the NRC, ACRS and A/Es. However, as specified in the governing regulatory guide the utility/owner is responsible for the unique plant security and sabotage requirements for his plant.

Result - GE's submittal to the NRC on the GESSAR docket in 1975 on design considerations for reducing sabotage risk has been reviewed and accepted by the NRC.

7.6 Mark I and Mark II Containment Loads

Issue - Additional analyses for the Mark I and Mark II containments to meet changing NRC requirements should be done quickly and planning to minimize plant unavailability due to any required retrofit work was recommended.

Response - GE established a centralized technical program management group to facilitate and control the program activities. In 1975 Mark I and Mark II owners established technical programs for addressing questions posed by the NRC.

Result - NRC acceptance of the work is evidenced in NRC Safety Evaluation of the Mark I Short Term Program in 1977 and the Long Term Program in 1980. In 1981 the NRC issued a regulatory guide addressing LOCA-related loadings and in 1982 a regulatory guide addressing SRV loads for the Mark II containment. Modifications required by NRC have been made or are in process.

7.7 NRC Submissions

Issue - Need to avoid unsubstantiated design requirements in licensing documentation.

Response - GE established a specific Safety and Licensing Management Group to ensure the quality, timeliness and completeness of material transmitted to the NRC. GE also established a system of approvals to ensure thorough review of submitted material.

Result - All responses to NRC have been satisfactory prior to licensing of each operating plant.

7.8 Anticipation of Regulatory Change

Issue - Systematic anticipation of future regulatory requirement was recommended.

Response - GE established a Safety and Licensing Program Manager responsible for regulatory guides and NRC/industry standards.

Result - BWR design is reviewed versus NRC regulatory guides to identify potential changes and incorporated where practical.

7.9 Period of Safety of Unattended Reactor

Issue - Determination and implementation of a logical basis for operator response time beyond the NRC 10-minute operator action time in use since 1973.

Response - GE BWR designs satisfy the industry standard on response times issued in 1984 by ANSI.

Result - Although 10-minute duration is the standard timing requirement, BWR plants are being modified to comply with additional new NRC rules that are more limiting.

7.10 Safety System Redundancy

Issue - Initiation of a GE study to meet the N-2 rule used by other countries was recommended. This involves assuming the two most critical pieces of emergency core cooling system (ECCS) equipment would be unavailable following a LOCA.

Response - NRC has maintained its position that the N-1 criterion (the most critical piece of equipment in the ECCS fails to operate upon demand <u>or</u> would be otherwise unavailable for use) is the appropriate conservative licensing basis. Because of NRC's position, the study was unnecessary.

Result - BWR plants meet all the NRC requirements based on the N-1 criterion.

7.11 Core Catcher

Issue - A study was recommended that answers the question, "Is a core catcher cost beneficial in reducing plant risk?"

Response - Evaluations performed by GE and others have shown that the addition of a core catcher to the BWR design was not cost effective in reducing plant risk in the event of a severe accident.

Result - NRC has never required a core catcher for a BWR plant to be licensed.

7.12 Fuel Transfer Accident

Issue - It was recommended that potential for one or two fuel bundles being stuck in the fuel transfer tube be studied.

Response - Contingency tooling has been designed and evaluated which is capable of removing the fuel assembly container and its cargo of one or two fuel assemblies. Extensive testing and Failure Modes and Effects Analysis were performed to verify reliability and safety adequacy of the design of the fuel transfer tube equipment.

Results - In 15 reactor years of BWR/6 operation no fuel bundles have become stuck in the fuel transfer tube under actual operating conditions.

7.13 Off-Site Radiation Exposure

Issue - A study to assure that off-site external radiation will be well below the anticipated new limit of 60 mrem per year at the site boundary.

Response - Radiation from plant equipment for a typical plant is about 10% of the regulatory limit.

Results - Plants in operation and new plant designs are complying with the dose limit requirements given in the EPA regulation issued in 1977.

7.14 Plutonium in Turbine

Issue - It was stated that plutonium had been detected on the inside of BWR turbines. Since its migration and concentration in the balance of plant was not well understood, it was recommended that this be studied.

Response - EPRI initiated a three-year program in 1976 to study this issue.

Result - In 1980 EPRI published their study results. Levels of plutonium detected in the turbine and calculated for the balance of plant were very low and do not constitute a health and safety concern. This is a closed issue with the NRC.

8.0 Engineering Quality and Cost Improvements

8.1 Engineering Manpower

Issue - Increased engineering manpower and clear-cut responsibility lines in specified areas was recommended.

Response - Engineering staff was increased by 14% from 1975 to 1978. The management system was modified and computer tracking implemented.

Result - Issues were resolved as planned, with more than 90% of engineering tasks completed on schedule.

8.2 Computer Facilities

Issue - It was recommended that computer capability be upgraded for automation of design applications and safety analyses.

Response - The computer facility was expanded and continually updated. The current facility has almost five times the capacity of that in 1975.

Result - The design process was automated, more sophisticated programs are operating, and the increased demands of regulatory bodies for expanded safety analysis have been met.

8.3 Procured Equipment Control

Issue - It was recommended that increased technical expertise oversee vendor supplied design and qualification of procured equipment.

Response - Reorganization and restaffing of engineering and quality functions allowed greater involvement of QA in procurement activities of vendors.

Result - Improved performance of procured products has been achieved.

8.4 Standardization

Issue - It was recommended that increased resources be applied to BWR/6 standardization.

Response - An organizational component was established to emphasize standardization.

Result - The attention of management and engineering was focussed on standardization during the BWR/6 final design period. The change control program was emphasized.

8.5 Test Facilities

Issue - It was recommended that additional test facilities test system components prior to installation to ensure improved reliability.

Response - Major testing facilities and programs were developed and completed: High Flow Hydraulics, Fuel Transfer, Control Rod Drive handling, Control Rod Drive Test, Recirculation Flow Control Valve, Main Steam Isolation Valve, Safety/Relief Valve, Reactor Water Cleanup Pumps, Pipe Test, Containment Test, and BWR Services Training.

Result - The equipment designs were tested and verified prior to use.

8.6 Quality Assurance

Issue - Increased staffing and organizational status of QA in engineering, communications, procedures etc. was recommended to handle increased regulatory demands.

Response - Quality functions were reorganized, reporting level and staffing increased, and procedures were revised and implemented.

Result - An upgraded quality program was implemented throughout the engineering functions.

8.7 Systems Design Organization

Issue - Strengthening of the systems engineering function to accommodate design reviews was recommended.

Response - A lead system engineer was assigned to each functional subsystem with responsibility for design and performance.

Result - A Composite Office of Lead System Engineers provided strong guiding influence on BWR design.

8.8 Field Experience of Design Engineers

Issue - Improved training of design engineers through field operations assignments was recommended.

Response - The use of engineers at plant sites was increased and communications between operating and design engineers were improved.

Result - Improved design knowledge in field and familiarity with field issues and personnel within the design group was achieved.

8.9 Reliability Program

Issue - It was recommended that a dedicated program to improve availability and capacity factor be established.

Response - Each area of concern was addressed; a program manager was assigned to each major activity, test facilities and programs were implemented, computerized data storage and retrieval was begun; and a formalized organization was put in place.

Result - The capacity factor of BWRs has increased significantly.

8.10 External Support of Development

Issue - Promotion of R & D programs with other organizations was recommended.

Response - Programs with DOE, EPRI, NRC, BWR Owners' Groups, and GE licensees were given added emphasis.

Result - Increased support for BWR development was achieved. A BWR Development Board was established with GE licensees, BWR owners' groups and with utilities.

8.11 Auxiliary Power System Integration

Issue - It was recommended that GE increase communication of responsibility and interface with customer and A/E in the area of auxiliary power systems to assure high reliability.

Response - Responsibility for power supply for GE systems was centralized organizationally in GE. Exchange of information assured coordination of the shared responsibility with customer/AE.

Result - Improved information exchange ensured the incorporation of latest requirements and improvements.

8.12 Fuel Cycle Cost and Fuel Performance

Issue - Approaches to fuel cycle cost reduction and improved fuel performance were identified for recommended study.

Response - The specific approaches were reviewed but each was rejected.

Result - Preconditioning by use of PCIOMR resolved the concern of fuel shuffling. No utility interest in going to 6 month fuel cycles has been expressed. Use of discharged fuel bundles as a blanket proved to be impractical.

8.13 Engineering/Manufacturing Communication

Issue - Geographic separation of Engineering from Manufacturing was an area recommended for study to determine the best personnel division.

Response - Alternatives were studied, and improvement in communications and organizational reporting relationships changed in lieu of relocation of people.

Result - Follow up of actions taken led to the conclusion that adequate liaison was established.

9.0 Project Management Effectiveness

9.1 Control and Instrumentation (C&I) Field Changes

Issue - The large number of C&I changes at sites signalled a need for improved procedures applicable prior to shipment of equipment.

Response - Programs were established and implemented to address these concerns.

Result - A number of programs were established to address the concerns identified in the Reed Report. These programs were effective in reducing C&I field changes in both design and procurement areas and streamlining the documentation process for necessary changes. Programs were developed to further improve implementation and tracking of necessary changes.

9.2 Productivity at Reactor Site

Issue - Areas were recommended for study that would improve site productivity of the combined contractor-A/E-utility-GE team

Response - Although GE's role at construction sites was quite limited compared to the utility and principal constructor(s), GE took steps to improve efficiency within its scope of responsibility.

Result - GE responsibility for construction site work has been concluded at all but one U.S. site.

9.3 Cofrentes Construction -- U.S. Design Schedule

Issue - The Cofrentes project schedule followed closely behind the standard BWR/6 design schedule such that unresolved U.S. licensing issues might affect the project schedule

Response - GE successfully supported the construction schedule of the Cofrentes plant.

Result - The Cofrentes plant went into commercial operation in 1985. Changes required including post-TMI requirements issued by NRC for U.S. plants such as emergency response information system, post-accident sampling and SRV position indicators have also been included in the plant.

9.4 Technical Direction of Information Scope

Issue - It was recommended that ambiguity in contract language for GE scope of technical direction be clarified.

Response - Contract language for GE technical direction is supplemented by a formal presentation by GE to the utility staff one year prior to start of work.

Result - There have been no significant difficulties over the scope of technical direction to be provided by GE. Such scope of work is complete for all U.S. construction sites except one.

9.5 Architect Engineer (A/E) Information Input

Issue - GE's access to A/E drawings and schedules was inadequate and it was recommended that this be clarified in contracts.

Response - Required A/E information is contractually covered and provided through a set of controlled documents. This is supplemented with kickoff meetings that identify needs at key stages in the project schedule (e.g., start of PSAR, FSAR, etc.).

Result - A system was successfully implemented which assured that GE had adequate access to A/E drawings and schedules. All GE-required A/E information for U.S. construction plants is complete except for one site.

9.6 Project Management Information System

Issue - Expedited completion was recommended for a planned streamlining of the current (1975) project information system.

Response - The basic elements of the Nuclear Program Control (NPC) computer system were put in place in 1977. Overnight updates replaced biweekly updates. This provided the needed "real-time" measurement.

Result - All GE project management uses the NPC system for construction projects.

9.7 Control and Instrumentation (C&I) Manufacturing Space

Issue - Increased space requirements for C&I manufacturing at the home office in San Jose (1975).

Response - A 100,000 square foot facility was leased in San Jose in 1976 for C&I manufacturing. This was expanded to 150,000 square feet by 1978. All of this was retired upon completion of the backlog.

Result - The manufacturing space was acquired and the work completed.

9.8 Backlog Plants

Issue - It was recommended that GE renegotiate backlog orders to incorporate standard plant features on a block basis.

Response - Limited changes were negotiated to simplify commitments and assure the reliability of supplied equipment.

Result - It was recommended that GE examine opportunities and cost benefits for renegotiating backlog orders in order to simplify commitments or to incorporate standard plant features on a block basis.

9.9 Change Control

Issue - The Change Control Board had not been operational long enough (in 1975) to prove its effectiveness.

Response - Continued use and monitoring of the Change Control Board took place with time.

Result - The Change Control Board proved to be an effective means of monitoring and authorizing changes.

9.10 Investment Emphasis

Issue - Investments to improve availability/capacity factor were recommended.

Response - Investments in engineering test facilities emphasize improved product performance. Further investment at Wilmington Manufacturing Lab focused on fuel quality.

Result - GE made substantial investments in engineering facilities and training facilities aimed at availability/capability improvements (see Items 6.2 and 8.5).

10.0 Future Product Offering

Issue - Several recommendations were made regarding future product offerings

Response - In 1978 an Advanced Engineering Team was formed to work with other BWR suppliers around the world in a feasibility study.

Result - Following joint development efforts, Tokyo Electric Power Company announced plans in 1987 to proceed with two advanced boiling water reactors (ABWR) at the Kashiwazake site in Japan.

UPDATE REPORT

1987 ANALYSES OF

FINDINGS AND RECOMMENDATIONS

IN

NUCLEAR REACTOR STUDY
(REED REPORT)

1.0 TECHNOLOGY AND DESIGN METHOD IMPROVEMENTS

1.1 DESIGN METHODS

Summary of Findings and Recommendations in Reed Report - 1975

The Reed task force noted the desirability of better verification of design methods in the area of coupling of nuclear and thermal-hydraulic processes. Rapidly evolving nuclear design methods were being continuously improved to correlate predicted and operating field data more closely and to strengthen calculational methods. Qualification and verification of these improved methods were underway, but were not yet complete. While method improvements had resulted from operating plant feedback, an increased base of coupled nuclear and thermal-hydraulic data was desirable for qualification of the analytical models used in BWR design. Tests involving more extreme conditions in operating plants and comparisons with multidimensional nuclear/thermal/hydraulic transient methods were desirable to better verify the adequacy of the design methods in use. The task force was concerned that plants might have to operate at power 5% to 10% less than rated (known as derating power) if this verification work did not demonstrate the design margins were adequate.

Update - 1987

Conclusion:

The design methods proved to be adequate and no derates resulted. An improved nuclear/thermal hydraulic transient model was also developed using actual plant performance data. The model has been documented and approved by the NRC.

Summary:

After the Reed Report was issued, GE continued its programs to further develop its base of coupled nuclear/thermal/hydraulic data from actual operating plant systems. This data involves three general categories: steady state core power distribution and critical eigenvalue, system response to pressurization transients, and system hydrodynamic stability characteristics. The data provided the required basis for further qualification and verification of already existing analytical models and for the development of more highly refined calculations.

Supporting Information:

The primary sources of the steady state core power distribution and critical eigenvalue data are process computer output during plant operation, gamma scans at refueling outages and destructive examination of exposed fuel rods to determine isotopic distribution. Large volumes of power distribution and critical eigenvalue data were acquired from the process computer as the natural result of the core management responsibility of approximately thirty operating plants. End of cycle power distribution was quantified using gamma scan techniques.

Four plant system responses to pressurization transients have been recorded in two plants: Peach Bottom 2 and KKM. These tests were conducted to assure a positive core power response during the pressurization transient. Specially installed instrumentation and a high-speed digital data acquisition system provided an extensive, accurate data set which was used as the primary source of qualification of a recently developed, improved plant transient model. This improved model was submitted to the U.S. NRC and approved.

Stability tests were performed in the Peach Bottom 2 reactor during 1977 and 1978. Later on, similar tests were performed at Vermont Yankee in 1981, at Caorso in 1983, and at Leibstadt in 1984. These tests included the use of special plant instrumentation which permitted the accurate determination of the system decay ratio when subjected to pressure perturbations. These data were used to qualify the analytical model. The model is now being reformulated to further improve the accuracy of the simulation of the coupled nuclear/thermal/hydraulic phenomena which control the system response under these conditions.

References

1. EPRI NP-564, "Transient and Stability Tests at Peach Bottom Unit 2 at EOC2," June 1978.

2. NEDE-24154-P, Licensing Topical Report, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volume 3, October 1978.

3. USNRC Safety Evaluation and Supplemental Safety Evaluation for the GE Topical Report, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDE-24154P, June 1980.

4. NEDE-25445P, "Vermont Yankee Cycle 8 Stability and Recirculation Pump Trip Report," August 1982.

5. NEDE-22277P-1, "Compliance of the GE BWR Fuel Designs to Stability Criteria," August 1984.

6. Letter, C.D. Thomas (USNRC) to H.C. Pfefferlen (GE), Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, "Thermal Hydraulic Stability Amendment to GESTAR II," April 24, 1985.

1.0 TECHNOLOGY AND DESIGN METHOD IMPROVEMENTS

1.2 EMERGENCY CORE COOLING SYSTEM ANALYSIS

Summary of Findings and Recommendations in Reed Report - 1975

Because emergency core cooling systems (ECCS) computer programs were still under development to meet new NRC requirements, the NRC had taken a conservative position which resulted in plants using the BWR/3 design being required to derate 5%. Analysis of new information about the behavior of the core spray in a steam environment was considered necessary, as well as further evaluation of the conservatism of the fission gas release calculations.

Update - 1987

Conclusion:

GE successfully developed qualified computer models and BWR/6 plants were never adversely impacted. The impact on earlier designs caused by this regulatory change was alleviated through the development by GE of new, qualified models, which NRC approved.

Summary:

In late 1973 the NRC issued a rule which dramatically affected the design of ECCS systems. The industry was required to revise designs or to demonstrate that existing designs met new requirements. Until verification work was completed for plants in operation, the NRC imposed conservatisms to ensure that plants in operation would satisfy the new rule. This resulted in the derate for BWR/3 plants noted in the Reed Report. No other plants, including those which used the BWR/6 design, were required to derate. The derating of BWR/3 plants was lifted by the NRC once new models were qualified and approved.

Core spray distribution test data and core spray cooling test data developed by GE were approved by NRC in 1974. GE's Evaluation Models to perform ECCS Analysis were reviewed and formally approved by the NRC in 1982. A substantial test program was developed and completed to expand the data base. Based on physical phenomena revealed by these experimental programs, a new computer code, SAFER/GESTR was developed, which demonstrated significant margin (approximately 1000 degrees F) to previously calculated peak cladding temperatures (approximately 2200 degrees F) using evaluation models. This new code was approved by the NRC on June 1, 1984.

Supporting Information

The test programs on Large and Small break blowdown heat transfer, single nozzle core spray distribution in steam, full scale core spray distribution in air using steam simulators, and full scale core spray distribution in 30 degree sector tests in steam environment were completed to confirm adequacy of GE's Core Spray Methodology. The test program to study upper plenum behavior and system interactions using large scale test facility with 58 simulated bundles was also completed. The refill/reflood phenomena revealed by these test programs were incorporated to develop a realistic ECCS analytical model, SAFER/GESTR, which also resolved the concern related to fission gas release. This new model demonstrates significant margin to allowed peak cladding temperature of 2200 degrees F, and has been qualified against a more detailed 3-D computer program, TRAC, for ECCS analysis.

<u>References</u>

1. General Electric Company Topical Report, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss of Coolant Accident," NEDE-23785-1-P (Proprietary); Volume 1, "GESTR/LOCA - A Model for the Prediction of Fuel Rod Thermal Performance," December, 1981; Volume 2, "SAFER - Long Term Inventory Model for BWR Loss-of-Coolant Analysis," December, 1981; Volume 3, "SAFER/GESTR Application Methodology," March 23, 1984.

1.0 TECHNOLOGY AND DESIGN METHOD IMPROVEMENTS

1.3 SEISMIC DESIGN

Summary of Findings and Recommendations in Reed Report - 1975

As the time of the Reed Report, the NRC had significantly changed seismic design requirements. The task force recognized the need to closely integrate the reactor design performed by GE with the Balance-of-Plant (BOP) design performed by the architect/engineer. The task force was concerned that the lack of an integrated response to these new requirements might result in major redesigns and possible retrofitting of plants already constructed. In line with this concern, it was recommended that the seismic design function within GE be centralized and upgraded to a subsection level.

Update - 1987

Conclusion:

GE and architect engineers successfully worked together to avoid adverse impacts which might otherwise have resulted from this regulatory change.

Summary:

General Electric worked closely with the architect engineers to incorporate the new seismic design requirements into plant design. Within GE, the recommendation to centralize seismic design was followed by placing responsibility in one subsection for methods, criteria, and computer program development with a second subsection having the responsibility for application of the methods and criteria to requisition plants and for interfacing with the architect engineer in matters regarding the accuracy and completeness of seismic design information they supplied to GE. This assured that seismic methods and procedures were maintained at current state-of-the-art levels within the requirements of applicable codes and regulations, and that seismic requirements were properly identified, analyzed and applied for equipment evaluation.

1.0 TECHNOLOGY AND DESIGN METHOD IMPROVEMENTS

1.4 FLOW INDUCED VIBRATION

Summary of Findings and Recommendations in Reed Report - 1975

The Reed task force noted the concern of GE's engineers that a phenomenon known as flow induced vibration had to be addressed in the design of components for the BWR/5 and 6 reactors, but that field experience in prior designs suggested that reliance on analytical methods had not proven to be entirely satisfactory. The phenomenon had resulted in some failures of components in operating plants and the engineers wanted to be sure that design margins for future designs were adequate. A testing facility had been conceived and was proposed as a means of providing further assurance that such failures could be avoided. The task force endorsed this recommendation.

Update - 1987

Conclusion:

Testing confirmed BWR/5 and 6 design margins were adequate.

Summary:

In 1975 General Electric management approved the design and construction of the High Flow Hydraulic Test Facility. This facility, which consisted of a full-size, sector model of the BWR/6 and a general purpose test tank for large scale components and models, was dedicated solely to the flow induced vibration testing of BWR reactor internals. Vibration tests were monitored by a special on-line computer system which could detect and evaluate vibration encountered during tests run at flow rates significantly higher than encountered during plant operation. Initial operation of this facility began in May 1978 with full-scale testing of four prototype BWR/6 jet pumps. From mid-1979 until early 1980, full-size flow tests of the BWR/6 lower plenum components were performed in the sector model. All flow testing of major BWR reactor internal components resulted in confirmation of design margins against flow induced vibration.

1.0 TECHNOLOGY AND DESIGN METHOD IMPROVEMENTS

1.5 PLANT RADIATION

Summary of Findings and Recommendations in Reed Report - 1975

At the time of the Reed Report, experience from operating plants showed that the buildup of radiation levels in plant components was increasing with operating time in all reactor types, including BWRs. The Reed study group noted the concern that this buildup could impede plant maintenance and service and require time consuming and costly decontamination. It was suggested that programs be initiated to address this issue.

Update - 1987

Conclusion:

Significant advancements have been made in understanding radiation buildup in operating plants and minimizing its impact on maintenance and servicing of these plants.

Summary:

Since the time of the Reed Report, GE and others in the industry have focused considerable effort in developing better tools, plans and procedures to minimize the impact of radiation levels on maintenance operations. To date, radiation buildup has not impaired plant operation, nor has it resulted in undue exposure of plant maintenance personnel, which is controlled by NRC regulation and plant procedures. The focus of ongoing programs in the industry is to reduce exposure and make maintenance easier.

General Electric has had a joint program with the Electric Power Research Institute (EPRI) since 1976 to assess and control the buildup of radiation in boiling water reactors (BWRs). This program is aimed at assessing the extent of the radiation buildup with time in operating BWRs, identifying the corrosion product precursor sources, understanding the activation and transport of the radionuclides, and suggesting operating techniques or design changes to be implemented into new or operating BWRs.

Considerable effort has also been directed at developing a better understanding of the process by which radiation buildup occurs. A significant data base has been developed from both domestic operating plants and those in Europe and Japan. This data base includes: (1) radiation buildup rates, (2) water chemistry history, (3) fuel deposit composition, (4) corrosion film composition, and (5) man rem exposure. Laboratory experiments have been completed which studied the cobalt release rates from cobalt base alloys.

The primary conclusion from studies to date is that radiation buildup on reactor piping occurs because radioactive isotopes, particularly Cobalt-60, are incorporated into the oxide corrosion film of the stainless steel piping materials. Emphasis on reducing radiation buildup is now focused on a) methods to reduce the corrosion film's capability to incorporate Co-60, or b) to reduce the inventory of Co-60 in reactor water.

Supporting Information:

The huge data base generated on radiation buildup behavior of operating reactors has led to several possible methods to control that phenomenon. Emphasis has now shifted from gathering more plant data to running controlled experiments in the laboratory where variables can be carefully controlled.

The first set of experiments, co-sponsored by EPRI, quantified the release rate of cobalt from the cobalt base alloy Stellite. These tests showed that a change from Stellite to a non-cobalt base alloy for control rod rollers and pins would be effective in reducing Co-60 coolant inventory. Concurrently, a joint EPRI/GE in-reactor demonstration program has shown that a PH 13-8M pin/Alloy X-750 roller combination works well.

In another program, co-sponsored by EPRI, an analysis of plant data indicated that trace amounts of zinc (10 ppb) in reactor water were acting as a corrosion inhibitor for stainless steel. Subsequent tests in the laboratory confirmed that low concentrations of zinc ions do indeed act as a corrosion inhibitor. A comprehensive testing program is underway to qualify this technique for reactor application.

The role of other plant chemistry variables on radiation buildup (pH, conductivity, specific impurity ions, and hydrogen water chemistry) are also being studied under a joint EPRI/GE program which has the objective of identifying plant chemistry parameters that can be controlled to positively affect buildup rates.

1.0 TECHNOLOGY AND DESIGN METHOD IMPROVEMENTS

1.6 NONDESTRUCTIVE TESTING

Summary of Findings and Recommendations in Reed Report - 1975

At the time of the Reed Report, nondestructive testing (NDT) efforts were diffused organizationally. It was recommended that the NDT effort be consolidated and strengthened.

Update - 1987

Conclusion:

The major NDT function has been strengthened, consolidated, and is managed by an NDT professional.

Summary:

The Reed study group noted the importance of NDT to critical plant components. At the time of the report there were separate NDT efforts in Development Engineering and in Design Engineering. In August of 1976 these two groups were combined to form the Nondestructive Test Engineering Unit of the Nuclear Technology Department. In July of 1979 the unit was upgraded to subsection level and currently, this subsection reports to the Plant Technology Section of the Nuclear Power Systems Engineering Department.

Reevaluation of NDT functions has resulted in the conclusion that not all NDT functions should be consolidated, such as field services and fuel inspections. Field Services are a commercial business venture offered to the utilities so that they can more readily meet their code requirements. Fuel inspections, on the other hand, are highly specialized and make use of only a few restricted NDT techniques. Quality control functions have also been kept separate, since these people primarily witness NDT efforts at vendors' plants and do not perform actual "hands on" examinations.

Primary efforts of the NDT component have focused on development and qualification of equipment and procedures for ultrasonic inspection of piping, the reactor pressure vessel, and reactor components. In addition, NDT personnel have performed site inspections and acted as third party consultants in cases where the utilities required special independent verification.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.1 CORE INTERNALS

Summary of Findings and Recommendations in Reed Report - 1975

It was postulated that radiation embrittlement and corrosion might limit the life of reactor internals to something less than the desired 40 years. Concern was specifically directed to core support structures which are made of stainless steel and receive more radiation exposure than other reactor internals. It was recommended that the design of removable reactor internals be considered in the event replacement should become necessary.

Update - 1987

Conclusion:

Evaluations of experimental data on materials subject to irradiation show that BWR/6 internal components should last their design lifetime. Notwithstanding this evaluation, BWR/6 reactor internals have been designed to be removable.

Summary:

The reactor internals (with the exception of fuel and control rods which are discussed elsewhere) were evaluated to determine if there were areas where the radiation effects on the materials used could shorten the component lifetime below the design life of 40 years. Under high radiation for long periods of time, stainless steel loses some of its ductility, but increases in strength. At the highest radiation levels, the present design assures that low stress levels are maintained for all normal operating conditions and that the stresses are conservative for even the most severe abnormal conditions, particularly in the weld areas. Analyses performed have considered the lower ductility, but have not taken credit for the higher strength. These evaluations determined that the 40-year design life of the reactor internals was not affected by radiation embrittlement and corrosion.

Supporting Information:

Regarding corrosion, the design of the core support structure keeps the long-term stresses to relatively low values, uses types of stainless steel that are relatively insensitive, and controls the number and shape of crevices. In addition, fabrication is closely monitored to control sensitization of the materials. A technical audit of the core support structure was performed in 1978 and determined that detrimental corrosion is unlikely.

Since the 1978 audit, several occurrences of stress corrosion cracking have been found in highly irradiated nonsensitized stainless steel components such as removable in-core instrument tubes and control rod blades. Subsequent studies have identified this cracking as irradiation assisted stress corrosion cracking (IASCC). While IASCC can limit the design life of these components, it does not represent a safety problem.

This issue of IASCC is also relevant to highly irradiated core internal structural components which experience fluences beyond the cracking threshold level. For such low stressed components, the fluence threshold is approximately $2x10^{21}$ nvt (E>1meV). In the BWR, the highest irradiated core structural component is the top guide. For the BWR/6 design the end life fluence is lower than earlier designs and just approaches the $2x10^{21}$ threshold in 40 years. At this fluence it is extremely unlikely that IASCC would occur. In fact, special in-service examination of an earlier design top guide with fluence at approximately $3x10^{21}$ showed no cracking. However, the top guide is a component that is subject to periodic in-service inspection, thus potential IASCC will continue to be monitored.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.? CONTROL BLADE LIFE

Summary of Findings and Recommendations in Reed Report - 1975

Routine destructive examination of a control rod which had been used in an operating reactor for some time revealed a few cracks in the absorber rods. These cracks were assessed to be caused by stress corrosion cracking of the tubing material. Although determined not to be a safety concern, a three-to-five year program to develop methods to extend control rod life was recommended.

Update - 1987

Conclusion:

Detailed evaluation of operating plant control blades has been used to predict blade lifetime at operating plants. In addition, a new control rod blade using hafnium has been developed to provide a longer life expectancy.

Summary

A model has been developed which correlates cracking and B_4C loss with absorber depletion. Identification of this cracking mechanism has resulted in new life limiting criteria. All operating plants have been informed of these findings and are now limiting blade life according to these recommendations. Recently a new control rod has been developed with hafnium rods replacing B_4C rods in the positions most likely to be limiting. These blades are available as replacement blades at the end of life of the old B_4C blades.

Supporting Information

Programs were developed to obtain additional data on control rods in service. Using this data, methods were developed for predicting control rod life.

Detailed metallographic examinations were completed at GE's Vallecitos Nuclear Center on two control rods from two operating plants. Additional results were obtained from a foreign BWR control rod examined in a European hot cell. Results showed some cracking and boron carbide (B_4C) loss in control rods examined, but not to an extent which would cause a degradation in plant safety.

Blade lifetime has been reported in a General Electric Report "Control Blade Lifetime Evaluations Accounting for Potential Loss of B_4C" by K. W. Brayman and K. W. Cook, NEDO 24232, January 1980, Class I. The report states, "The B_4C loss is a slow, predictable process which has been modeled and accounted for in control blade lifetime evaluations."

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.3 CONTROL BLADE TOLERANCES

Summary of Findings and Recommendations in Reed Report - 1975

During the initial phases of BWR/5 control blade production, there was concern over the ability to maintain adequate control of tolerances during the manufacture of control blades. It was also noted that excessive scrap or rework might result if drawing requirements were not met during the initial production cycle.

Update - 1987

Conclusion:

Current experience as measured by quality checks and performance feedback demonstrates that these products meet manufacturing tolerances and other requirements. Excessive scrap and rework were avoided.

Summary:

Tolerance problems encountered in the manufacturing process have been resolved by improving handling and forming techniques of thin wall piece parts and by improving the sequence of the welding operations, thus minimizing the distortion of thin wall parts.

Supporting Information:

Inspections performed prior to shipment demonstrate that BWR/5 control blade production has few envelope deviations and thus minimal rework or scrap.

In addition, the control rod was redesigned for BWR/6 application and further actions were taken to improve manufacturing productivity. As a result BWR/6 control rod production has been virtually free of the early BWR/5 problems. The envelope requirements for BWR/6 are tighter than those for BWR/5 and have been met consistently during manufacturing.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.4 SETPOINT DRIFT

Summary of Findings and Recommendations in Reed Report - 1975

Unplanned changes in the setpoints of BWR protective circuitry caused the setpoints to be out of compliance with plant specifications resulting in an unnecessarily high frequency of abnormal occurrence reports to the Nuclear Regulatory Commission. It was found that, in many cases, the technical specifications were more conservative than necessary and that the comparatively high occurrence of setpoint drift was a source of extra maintenance work and an inconvenience to plant operators. Study and resolution of this problem area was recommended.

Update - 1987

Conclusion:

New setpoint values were developed using new statistical selection methods approved by NRC. Operating plant data shows consistency with the new statistical selection methods. Unnecessary abnormal occurrence reports are avoided.

Summary:

GE began a program to develop an improved methodology for calculating instrument setpoints. As part of this Instrument Setpoint Methodology Program an explicit statistically based calculational method was developed for defining setpoints. This methodology includes explicit methods of determining expected drift based both on manufacturer supplied performance data and field data. In addition, this methodology includes a requirement to examine the statistically expected combination of drift, instrument loop inaccuracies, and calibration errors and ensure that the margin between the nominal trip setpoint and the technical specification allowable value is sufficient. Finally, actual field data has been collected from several operating plants for reactor protection system instrumentation (Rosemount devices or equal) and a statistical evaluation performed which demonstrates that the field data shows performance statistically consistent with the assumptions of the GE Improved Setpoint Methodology.

Supporting Information:

The GE Improved Setpoint Methodology was approved by NRC. The methodology has been used to check the conservatism of reactor protection system setpoints for the power plants recently licensed in the US. The GE Improved Setpoint Methodology is also being applied to current programs where GE is responsible for providing revised setpoints to state-of-the-art standards, such as the GE Equipment Qualification, ATWS and ABWR programs.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.5 CONTROL AND INSTRUMENTATION EQUIPMENT

Summary of Findings and Recommendations in Reed Report - 1975

Historically the electrical, control and instrumentation systems had not been a major contributor to BWR plant down-time. Several new developments were being introduced into the BWR product line and it was urged that extra design review, analysis and testing be performed to assure continuing high quality in the new equipment performance.

Update - 1987

Conclusion:

Operating data received from the field demonstrates the continuing high quality and reliability of the new systems.

Summary:

Significant developments were underway at the time of the Reed Report in the solid state safety system, the neutron monitoring system and improved control room equipment known as Nuclenet. Progress of these designs was closely monitored, including through a number of independent design reviews held on both the solid state safety system and Nuclenet. An extensive reliability analysis was carried out for the entire solid state safety system. Development and qualification testing has been performed. In the case of the improved in-core sensors for the neutron monitoring system, production units were installed in operating reactors to verify their capability.

A dedicated reliability and design review function was established within the Control and Instrumentation Engineering organization to provide consultation on reliability and availability and to organize the design review effort.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.6 FLOW CONTROL VALVE

Summary of Findings and Recommendations in Reed Report - 1975

Because the size of the BWR/5 and 6 flow control valves put them beyond the range of experience for valves in similar applications, it was recommended that a task force conduct a design review of both flow control valve designs. It was also recommended that more testing be performed on production valves before operation.

Update - 1987

Conclusion:

Test results led to changes in the preliminary designs. The final design has performed well with normal periodic maintenance in 13 operating plants.

Summary:

A special task force was organized in 1975 to perform a detailed design review of the control valves and to identify areas for further testing. As a result of the design review and tests, the preliminary design of the flow control valve was modified to improve component reliability.

Testing was completed on the final design and control system in 1981. These tests were conducted in a specially developed GE test facility which simulated actual system fluid loading on the valve using a mechanical ball loader. The facility provided prototypical water chemistry, pressure and temperature. All valve models successfully met their system performance requirements. In addition, design changes to the associated control system reduced the duty cycle on these valves in the load following mode by a factor of six.

As of June 1987, thirteen operating plants are using these flow control valves. The valves have performed well with normal periodic maintenance.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.7 SAFETY/RELIEF VALVE

Summary of Findings and Recommendations in Reed Report - 1975

There was a concern that, with the introduction of direct spring-loaded safety/relief valves (SRV) in BWR/5 and 6, frequent SRV blowdown would require additional maintenance to assure leak tightness of the valve, contributing to plant unavailability. It was recommended that, to relieve the duty cycle on the lowest SRV setpoint valve, an evaluation be performed to establish the feasibility of adding two or three additional steam bypass valves in parallel with the SRVs.

Update - 1987

Conclusion:

In-service performance of the valve to date demonstrates significantly improved leak tightness, which improves further after the initial fuel cycle. Performance to date has not reduced plant availability. All SRVs are checked and maintained periodically in accordance with applicable codes.

Summary:

To preclude the need for adding two or three additional non-safety grade steam bypass valves into the system, product improvements were made and incorporated into the design of the direct-acting safety/relief valves (Crosby and Dikkers) to extend cyclic versus inherent SRV leakage (leak tightness) performance. Also, on BWR/6, in order to limit system repressurization peaks, the designated low-low set (LLS) safety relief valve (SRV) can be alternated to another SRV after each outage, thus further reducing duty (cycle) on any specific LLS designed SRV as well as challenges to the other SRVs installed.

Supporting Information

Extensive life cycle, seismic and environmental qualification tests were performed and evaluated for the SRVs. Early in-service experience was obtained on two of these new SRVs which confirmed that the inherent leak tightness improvements made to the SRV were compatible with BWR system operation. Several of these valves have operated in plants for up to six years and have demonstrated significantly improved leak tightness. In addition, SRV simmer margins (difference between operating pressure and the spring setpoint of the valve) have been increased to further enhance SRV leak tightness capability. Observations of actual in-service SRV leak tightness performance indicate that leak tightness has improved after the first plant cycle. Based on current ASME Section XI requirements, all installed SRVs are checked, maintained and verified for operability at least once every five (5) years.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.8 MAIN STEAM ISOLATION VALVE

Summary of Findings and Recommendations in Reed Report - 1975

Main steam isolation valves (MSIV) had historically required considerable maintenance to assure they would continue to meet leak tightness specifications. It was recommended that a test facility be developed and a program implemented to identify ways to improve the leak tightness of these valves.

Update - 1987

Conclusion:

Application of MSIV improvements to thirteen operating plants has proven beneficial.

Summary:

An MSIV test facility was established to investigate the causes of valve seat leakage, determine corrective action and develop improved maintenance procedures and techniques. Programs were also conducted in cooperation with the valve vendors to provide leak tightness improvement packages that could be installed during normal valve maintenance periods.

Supporting Information

- A leak tightness program was developed to determine the parameters which affect the ability of the valve to seal. In this program a full size valve was tested in the valve facility. Tools for improved maintenance procedures and techniques were also developed and tested. Findings on ways to improve MSIV maintenance were published and conveyed to the BWR owners under INPO auspices.

- GE worked with Atwood and Morrill Company, the supplier of over forty percent of the MSIVs to jointly develop an advanced leakage improvement package. The improvements include a poppet guidance system, and an anti-rotation locking feature. In two plants with sufficient operating experience with these design improvement features incorporated into the MSIV, the leak tightness success rate improved significantly. These features are now installed in seven reactors.

- GE also developed a leakage improvement package with Rockwell International Corporation, the other major MSIV supplier. The package included improved packing, improved guidance, and improved connections. Portions of this package have been installed in six plants; favorable results have been obtained with up to two years of operation.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.9 JET PUMPS

Summary of Findings and Recommendations in Reed Report - 1975

Mechanical failures had occurred in jet pumps of earlier design due to flow-induced vibration. It was recommended that the new BWR/6 jet pump design undergo thorough testing before being placed in service.

Update - 1987

Conclusion:

All BWR/6 jet pump data to date indicate jet pump vibration levels meet acceptance criteria.

Summary:

All BWR/6 jet pump sizes were tested in a development facility. Design changes and installation procedures were implemented to reduce vibration in BWR/6 jet pumps. Vibration tests are also performed on jet pumps in each BWR/6 reactor during startup.

Supporting Information

The jet pump, which is a simple static device that circulates core cooling water, is totally contained within the reactor pressure vessel. It has no moving parts and is designed for a 40-year life. However, flow induced vibration is a concern.

Failures of jet pumps had been observed in two different bolted connections between components of the pump. One of the bolted connections was eliminated in the BWR/6 design. The assembly technique of the other bolted connection was improved to provide positive visual procedures to ensure correct assembly of the BWR/3, 4, 5 and 6 pumps.

Testing of jet pumps to determine vibration characteristics has been part of the pump program since 1970. The testing program in the High Flow Hydraulic Facility has been completed and all jet pumps for the BWR/6 reactors have been vibration tested in the development test facility at reactor operating conditions.

As plants are completed, instrumented in-reactor vibration testing of each unique jet pump design will be conducted during plant startup. The first BWR/6-218 size plant (Kuosheng 1) was subjected to cold flow and power operational testing. Instrumented vibration testing has been completed at a BWR/6-251 size plant and is ongoing at a BWR/6-238 size plant. All results to date indicate jet pump vibration levels to be below a conservative acceptance criteria.

In addition to these first-of-a-kind tests, each separate jet pump will be cold flow tested as part of the startup procedure. After this operation, jet pumps and other internal hardware will be visually examined for indication of abnormal performance. There has been no unacceptable vibration in any of the affected plants or in any pumps which have been cold flow tested.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.10 REACTOR PRESSURE VESSEL

Summary of Findings and Recommendations in Reed Report - 1975

While a sudden disruptive failure of the reactor pressure vessel (RPV) is judged to be extremely unlikely, it was recommended that updated analyses be performed for the most severe conditions. Areas to be addressed included: RPV integrity under loss of coolant accident (LOCA) conditions (to update a 1968 analysis), RPV integrity under the new NRC criteria for anticipated transient without scram (ATWS), need for annealing non-jet pump vessels due to radiation embrittlement and improvement of inspectability techniques and equipment for pre-BWR-6/RPVs.

Update - 1987

Conclusion:

Analyses reconfirm that reactor pressure vessel integrity is maintained for at least the 40-year design life.

Summary:

A series of LOCA reanalyses were performed which reconfirmed vessel integrity even under the extreme ATWS event. Further evaluation of vessel material properties versus neutron radiation verified that the degree of vessel embrittlement continued to meet 40-year life requirements.

A series of analyses was conducted in 1977 and 1978 which updated the LOCA analyses of 1968 and documented vessel integrity for an ATWS event. The principal conclusions of these analyses were:

- The pressure vessel integrity was reconfirmed to meet 40-year design life requirements.
- Brittle fracture will not occur during a postulated loss of coolant accident.
- Cracks will not propagate in a brittle manner.
- The automatic operation of safety relief valves, as well as automatic trip of recirculation pumps, assure vessel integrity during the ATWS event by limiting and controlling reactor pressure.

The embrittlement of pressure vessel materials due to neutron irradiation is monitored by dosimeters inserted in the reactor and periodically removed to establish neutron irradiation. The embrittlement effects are then determined from known behavior of pressure vessel materials versus neutron irradiation. Based upon evaluation of operating plant data, annealing of non-jet pump vessels will not be necessary. The degree of embrittlement is monitored to ensure that fracture toughness remains in the safe range. Additionally, brittle fracture calculations have been performed and updated in accordance with the latest NRC regulations.

For BWR/2-4 plants, internal surfaces are examined by remote visual inspection techniques.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.11 VESSEL NOZZLES AND SPARGERS

Summary of Findings and Recommendations in Reed Report - 1975

Cracks had been observed in the cladding around feedwater nozzles in some operating plants. All other operating plants had been instructed by the NRC to specifically look for the problem. A program was to be undertaken to determine the causes of the cracking and to develop solutions.

Update - 1987

Conclusion:

Design changes and repairs were made in most plants. Experience to date shows no additional cracking at these plants. Improved ultrasonic inspection techniques are used to monitor for cracking.

Summary:

The cause of cracking in feedwater nozzles and spargers was identified. Design changes were implemented on plants under construction. Repairs to operating plants were accomplished on the majority of the plants. In-service inspection programs were established at operating plants for early detection of defects. To date no further cracking has been detected in operating plants that have replaced feedwater spargers.

Supporting Information:

The cause of crack initiation was determined to be high-cycle thermal fatigue produced by mixing of the colder feedwater and the hotter reactor water. The crack growth due to these surface effects is limited and further growth would be dependent on reactor and feedwater system operational cycles. Analysis, correlated with tests, demonstrated that even if cracks were undetected and not repaired, the ultimate result would be detectable vessel leakage rather than vessel failure.

A design change was developed to preclude nozzle cracking which included:

1. Nozzle clad removal to increase the tolerable thermal cycling and to reduce the probability of initial defects.

2. A new feedwater sparger design to reduce the magnitude of thermal cycling.

3. Feedwater system operational procedure improvements to reduce the thermal cycling potential and duty.

Design changes were implemented in all plants under construction and an inspection and repair program has been implemented on most of the operating plants. In conjunction with the repair, an in-service inspection program has been established which utilizes ultrasonic examination from the outside of the nozzle, liquid penetrant examination of the nozzle ID surface, and visual examination of the sparger. Also, some operating plants are utilizing temperature sensors on the nozzle OD surface as a technique for early detection of leakage flow through the interference fit thermal sleeve seal. The inspection program which is being implemented at operating plants is defined in NRC requirements.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.12 RADIOACTIVE WASTE PROCESSING

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that equipment limitations, input waste volume and quality variations were the primary causes of radwaste operating problems, with operational methods a contributing factor. The level of effort being devoted to these factors in the new BWR/6 system was seen as appropriate. It was recommended that better communication and feedback channels on radwaste issues be explored with utilities.

Update - 1987

Conclusion:

GE's current BWR/6 radwaste processing system provides an improved, more efficient design but GE maintains communication and feedback channels with customers on all issues, including radwaste.

Summary:

The BWR/6 radwaste system was designed to accept a wide range of liquid waste volumes with maximum recycle of treated water back to the condensate system. An improved evaporator design provides increased corrosion protection and heat exchange capability by utilizing a high-flow circulation with an external, submerged tube heating exchanger. Redundant installations assure capacity over a wide range of input variations. Additionally, a flat bed, traveling belt-type filter is incorporated for removal of crud from low conductivity waste water. This design simplifies precoat procedures and discharges a dry solid cake, eliminating the need for a downstream dewatering device prior to solidification.

To ease the operator load on these remotely operated processing units, a mode control method, using programmable controllers for operational sequence control and initiation, is also incorporated in the present design. This method reduces the risk of operator error and exposure by allowing control room system operation and planning.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.13 GASKETS, SEALS AND PACKING

Summary of Findings and Recommendations in Reed Report - 1975

Programs to improve gasket, seal and packing lifetimes were recommended. Areas of concern included:

- Problems with packing, gaskets and seals in both GE manufactured components, such as control rod drives, and vendor-supplied items, such as valves and pumps.
- Possible resonance and excessive vibration as a result of using discharge throttling rather than pump speed change to regulate recirculation flow in BWR/5s and 6s.
- Need for improved recirculation pump shaft seal lifetime.

At the time of the Reed, efforts were being made to increase the lifetime of various gaskets, seals and packings. GE's Service Information Letter (SIL) 128 was issued to utilities in March 1975 and other SILs were issued as programs led to the development of improved products in this area.

Update - 1987

Conclusion:

Programs to improve designs and to identify optimum replacement intervals were completed. Field experience has demonstrated that the identified problems have been resolved.

Summary:

Service Information Letters (SILs) were issued recommending replacement intervals for equipment parts. Operating experience with reactor water cleanup pumps was improved. As of 1986 a new advanced, seal-less design is being offered to plant owners. Recirculation pumps have incorporated a seal purge extending seal replacement to four years from an earlier two-year interval.

Supporting Information:

Field experience has demonstrated that control rod drive seals have acceptable service life. In-service tests are performed regularly to identify control rod drives for refurbishment. A Service Information Letter sent to all operating plants suggested regular replacement intervals for nonmetallic parts in control rod drive hydraulic control units.

With respect to valve stem packings, a study of field experience led to issuance of a Service Information Letter in 1977 recommending packing replacement with Grafoil and anti-extrusion rings.

Mechanical seals on the vendor supplied reactor water cleanup pumps have undergone thorough review during 1982 and 1983. The results led to design modifications inside the pump, improved maintenance instructions, and system operation guidelines. Operating experience has shown that seal life has measurably improved since these recommendations have been implemented. New design seal-less pumps which promise a substantial increase in trouble free service were offered starting in 1986 and were first applied in 1987.

Regarding recirculation pump vibration, tests were conducted in 1977 at Tokai-2 at constant recirculation pump speed and variable discharge throttle valve setting to assure freedom from resonance and excessive vibration. Favorable experience was obtained from plants operated with flow control valves over the full operating range through 1984. More recently, pumps made by a second vendor have experienced some vibration which is currently (1987) under investigation.

Regarding recirculation pump seal life, a seal purge system was implemented on all plants which started up after 1973 in an effort to improve recirculation pump shaft seal performance. This and other improvements has doubled the seal life to an average of approximately four years between maintenance intervals.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.14 RECIRCULATION BYPASS VALVE

Summary of Findings and Recommendations in Reed Report - 1975

Vendor design details of the flow control bypass valve were not well known to GE. Thorough testing was recommended.

Update - 1987

Conclusion:

It was concluded during the detailed design phase of the BWR/6 that the bypass system was not required, so it was deleted from the design.

Summary:

The flow control bypass valve has been eliminated from the BWR/6 design. This finding is no longer applicable.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.15 RELIEF VALVE AUGMENTED BYPASS (REVAB)

Summary of Findings and Recommendations in Reed Report - 1975

REVAB was an option offered on BWR/4, 5 and 6 to provide the capability of accepting sudden loss of electrical load without reactor scram and to perform this function at an equipment cost significantly less than that of a more conventional full steam bypass system. REVAB systems had not yet come on line. Recent transient studies resulting from other system considerations had raised doubt as to the ability of the REVAB option to perform its intended function effectively (i.e. prevent scram on loss of electrical load), particularly near the end of a fuel cycle. It was recommended that a review be conducted to evaluate the ability of REVAB to meet its design objectives.

Update - 1987

Conclusion:

After further study the REVAB system was deleted from the BWR/6 design.

Summary:

Based on the results of further studies, GE withdrew the REVAB option from the product offering. This finding is no longer applicable.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.16 CONTROL ROD DRIVE (CRD) MATERIALS

Summary of Findings and Recommendations in Reed Report - 1975

A crack discovered in the collet retainer tube section of an operating plant control rod drive (utilities were notified by SIL 135, July 18, 1975) was considered significant, and it was recommended that priority be given to determining its cause and to identifying steps necessary to prevent additional incidence of such cracking. It was also recommended that there be a detailed metallurgical review of an alternate index tube material for fast scram CRD applications, as well as an evaluation of index tube materials selection criteria.

Update - 1987

Conclusion:

Extensive testing of cracked collet retainer tubes obtained from the field showed operability beyond six times the design life. Nevertheless, in 1976 an improved design was put in service. Operation of the new design has been satisfactory. Continuous monitoring of the old design where still in use assures that life expectancy is not changing.

Summary:

Extensive cyclic testing was performed on cracked collet retainer tubes retrieved from the field, and it was determined that the tubes would operate to beyond six times the 40-year design lifetime without failure of cracked parts.

Nevertheless, based on identification of factors causing cracking, an improved replacement design of a new material was developed and proof tested. The improved collet retainer tube was introduced in 1976. In the Fall of 1978, the NRC (Division of Operating Reactors) issued a statement in support of the drive modification. Since 1976, drives with the improved design have been utilized in all plants under construction. Periodic inspections of the old design have been continued in operating reactors. Drives with crack indications have been replaced with the improved design.

The results of the investigation of the collet retainer tube cracking has been reported by S. Ranganath, et al*.

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* "Failure Analysis, Testing and Product Improvement of Control Rod Drive Component From a Boiling Water Reactor," Symposium Paper: Failure Prevention and Reliability Presented at the Design Engineering Technical Conference, Chicago, Illinois, September 26-28, 1977 sponsored by the Reliability, Stress Analysis and Failure Prevention Committee of the Design Engineering Division, ASME.

Supporting Information:

Collet Retainer Tube

Following the discovery of the collet retainer tube cracking in June 1975, a vigorous investigation was initiated to assess the situation. Intensified field inspections and operating checks were made, metallurgical investigations were conducted, hot and full-scale overtesting was performed and a recalculation of stresses at the location of cracks was done.

These investigations had shown that the cracks observed in the collet retainer tubes of CRD mechanisms were caused by high, thermally induced cyclic residual stresses in a part sensitized by nitriding and exposed to an environment conducive to intergranular stress corrosion cracking.

As a conservative measure, an improved replacement design was developed and extensively proof-tested. Since the cracking was caused by stress corrosion, the three contributing conditions (susceptible material, stress and environment) were addressed, and the following steps to product improvement were identified:

a) Change in material to CF3 with controlled ferrite to provide resistance to stress corrosion.

b) Redesign of the collet retainer tube to reduce the residual and operational stresses.

c) Recommendations to operating plant utilities to provide low oxygen coolant for the drives during operation to reduce the aggressiveness of the environment.

Index/Piston Tube

Destructive evaluations (performed under the General Electric long-term component surveillance program) on control rod drive index and piston tubes had revealed isolated areas of local intergranular attack. This attack was found in creviced and noncreviced areas, with the largest area of attack having less than one-fifth the area required to cause component failure. Further destructive evaluations were made, and safety analysis evaluations were performed to determine the consequences of a failed index or piston tube. It was concluded that the worst potential occurrence would be an immobilization of the CRD after an insertion cycle, a condition equivalent to a coupling failure and one that would have no safety impact.

An improved material, XM-19 for index and piston tubes was put into full production in 1976 as a design improvement.

An extensive materials property verification program found that XM-19 is highly resistant to stress corrosion, sensitization, and intergranular attack when subjected to a specific process thermal treatment.

2.0 EQUIPMENT RELIABILITY ASSURANCE

2.17 PRESSURE CONTROL SYSTEM

Summary of Findings and Recommendations in Reed Report - 1975

There had been early reliability problems with the pressure control system, but reliability was improving. System changes were under way which, when completed, were expected to have their first applications in overseas plants. It was recommended that GE maintain a complete set of qualified system hardware in San Jose in order to quickly develop and test solutions to any problems that might develop in the field with the redesigned system. It was also recommended that consideration be given to transferring responsibility for design of electronic components of the system to Control and Instrumentation Engineering.

Update - 1987

Conclusion:

The system has been performing well in operating BWR/6 plants.

Summary:

The pressure control system for the BWR/6 has undergone full design qualification testing and each individual system was functionally tested prior to shipment and again at the site prior to startup. Should any problems develop, a solution would be developed through joint efforts of GE engineers in San Jose and in the field.

Design, manufacturing and quality assurance testing functions have been the responsibility of the Control & Instrumentation organization since 1976.

Supporting Information:

GE performed and documented a comprehensive design qualification test of the prototype BWR/6 steam bypass and pressure regulation system in 1976. Necessary corrective actions identified from this testing were incorporated in the final design. The system electronic control equipment was subjected to GE's rigorous quality assurance testing prior to shipment. At the site, the entire system, which includes the control electronics, bypass valves and hydraulic power unit, is functionally tested prior to plant startup. During plant startup, system operation is tested and adjusted to ensure proper operation. Approved procedures for these tests are in place.

Site calibration and testing of this system are performed under the direction of GE's site Process Control & Instrumentation Engineer. Direct site participation of system testing during the preoperational and startup phases by the Lead System Engineer ensure quick and effective resolution of any problems associated with the system.

The system has been performing well in operating BWR/6 plants.

3.0 FUEL & CORE IMPROVEMENT

3.1 FUEL MECHANICAL DESIGN

<u>Summary of Findings and Recommendations in Reed Report - 1975</u>

It was postulated that transient conditions such as loss of feedwater heating or accidental initiation of emergency core cooling system might result in fuel failures. It was recommended that fuel design and analyses should consider the fuel duty imposed by plant transients.

<u>Update - 1987</u>

<u>Conclusion:</u>

The fuel design has proven adequate. Transients in operating reactors have not resulted in significant fuel failures.

<u>Summary:</u>

Analyses and tests demonstrated that abnormal operational transients would only result in an acceptably-small number of fuel cladding perforations for non-barrier fuel designs. This conclusion was supported by:

1. <u>Operating Experience in Commercial Reactors</u>

   Through 1983 General Electric BWRs experienced over 500 operating transients at 26 different U.S. BWRs, which represented approximately 200 reactor years of experience. These abnormal operational transients included loss of feedwater heating transients. None of these transients resulted in significant numbers of fuel failures.

2. <u>Specialized Testing</u>

   The OPTRAN Program was a U.S. government (NRC) sponsored program conducted by EG&G Idaho during 1982-83 for the specific purpose of evaluating the consequences of BWR abnormal operational transients. The program was conducted at the INEL Power Burst Facility with General Electric non-barrier fuel rods. The fuel rods were subjected to multiple power transients representative of, or more severe than, the largest overpower BWR operational transients with no resultant fuel rod failures.

General Electric's current fuel product lines, including barrier fuel, have been demonstrated to provide increased failure protection during severe overpower events.

3.0 FUEL AND CORE IMPROVEMENTS

3.2 FUEL FAILURES - SHORT TERM

Summary of Findings and Recommendations in Reed Report - 1975

Fuel operating recommendations had been effective in reducing fuel failures, but capacity factor losses caused by following the recommendations (PCIOMR) were significant. It was postulated that it might be possible to improve the effectiveness of the PCIOMR recommendations while simultaneously reducing the capacity factor losses. It was recommended that core and fuel design changes be considered which might be effective in the near term. These included thick clad on the corner rods (which have the highest duty in a bundle), and core management schemes which avoid moving fuel from a low power to a higher power region in the core.

Update - 1987

Conclusion:

GE had initiated work in areas covered by these recommendations prior to the Reed study. Improvement in designs, core management schemes and in PCIOMR procedures resulted in significant improvement in capacity factor. Plants that use GE's new barrier fuel design do not need PCIOMR.

Summary:

Improvements have been made in the following areas resulting in a substantial reduction in capacity factor losses.

1. Replacement of 7x7 lattice design fuel bundles with 8x8 lattice design fuel bundles.

2. Further qualification and subsequent implementation of an exposure dependent threshold for initiation of pre-conditioning interim operating management recommendations (PCIOMR) ramp rates.

3. Further qualification and subsequent implementation of a PCIOMR ramp rate increase.

4. Increased fuel rod prepressurization to further reduce fuel rod peak center temperature.

5. Improved procedures and process computer assistance to the operator.

6. In-reactor qualification and implementation of control cell core which isolates control blade movement to low temperature low power fuel bundles.

7. Introduction of barrier fuel.

The above changes have significantly reduced capacity factor losses.

Supporting Information:

An increase in ramp rate (from .06 to .11 kw/ft/hr) was released in 1974 as a recommendation to utilities, based on test reactor experiments and confirmed by an actual operating plant. This recommendation has been implemented in most plants and results have shown more than a 1% increase in capacity factor associated with PCIOMRs (Pre Conditioning Interim Operating Management Recommendations).

The introduction of control cell core has also reduced losses associated with PCIOMR. An exposure dependent threshold for initiation of PCIOMR ramp rates has been applied which enables fresh fuel to operate unrestricted for almost a year. Capacity factor losses associated with PCIOMRs have been significantly reduced for the seven reasons noted above.

3.0 FUEL AND CORE IMPROVEMENTS

3.3 FUEL FAILURES - LONG TERM

Summary of Findings and Recommendations in Reed Report - 1975

Fuel failures were identified as a high-priority concern which presented a significant technical challenge, both from the standpoint of identifying the failure mechanism and of determining a solution. It was recommended that two or three approaches be selected for long-term development. It was also recommended that 9x9 non-barrier fuel be evaluated as a means for reducing failures.

Update - 1987

Conclusion:

Introduction of several design improvements into an 8x8 arrangement has resulted in a reliability of 99.98% (all causes considered) measured over 2.4 million 8x8 fuel rods. See also Item 3.2 discussing barrier fuel introduction.

Summary:

Following the occurrence of a number of fuel failures, an intensive effort was mounted within General Electric to both identify the specific failure mechanism and to develop means for eliminating these failures.

The specific failure mechanism has been identified as being caused by localized stresses which in the presence of corrosive fission products (cadmium and/or iodine) can cause stress-corrosion cracking during rapid power increases.

Early in the investigation several approaches were identified which could either minimize the localized stresses or protect the cladding from the specific fission products. These approaches were evaluated both analytically and experimentally. Samples of each type of potential countermeasure were fabricated into prototype fuel rod segments and irradiated to significant burnups in operating reactors. Power ramp tests of these fuel rod segments in test reactors have provided direct comparative data on the effectiveness of each potential remedy and led to the selection of a reference PCI resistant fuel design (barrier fuel).

The proposed 9x9 non-barrier fuel was shown by ramp testing of non barrier fuel rods to be incapable of completely eliminating PCI failure either during normal operating conditions or during abnormal operational transients.

Supporting Information:

Since the early 1970s when fuel failures in 7x7 fuel bundles were identified as a concern there have been a series of evolutionary fuel design improvements which have resulted in current fuel performance that is highly reliable. In

1972 improved fuel containing shorter chamfered pellets, hydrogen getters and annealed cladding was placed in operation. This was followed in 1974 by the introduction of the lower duty 8x8 fuel, in 1977 by incorporating the previous improvements into 8x8 fuel, and in 1979 by fuel prepressurized to three atmospheres. The latest improvement, introduced in 1983, is the prepressurized 8x8 fuel with barrier cladding.

The 7x7 fuel, which had experienced a high failure rate, has all been replaced with the 8x8 fuel designs. More than 2.4 million fuel rods of the 8x8 fuel designs have completed at least one cycle of operation since the first 8x8 introduction in 1974, and the total population of 8x8 rods that have completed at least one cycle of operation as of January 1, 1987 has demonstrated reliability of greater than 99.98%.

3.0 FUEL AND CORE IMPROVEMENTS

3.4 FUEL MANUFACTURING

<u>Summary of Findings and Recommendations in Reed Report - 1975</u>

It was recommended that efforts continue to develop programs which would assure superior quality in manufactured fuel while maintaining production costs at an acceptable level.

<u>Update - 1987</u>

<u>Conclusion:</u>

Fuel reliability of 1st cycle 8x8 fuel now exceeds 99.999% where failure cause originates in manufacturing.

<u>Summary:</u>

General Electric actions to improve fuel component quality include:

1. Areas where quality improvements were needed in production were identified by responsible quality assurance and engineering organizations.

2. Programs for improvement were defined for each area.

3. The quality assurance organization was increased in size and scope at all levels to implement these programs.

As a result of increased quality assurance, the quality of Zircaloy tubing produced has improved. Particular improvements were made in the ultrasonic testing for tube defects.

In the ceramic fuel area additives have been employed to improve product thermal stability and quality.

As a result of increased quality assurance and improved manufacturing processes, GE's fuel reliability rate during the first cycle of operation for manufacturing process flaws has increased to a present level of greater than 99.999% for all 8x8 fuel rods completing one cycle of operation as of 1/1/87.

3.0 FUEL AND CORE IMPROVEMENTS

3.5 NEW FUEL FAILURE MECHANISMS

<u>Summary of Findings and Recommendations in Reed Report - 1975</u>

Limited data were available on average fuel bundle exposures over 15-20,000 MWd/t, and it was noted that as pellet clad interaction failures decreased and physics limitations were overcome, exposures would increase and other failure modes might be expected to show up. It was recommended that programs be developed for analysis, surveillance and testing of fuel to identify any new failure mechanisms that might occur.

<u>Update - 1987</u>

<u>Conclusion:</u>

Use of data acquired from surveillance and testing forms the bases of the overall reliability record of 99.98% on 8x8 fuel.

<u>Summary:</u>

As of 1/1/87 approximately 2.4 million 8x8 fuel rods have completed at least one cycle of operation in domestic and foreign boiling water reactors. As of 1/1/84 over 12,000 of these bundles (more than 750,000 rods) have achieved more than 20,000 MWD/MT of exposure. General Electric activities aimed at identifying new failure mechanisms have concentrated on a systematic program of laboratory testing, test reactor experimentation, surveillance examinations (at reactor sites) of lead test fuel bundles and components, and detailed post-irradiation examinations of selected fuel and fuel bundle components after completion of reactor service. This systematic design approach has resulted in a fuel rod reliability of greater than 99.98% for 8x8 fuel rods that have completed at least one cycle of operation as of 1/1/87.

3.0 FUEL AND CORE IMPROVEMENTS

3.6 FINE MOTION CONTROL ROD DRIVE (FMCRD)

Summary of Findings and Recommendations in Reed Report - 1975

It was postulated that a fine motion control rod drive might be helpful in the overall solution of the fuel failure problem by, for example, making operating recommendations easier to implement. It was recommended that a program be initiated to choose a fine motion control rod drive.

Update - 1987

Conclusion:

Test program conducted as recommended. Success of other fuel programs and operating improvements made FMCRD unnecessary (see Items 3.2 and 3.3).

Summary:

The application of a fine motion control rod drive was one of several early solutions proposed for reducing the number of fuel failures. Studies and preliminary designs were made on the assumption that such a drive would be developed for later BWR/6 plants, with the possibility of retrofitting to earlier systems if required. Design changes have been made in all BWR/6 drive housings to allow for such an eventuality. Some prototype testing was completed before the decision was made to terminate the program. Other solutions, such as fuel preconditioning and the 8x8 fuel designs, were developed more rapidly and proved to be sufficiently effective in reducing fuel failures to warrant discontinuing the fine motion drive program for the BWR/6.

3.0 FUEL AND CORE IMPROVEMENTS

3.7 FUEL TEST FACILITIES

Summary of Findings and Recommendations in Reed Report - 1975

The General Electric Test Reactor (GETR) required modification and license renewal to support the GE fuel testing program. The timely relicensing of GETR was identified to be a top priority objective. In addition, it was recommended that the possibility of developing a noncommercial reactor licensed for fuel and components testing be explored with utility and government organizations.

It was also recommended that capability at the Radioactive Materials Laboratory (RML) be expanded. In the short term, added staffing of the RML and provisions for increased fuel transport were seen as essential. It was recommended that the RML be modified for full-length fuel handling.

Another essential activity identified was the maintenance of a fuel surveillance base for continual comparison and evaluation of operating experience.

Update - 1987

Conclusion:

Alternate facilities worldwide are available to meet GE's needs. Utilizing other facilities, high quality and timely surveillance and testing programs have been carried out routinely as recommended.

Summary:

Fuel testing has been continued by GE in Norway and was expanded to Sweden subsequent to qualifying the Swedish test reactor facilities.

The staffing of the RML has been determined to be adequate to meet the examination requirements of various fuel and materials programs. Provisions for fuel transport to and from the RML also have been adequate to meet all testing needs. Design of a shipping container which will permit shipment of failed fuel capsules has been completed. High-quality and timely examinations of fuel rods have been conducted in a routine manner through the segmented rod program which allows shipment of the segments to the RML. In instances where it was technically desirable to examine full-length fuel rods, contract arrangements have been made with a commercial research organization with the required facility.

General Electric has maintained an experienced fuel surveillance staff and has invested in advanced scientific equipment to upgrade the efficiency and quality of fuel surveillance measurements at commercial reactor sites. The fuel surveillance staff has been active at numerous domestic and foreign reactor sites and has developed a substantial data base to evaluate the performance of core and fuel components.

3.0 FUEL AND CORE IMPROVEMENTS

3.8 FUEL SPACER

Summary of Findings and Recommendations in Reed Report - 1975

There was concern about whether the fuel spacer for the 8x8 fuel array had adequate seismic capability. It was recommended that GE continue development on the spacer to assure that it provided high seismic capability. It was also noted that the 8x8 spacer design might not be suitable for 9x9 fuel, should it be adopted, and it was recommended that a program be initiated to develop a 9x9 spacer.

Update - 1987

Conclusion:

Over 170 tests have demonstrated the seismic adequacy of the 8x8 spacer. Success with the 8x8 programs obviated the need for a 9x9 bundle.

Summary:

The function of the spacers is to maintain the lateral spacing between adjacent fuel rods and between the fuel rods and channel while allowing differential axial expansion between the individual rods and the channel. The spacer must also allow for the axial passage of the moderator fluid (water) through the fuel assembly.

Starting in 1974 a seismic testing program was initiated which demonstrated the seismic capability of the 8x8 fuel rod spacer. The testing program consisted of a series of static and cyclic loading tests of spacers in which the major variables affecting seismic capability were considered, such as:

- o thinning due to corrosion
- o irradiation
- o temperature
- o mechanical design variations
- o direction of applied load

Over 170 individual tests were performed and results demonstrated that the 8x8 spacer is capable of withstanding seismic loads. NRC has approved the 8x8 fuel design and it is in use in operating reactors.

4.0 MATERIALS AND PROCESSES IMPROVEMENTS

4.1 STRESS CORROSION CRACKING

Summary of Findings and Recommendations in Reed Report - 1975

Stress corrosion cracking (SCC) had occurred in furnace sensitized 304 stainless steel and, to a lesser degree, in nitrided stainless steel parts, weld sensitized parts, and cold worked bolts. Laboratory tests had shown that the BWR environment could accelerate fatigue crack growth in low alloy and carbon steel, but no field failures attributed to stress corrosion cracking had resulted. The occurrence of crevice corrosion was found to be low in BWRs, but further study of the problem was recommended. It was recommended that stress corrosion cracking problems be given highest priority, with programs established to determine its cause and the possible changes that might be required in environment, materials and design of the BWR to mitigate SCC.

Update - 1987

Conclusion:

The cause of cracking and suitable countermeasures have been identified. These have been and are continuing to be implemented in both operating plants and those under construction.

Summary:
Stress corrosion cracking is a complex, industry-wide problem, affecting both BWRs and PWRs, and relates fundamentally to the harsh environment in which components and piping must operate in nuclear plants. It has taken the industry many years to gain an understanding of the cause of the cracking and to identify countermeasures.

Actions taken by General Electric to upgrade BWR reliability as impacted by stress corrosion cracking can be separated into four tasks: (1) understanding the cause, (2) developing and implementing countermeasures for operating plants, (3) developing and implementing countermeasures for plants in construction, and (4) incorporating solutions into future plants. These activities have resulted in identification of the cause(s) of the SCC in BWRs and in development of materials, procedures and equipment modifications to provide solutions.

Supporting Information:

The cause of cracking was identified as a unique combination of an oxidizing environment, sensitized material and stresses above yield stress due to a combination of all sources of stress including weld residual stresses. The fact that all three factors are required explains the statistical nature of cracking observed in boiling water reactors. General Electric has built a pipe test laboratory to test prototypic pipes with diameters from 4" to 16". Results from a joint program with the Electric Power Research Institute confirmed the above conclusion on the cause of cracking.

For operating plants, General Electric has recommended that all plants with furnace sensitized stainless steel safe-ends do the necessary advanced planning and materials procurement to replace these safe-ends if defects are detected during in-service inspection. General Electric has also recommended that deaeration of the reactor water be used during shutdown, layup and startup to improve the environment. In conjunction with EPRI, General Electric has also completed qualification of several additional remedy options for operating plants including 316 Nuclear Grade stainless steel replacement pipe and Induction Heating Stress Improvement.

Since 1982, the reported incidence of stress corrosion cracking in recirculation system piping has increased significantly especially in larger diameter twelve through twenty-eight inch piping where it had not previously been observed in the US. This increase is attributed to greater in-service inspection frequency and the use of more effective ultrasonic testing techniques. The cracking continues to be consistent with the understanding developed since 1975 and the countermeasures developed to mitigate cracking remain valid solutions.

For plants in construction, the same recommendations were made as were made for operating plants. In addition, General Electric undertook a joint program with the Electric Power Research Institute (EPRI) to qualify countermeasures for stress corrosion cracking that could be implemented during construction and thus further improve plant reliability. Heat sink welding, solution heat treatment, and corrosion resistant cladding have been qualified. All of these countermeasures have been implemented on a widespread basis in plants under construction. Implementation is rapidly increasing in operating plants.

Laboratory and plant testing to date also indicates that Hydrogen Water Chemistry mitigates stress corrosion cracking and can arrest pre-existing stress corrosion cracks.

4.0 MATERIALS AND PROCESSES IMPROVEMENTS

4.2 MATERIALS EFFORT AND CONTROL OF APPLICATIONS

Summary of Findings and Recommendations in Reed Report - 1975

While active work was in progress in many materials areas and, with the exception of stress corrosion cracking (see Topic 4.1), no significant problems had been identified, it was recommended that additional effort be spent in materials engineering development and design. It was also noted that the materials effort was organizationally diffused and that the test facilities needed to be expanded and made more accessible for materials engineering. Other areas requiring attention were the possible need for tightening and control in the application of material specifications and the need to complete development changes before the start of production. Programs were recommended to address these areas.

Update - 1987

Conclusion:

Over 400 materials and processes specifications have been changed or issued. Procedures for control of materials application and implementation are in place. Materials effort was augmented and organizationally focused.

Summary:

General Electric's action to upgrade the materials area included a reorganization which consolidated all fuel and structural materials engineering into two components. Substantial support is also derived from the GE Corporate Research Center and from licensees in Japan, Sweden and Germany through exchange programs. Additional testing facilities have been developed at GE to facilitate materials development and analysis efforts.

The fuel and structural materials organizations have responsibility for assuring that all materials and processes used in the nuclear steam supply system include the latest technical requirements by verifying all drawings and specifications. These same materials application engineers have responsibility for working with vendors who supply material to General Electric specification which have requirements more stringent than industry codes and standards. These engineers audit the vendors' plants to assure they understand the GE requirements, and that they are capable in terms of equipment and procedures to meet the specifications. Nuclear grade materials specifications control procurement so that these special materials can only be procured from technically qualified sources. Welding procedures for installation of pipe and components in plants under technical control of General Electric are approved by these materials application engineers.

Processing rules and design rules for stress corrosion in stainless steel, both with and without crevices and for corrosion fatigue of carbon steel, stainless steel, and low alloy steel, have been developed and specified as design requirements over and above the standard Code requirements where they apply.

4.0 MATERIALS AND PROCESSES IMPROVEMENTS

4.3 CORROSION FATIGUE

Summary of Findings and Recommendations in Reed Report - 1975

Several materials areas were identified as requiring continued study in order to determine if there were potential problems for which solutions should be co - sidered in the BWR design. These areas included fatigue crack propagation in carbon, low alloy and sensitized steel; crevice corrosion; crud contribution from Inconel 600 corrosion and the material effects of weld sensitization, cyclic frequency and wave form.

Update - 1987

Conclusion:

The potential concerns have been addressed and, where appropriate, actions have been taken such as issuance of corrosion fatigue design rules and the development of hydrogen water chemistry to minimize environmental effects. No open issues remain in this area.

Summary:

Special design rules to prevent corrosion fatigue in carbon steel have been developed and approved for use in new designs. Hydrogen water chemistry has been identified as a remedy for crevice corrosion of stainless steel, stress corrosion or corrosion fatigue of low alloy steel, and stress corrosion of Inconel 600 and its weld metals. Inconel 600 has been evaluated as being a negligible source of crud. (Also see Topic 4.1 - Stress Corrosion Cracking.)

4.0 MATERIALS AND PROCESSES IMPROVEMENTS

4.4 WELD BEAD STRAIGHTENING

Summary of Findings and Recommendations in Reed Report - 1975

It was postulated that the practice of weld bead drawing or straightening of the jet pump and riser might introduce undesirable residual stresses in the 304 stainless steel alloy. It was recommended that stress-related effects of weld bead straightening and drawing be reviewed.

Update - 1987

Conclusion:

The magnitude of the weld bead straightening stresses and degree of sensitization were found to be within the bounds of conventional weld joints. However, to provide added margin, controls for weld bead str ightening were defined in design documentation.

Summary:

Test specimens were removed from mockups of jet pump components which received the most repetitive weld bead straightening as well as normal component welding (no straightening). Specimens were subjected to electropotentiokinetic reactivation analysis. It was determined that the weld straightening operation did not significantly add to the sensitization of the base material (Type 304 stainless steel).

Evaluations have shown that residual stresses due to weld bead or draw bead straightening are bounded by the residual stresses of structural or attachment welds on the components. Current specifications have also been revised to limit and control the application of weld or draw bead straightening. Straightening beads are applied to existing weld joints wherever possible. When required on base material, heat input, bead position and the extent of welding are controlled to minimize sensitization and residual stress.

A review of the jet pump assembly was conducted in 1976, and the Type 304 stainless steel components were changed on later plants to Type 316L material. Most other portions of the jet pump assembly are fabricated from Type 304L or CF-8 stainless steel, which are less susceptible to cracking.

4.0 MATERIALS AND PROCESSES IMPROVEMENTS

4.5 FUEL CHANNELS

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that fuel channels had experienced corrosion and creep (longitudinal bowing or distending of channel walls) during operation, and there was a concern that excessive creep could cause binding of a control blade. The problem was being addressed through development of thicker walled, creep-resistant channels, but some fabrication difficulties had been encountered. It was recommended that the process for fabricating thicker walled channels be perfected and that methods be developed to improve resistance to corrosion and dimensional changes.

Update - 1987

Conclusion:

Channel corrosion has been reduced and an improved understanding gained of the creep process. No control blade binding has been experienced.

Summary:

The manufacturing process employed to form bends of thicker channel material into fuel channels was evaluated to assure that the process did not cause cracks on the inside corners of the bends. The manufacturing process has been modified to eliminate any potential for causing cracks.

General Electric has monitored corrosion and dimensional changes in irradiated channels to obtain an improved understanding of channel lifetime and creep. Corrosion resistance of channels has been improved by material changes implemented in the 1973-1974 period. A new channel heat treatment process was also developed and put into operation in 1979 to obtain additional corrosion resistance. Experimental data on irradiated channels indicate that this new process will reduce corrosion. No control blade binding has been experienced.

4.0 MATERIALS AND PROCESSES IMPROVEMENTS

4.6 MANUFACTURING PROCESS LABORATORY

Summary of Findings and Recommendations in Reed Report - 1975

The Wilmington plant was setting up a structural materials quality control laboratory and plans were also underway to establish a manufacturing process development and control laboratory. It was recommended that Wilmington also establish both a reactor equipment manufacturing laboratory and a fuels quality control and manufacturing process laboratory.

Update - 1987

Conclusion:

The recommended laboratories were completed. New processes have been developed for manufacturing control as well as enhancements used for fuel and structural components manufactured at Wilmington.

Summary:

Laboratories have been established and are functioning in the areas of automation, quality technology development, welding technology, uranium manufacturing processes, chemical processes, and zirconium corrosion, PCI-resistant cladding, and manufacturing processes for corrosion-resistant steels. The laboratories have been staffed with highly qualified professionals. New processes have evolved related to tube quality, channel corrosion resistance, uranium pellet properties, steel and zirconium welding, measurement of fuel rod contents, uranium accountability, criticality safety and others.

4.0 MATERIALS AND PROCESSES IMPROVEMENTS

4.7 INSTALLATION OF REACTOR INTERNALS

Summary of Findings and Recommendations in Reed Report - 1975

Installation of the reactor internals for Hanford 2 had been successfully completed at CBI Nuclear, GE's reactor pressure vessel manufacturing facility. This was the first time the installation had been performed in the factory rather than in the field, and there were a number of advantages with this method. It was recommended that it become the policy to perform installation of internals in the factory whenever practical.

Update - 1987

Summary:

The above recommendation was implemented. The installation of reactor internals for one other BWR/5 reactor pressure vessel and for eleven BWR/6 reactor pressure vessels was completed at the CBIN facility in Memphis, Tennessee.

5.0 DESIGN MARGIN AND OPERABILITY IMPROVEMENTS

5.1 DESIGN THERMAL MARGINS

Summary of Findings and Recommendations in Reed Report - 1975

At the time of the Reed Report, design work for the BWR/6 was still in its preliminary stages. A concern was noted that upon completion of the design, a need might exist for added core design margins to avoid forced power reductions due to licensing limits. In addition, concern was noted that margins be adequate to accommodate uncertainties in calculational models, future regulatory changes, and needed operational flexibility. It was recommended that consideration be given to reducing the core power density in future offerings by up to 20 percent to provide increased margins. Among possible methods proposed for accomplishing this goal were the use of next larger vessel sizes, which would allow for more fuel and higher flow capacity, and the use of six-month refueling intervals. It was also noted that some pre-BWR/6 plants had to operate at reduced power at certain times in operating cycles because of core margins. Concern was also expressed regarding cold shutdown margin adequacy.

Update - 1987

Conclusion:

Core design margins in the final BWR/6 designs are larger than even those predicted in preliminary design analyses. In fact, the BWR/6 design has so much margin that improvement programs permitting higher power levels and greater operating flexibility than originally sought have been licensed by the NRC and plants are operating in these modes today. The Reed task force concern that power reductions might be necessary has proven to be unfounded.

Summary:

During the completion of the BWR/6 design, a number of planned improvements in the preliminary design were made to assure that there would be more than adequate margins. Improved calculational models checked against an increased field data base have also increased the reliability of design predictions, further assuring adequate margins.

Consideration was given, as suggested in the Reed Report, to power density reductions through use of larger vessels and through six-month refueling intervals. Because acceptable thermal margins were assured through other programs, these design changes were not made. Adequate cold shutdown margin was also verified for the completed BWR/6 design.

The power restrictions relating to core margins noted in the Reed Report for certain pre-BWR/6 designs were due to regulatory requirements. The original design margins had been adequate but were eroded due to the changing NRC requirements. Many of the margin improvement programs developed for the BWR/6 design can also be used in pre-BWR/6 plants and have been used in some such plants to avoid power restrictions.

Supporting Information:

Actions taken to improve thermal margins included:

- o Qualification of the fast scram control rod drive.
- o Design implementation of a trip of recirculation pumps following closure of turbine stop and control valves. (This reduces the power increase in such a transient.)
- o Incorporation of a trip system based on simulated thermal power rather than neutron flux and a lowering of the trip setpoint. This reduced the magnitude of potential power increases following loss of feedwater heating.
- o Improvements in the control system which reduce the potential power increase associated with erroneous withdrawal of a control rod.

5.0 DESIGN MARGIN AND OPERABILITY IMPROVEMENTS

5.2 POWER MANEUVERING

Summary of Findings and Recommendations in Reed Report - 1975

The BWR/6 automatic flow control range from 75 percent to 100 percent of full power, while accepted by customers, was perceived to be less than optimum for load following needs. There was a concern that should rod motion be required to accommodate desired load following, this would tend to increase fuel failures and/or the time to precondition fuel might increase. Other perceived utility operating requirements identified were automatic generation control and frequency control. It was proposed that studies be made to determine feasibility and cost of measures to increase load following capability. It was also recommended that evaluations be performed to determine technical and economic feasibility of providing increased BWR capability for grid frequency control duty and for coping with network disturbances.

Update - 1987

Conclusion:

BWR/6 design accommodates the power maneuvering interests of utility customers in all cases except local governor control which can be added if desired.

Summary:

A substantial survey to determine utility system interests in power maneuvering was conducted by General Electric in 1976- 1977. Ten operating utilities and utility groups were questioned. The results of the survey were compared to expected BWR performance. The BWR/6 performance meets or exceeds the interests identified in the survey.

Supporting Information

An important factor in the ability of the BWR to vary its generation output is the control of core recirculation flow over a wide range. This range is determined analytically for automatic control by using results from stability analysis. Testing conducted at Peach Bottom indicates that the analytically predicted stability margin is conservative. Since the stability analysis predicts a 25 percent range, there is high confidence that the automatic flow control range on BWR/6 will achieve at least the 25 percent. Under manual control by the operator, power can be reduced further by use of flow control. At all these power levels, control rods can be moved without risk of barrier fuel failure. In the case of non-barrier fuel, observance of PCIOMRs will prevent fuel failures. BWR experience to date has confirmed its maneuvering capability.

5.0 DESIGN MARGIN AND OPERABILITY IMPROVEMENTS

5.3 FAST SCRAM DRIVE

Summary of Findings and Recommendations in Reed Report - 1975

It was recommended that the development of a fast scram drive for the BWR/6 continue on the course being followed and that extensive development and qualification tests be performed to assure satisfactory performance of the drive in the field.

Update - 1987

Conclusion:

The fast scram drive was proven to be fully qualified for reactor application and is performing well in operating plants.

Summary:

The fast scram control rod drive design was subjected to a comprehensive test program. These drives are now operating in BWR/6 plants, which have now accumulated 15 reactor years of operation.

Supporting Information

The fast scram drive test program consisted of three phases plus in-reactor operational test of a prototype model. A brief description of these test phases is outlined below:

1. Development Testing (1974 to 1976) - This testing was done in order to confirm that the proposed control rod drive design changes would meet the functional requirements. Individual piece parts and full assemblies were tested in order to verify that the basic control rod drive and related hydraulic system changes were compatible and that the full system was capable of producing the faster scram times.

2. Design Acceptance Testing (1976) - This test subjected one prototype control rod drive to the equivalent of 70 years of reactor service at reactor conditions--nearly double its actual design life--in order to verify that this design met all design specification requirements with satisfactory margin.

3. Production Qualification Testing (1977 to 1978) - This testing was performed at actual reactor conditions to establish a firm statistical base on control rod drive performance by testing a quantity of preproduction drives. The results further established design and reliability margins.

The great majority of the testing described above has been conducted at actual reactor operating conditions of temperature, pressure, etc., using test pressure vessels specifically constructed with reactor hardware which accurately simulates the reactor geometry.

5.0 DESIGN MARGIN AND OPERABILITY IMPROVEMENTS

5.4 FEEDWATER/LEVEL CONTROL

Summary of Findings and Recommendations in Reed Report - 1975

There was a concern regarding the capability of the feedwater control system level control to prevent high or low water scram during load rejection or turbine trip for plants with the Relief Valve Augmented Bypass (REVAB) or 100 percent bypass. The objective was to avoid such scrams in order to improve plant availability.

Update - 1987

Conclusion:

Analysis and operating experience has shown that this concern was unfounded for the 100% bypass system. The REVAB system was always an optional design and was never incorporated into an operating plant.

Summary:

Design analysis indicated no scrams would occur in the circumstances described on plants employing the 100 percent bypass feature. Recent startup tests at one BWR/6 with 100% bypass in which nearly 100% load was rejected confirms this conclusion.

6.0 PRODUCT SERVICE IMPROVEMENTS

6.1 REFUELING TIME

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that plant availability could be significantly increased by reducing the time required for refueling. It was recommended that studies be undertaken to develop procedures and tools to significantly reduce refueling/maintenance outages.

Update - 1987

Conclusion:

- Improvements have been made in the refueling platform and grapple, fuel transporting, quicker vessel head removal, faster fuel sipping, improved undervessel servicing equipment and construction of an $8.5 million refueling floor mockup facility to proof-test new tools and equipment.

- These improvements have streamlined outage tasks; however, total outage times have varied widely depending on many factors such as number of regulatory modifications implemented.

Summary:

Studies undertaken to develop means to reduce refueling/maintenance outages have been successfully completed. The studies have helped to direct engineering and implementation of BWR product improvements and revisions in procedures that have contributed to increased BWR operating plant availability. The knowledge gained has also been factored into the design of BWRs presently under construction. In addition, significant progress toward reduction of refueling/maintenance outage frequency has been achieved through implementation of GE's Advanced Maintenance Planning Services and the development of capability for longer periods between refuelings.

Supporting Information

Examples of BWR product improvements that are being offered and which contribute toward reduced refueling/maintenance outage length include the following:

1. Installation of upgraded refueling platform grapple on operating plants.

2. Redesign of refueling platform to provide faster fuel handling capability.

3. Equipment to remove and install the reactor vessel head more rapidly.

4. Equipment to provide faster fuel transport capability.

5. Equipment and techniques to provide faster sipping for failed fuel detection.

6. Undervessel servicing equipment to reduce maintenance operation time.

7. A complete reactor refueling floor and reactor vessel mockup facility. This facility known as the BWR Services Training Facility is used for qualification of outage crews, checkout of special tools and proof testing of various retrofit equipment prior to use.

6.0 PRODUCT SERVICES IMPROVEMENTS

6.2 BWR SERVICE

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that an active approach to providing product services was being pursued and it was recommended that this effort be continued and substantially expanded to assist operating BWRs to improve plant availability.

Update - 1987

Conclusion:

The GE product service program has substantially expanded and includes: AMPS, an Advanced Maintenance Planning Service tailored to each plant's specific needs; CIRS, a Component Information Retrieval System providing statistics on operational performance; placement of GE senior engineers at each reactor site to identify and implement specific improvement programs; COPs, Contingency Operating Programs to cover time when systems are out of operation, and many others such as annual conferences on specific technical problems, spare parts, computers and training all to share experience and new techniques. The expanded GE product service program contributes to better performance and better availability.

Summary:

General Electric has placed increasing emphasis on the service support for operating plants. This emphasis was reflected in the early formation of a nuclear services organization with the specific task of improving BWR reliability, availability and capacity factor. A number of major programs have been implemented to address these goals as discussed below.

Supporting Information

To perform the task of improving BWR performance, GE has analyzed in detail the causes of forced and extended outages. As a result of these analyses, specific recommendations have been made to utilities on a plant specific basis. Where generic situations have been identified, generic recommendations have been made to all plants. Some of the major General Electric programs to analyze plant performance, develop recommended retrofits and modifications and implement recommended actions are discussed below.

1. The Advanced Maintenance Planning Service (AMPS) is offered to BWR owners to identify both generic and plant specific improvement programs, data, recommendations and other information that can be used by BWR owners to improve BWR availability and capacity factor.

2. The Component Information Retrieval System (CIRS) has been established and is maintained to provide an operating plant data base which is used by General Electric engineering and services personnel to identify and implement specific BWR availability and capacity factor improvements.

3. Senior engineers have been placed on most BWR sites to work with operating plant personnel to identify and implement specific BWR performance improvements. These senior on-site engineers maintain a close liaison with General Electric service and engineering organizations.

4. A dedicated Operating Plant Engineering Section has been established for the sole purpose of supporting BWR operating plants. This organization has the capability to provide timely response to the engineering needs of BWR operating plants to improve their performance.

5. A BWR Operating Plant Retrofit Program has been established to develop hardware retrofits to improve operating plant performance. Specific retrofit offerings are developed based on analyses of BWR operating plant data and on analyses of state-of-the-art improvement being implemented in new product line designs as well as in response to NRC requirements.

6. A series of contingency operating programs have been established to cover equipment that may experience operating problems. These programs provide on-the-shelf contingency operating procedures with the necessary supporting analyses and, in many cases, replacement hardware to allow rapid regulatory review and approval to permit continued power operation.

7. Customer Service Managers are assigned to operating BWR plants to provide an orderly turnover of the plant to the owner and to serve as a focal point for communication between the utility and General Electric during the life of the plant. These individuals are located in Regional offices close to the customers and include engineering supporting staffs.

8. Other service actions include:

   o BWR Operating Plant Technical Conferences are held periodically to provide a forum for open discussion with BWR owners of operating plant issues.

   o A Spare Parts Section has been formed to ensure a full scope of services to BWR owners, including engineered spares and spare and renewal parts.

   o Most field support services have been grouped into one organization to provide integrated services during BWR construction, testing and operation.

   o Process Computer Users Conferences are held periodically to provide for information exchange between BWR owners and General Electric and to identify program improvements.

- o An annual Training Conference is held with BWR owners to exchange training experiences and discuss implementation of new training requirements.

- o A new maintenance/refueling training facility has been constructed by GE in San Jose to train GE field service crews, check-out repair of retrofit installation procedures and train BWR owners' key maintenance staff personnel (see also Item 6.1, Supporting Information).

The structural approach taken to provide full scope services to BWR operating plant owners has contributed to significant improvements in BWR availability and capacity factor.

6.0 PRODUCT SERVICES IMPROVEMENTS

6.3 REPLACEMENT FUEL

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that, in the event of an unscheduled shutdown due to failed fuel, a plant could lose availability because of fuel unavailability. It was recommended that plans be considered for assuring immediate availability of replacement fuel in such an event.

Update - 1987

Conclusion:

- Initially GE provided a contingency pool of fuel bundles for the eventuality cited in the recommendation. It became evident through operating experience that fuel failures did not cause unplanned shutdown so the contingency pool was discontinued.

- GE fuel performance in BWRs has been excellent. Normally planned refueling outages replace 20 to 30 percent of the fuel since it has produced its designed energy. Replacement of any failed fuel at this time can be accomplished with minimum effect on plant downtime.

Summary:

In the early 1970s, GE implemented a plan to have available a modest pool of fuel assemblies which were not targeted for any particular reactor, but would be available as contingency. By the mid-1970s, it became apparent that such a pool was not necessary since fuel had, by that time, demonstrated excellent performance in statistically significant quantities. That excellent fuel performance trend has continued to the present. Since the fuel experience has proven that the normal fuel replacement process is not significantly perturbed by isolated fuel failures, the contingency pool has since been phased out.

Supporting Information

Under normal circumstances, approximately 20 to 30 percent of the fuel assemblies in a reactor are replaced every 12 to 18 months during a planned shutdown for refueling. These fuel assemblies are replaced once they have produced their designed energy. A specified period of time is allotted for this replacement process during the refueling outage. In the event that some fuel assemblies fail during operation, these assemblies would be replaced during the refueling outage.

It should be noted that GE's fuel manufacturing operation has sufficient flexibility and capacity to produce a significant number of fuel assemblies on relatively short notice if necessary.

6.0 PRODUCT SERVICES IMPROVEMENTS

6.4 FAILED FUEL DETECTION

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that a more accurate means might be needed to detect fuel bundles with clad perforations. This would minimize the impact on reactor down time during an outage where testing is necessary to locate leaking bundles. It was recommended that a program be initiated to develop an improved failed-fuel sensor.

Update - 1987

Conclusion:

- An improved wet sipping system has been developed to check a fuel bundle's integrity quickly. Sampling for a discrete radioisotope concentration gives a precise indication of a failed bundle. In addition, significantly improved fuel performance has led to a reduced number of fuel failures.

- Failed fuel detection is now performed with minimal effect on outage time. In addition, fuel performance has achieved a reliability level of 99.98%, reducing the need to detect failure.

Summary:

GE has developed a combination of improved testing equipment, fuel design changes and testing criteria which has significantly reduced the impact of failed fuel detection on reactor down time.

Supporting Information

Several programs have been developed to evaluate means of improving the monitoring of fuel performance. These programs are ongoing and have the specific goal of continuously improving fuel surveillance techniques. Under these programs two improved fuel testing systems have been developed and implemented. These new systems have greater sensitivity than previous equipment.

Two additional factors have contributed to increased fuel testing sensitivity. These are:

1. A more clearly defined testing criteria based on fuel inspection sensitivity.

2. Improved fuel designs which have significantly reduced fuel clad perforations.

7.0 REGULATORY/LICENSING

7.1 ANTICIPATED TRANSIENTS WITHOUT SCRAM

Summary of Findings and Recommendations in Reed Report - 1975

An anticipated transient without scram (ATWS) involves the hypothesized occurrence of three sequential events: first, an abnormal plant transient; second, a limiting plant parameter reaches a pre-selected value which signals control rod insertion to shut down (scram) the reactor; and third, the control rods fail to fully insert because of some hypothesized common cause failure.

In 1973 the Atomic Energy Commission (now Nuclear Regulatory Commission) issued a Staff position report which concluded that some type of independent ATWS mitigation should be installed in new reactors. In response to this report, General Electric described a possible mitigation system using automatic insertion of liquid boron into the core to provide an alternate means of reactor shutdown in case of an ATWS event.

As part of the Reed review, concern was expressed over what the cost/benefit of this mitigation system would be, considering the extremely remote probability of an ATWS event. Also of concern were the potential plant startup delays or downtime due to mitigation system installation and the potential loss in electrical power generation due to possible inadvertent actuations of the automatic boron systems.

Update - 1987

Conclusion:

A GE study submitted to the NRC in 1976 showed that the BWR scram system possessed high inherent reliability and since the probability of an ATWS was so low, the additional automatic boron injection system was not necessary. In 1984 the NRC issued a ruling requiring that the BWR have an automatic recirculation pump trip, an alternate rod insertion system and manual initiation of the liquid boron system. BWR plants are in the process of installing these added devices on a schedule agreed to with the NRC.

Summary:

Consistent with the Reed review concern, GE initiated a study of the present BWR scram system, including potential common cause failures. This study, submitted to the Nuclear Regulatory Commission in September 1976, confirmed the previous GE position that the scram system possessed high inherent reliability, therefore obviating the need for an automatic boron injection system. This same study also identified areas where the scram system reliability could be even further improved if deemed advisable.

General Electric's position regarding ATWS was that the probability of the ATWS event was so low that no further plant modifications were required.

The NRC issued a rule on July 26, 1984 defining the requirements for resolving the ATWS issue. For the BWR this rule requires automatic recirculation pump trip, an alternate rod insertion system and manual initiation of the liquid boron system with increased flow capacity (equivalent to operating both pumps simultaneously). The NRC has concluded that this capability in conjunction with updated emergency operating procedures provide an acceptable solution to this concern.

7.0 REGULATORY/LICENSING

7.2 MARK III DYNAMIC LOADS

Summary of Findings and Recommendations in Reed Report - 1975

It had been determined prior to the Reed Report that substantial testing of hydrodynamic phenomena would be required for the Mark III containment. Suppression pool loading phenomena during a loss-of-coolant accident (LOCA) had been defined, but loads due to safety/relief valve (SRV) discharge needed further evaluation and perhaps mitigation. The NRC indicated that combined loads from LOCA and SRV might have to be considered and that structural changes to Mark III might be required. There was also a concern that some Mark III designs might not have adequate provisions against seismic-induced pool sloshing.

It was recommended that highest priority be given to resolving the containment questions and that, if necessary, manpower be reallocated to accomplish this goal. Further, it was recommended that a preliminary set of containment loads due to the combination of SRV operation and a LOCA event be developed. In addition, it was recommended that full scale testing be considered, if necessary, to resolve uncertainties.

Update - 1987

Conclusion:

- LOCA and SRV testing was completed in 1979 and a final loads report was issued in 1980.

- The NRC reviewed the Mark III LOCA and SRV hydrodynamic loads report and issued its findings. Overall NRC acceptance of Mark III containment was indicated by issuance of the GESSAR Final Design Approval and the Grand Gulf, Perry, Clinton and River Bend operating licenses.

Summary:

By early 1975, the NRC had modified its regulatory requirements and considered this subject as a generic issue. At the time of the Reed Report, the initial definitions of LOCA and SRV related hydrodynamic loads to meet these new regulatory requirements had been identified based on analytical models and small scale test observations. This information was being used as a basis for the initial design work while a large scale test program was underway to confirm the key containment response analytical models and obtain additional data on the dynamic conditions that occur within the suppression pool. This ongoing test program had been established in close cooperation with the NRC and was being closely monitored and reviewed by their technical staff.

Shortly after the Reed Report was issued, one of its recommendations was achieved when a preliminary Mark III Load Definition Report (LDR) was issued. This report was for use during the Preliminary Safety Analysis Report (PSAR) phase of Mark III design and licensing. This LDR incorporated a "quencher" device for mitigation of the SRV dynamic loads. This report was incorporated in GESSAR (General Electric Standard Safety Analysis Report) as Appendix 3B. Meanwhile, confirmatory test and analysis programs continued. The last of the LOCA and SRV testing was completed in 1979 with a final loads report published in February of 1980.

The NRC has reviewed the Mark III LOCA and SRV hydrodynamic reports and issued their findings in NUREG 0978 (8/84), 0802 (10/82), and 0763. Overall NRC acceptance of the Mark III containment design has been indicated in the GESSAR Final Design Approval (FDA) and the Grand Gulf, Perry, Clinton and River Bend operating licenses.

Supporting Information

The Mark III Program consisted of air and steam LOCA testing for a full-scale unit cell (8 degree sector) and 1/3 area unit cell. These tests, along with scaling studies, confirm the design loads specified for the Mark III containment. Confirmatory multivent steam tests, consisting of three unit cells of 1/9 area scale, were also performed. All testing was completed in late-1979.

The initial Mark III Program SRV load empirical models were developed in 1975 based on full and scaled tests. A muffler device (X-Quencher) was incorporated in GE's Mark III standard plant in 1975 and full-scale in-plant tests were performed at the Caorso plant in Italy in 1978-79 to confirm the loads that had been specified for design. The loads observed during this in-plant test were significantly less than those used for design, thus demonstrating the type of conservatism that was incorporated in the load definitions to achieve acceptance from the NRC.

Throughout the conduct of these technical programs the NRC played a significant role in the establishment of the final empirically based loads through their frequent changes in their requirements for the analytical models for generating dynamic loads and/or structural responses, the load combinations and methods of combining loads and the establishment of the acceptance criteria that the design had to meet. The major civil structure changes that were required to assure a containment design satisfactory to NRC were actually identified and being incorporated prior to the issuance of the preliminary LDR in 1975. These changes included raising the steam tunnel, adding pool swell deflectors, stiffening the steel containment shell in the area of the suppression pool and incorporating SRV quenchers with optimized SRV discharge line routing. There were substantial reanalysis efforts as refinements of the individual hydrodynamic loads were made during the completion of the testing activities. The final LDR was issued in 1980 for FSAR licensing reviews. The changes that were subsequently required involved modifications to address piping and component requirements, not the containments' structural integrity.

Other Reed Report issues such as jet impingement on the weir wall and seismic sloshing were evaluated and determined to be non-problems. Direct jet impingement on the weir wall and its asymmetrical effects have been examined and found to be acceptable. These asymmetric loads are specified in the LDR. Pool sloshing was examined analytically and testing at 1/30 scale was also conducted. Maximum wave heights and pressures were determined to be acceptable even with the 0.3g earthquake and the new Regulatory Guide 1.61 spectra.

7.0 REGULATORY/LICENSING

7.3 DYNAMIC BUCKLING

Summary of Findings and Recommendations in Reed Report - 1975

The architect-engineers had expressed confidence that they could specify an adequate Mark III steel containment provided there was not a substantial increase in static and dynamic loads. However, it was recommended that if the dynamic buckling behavior of the containment could not be satisfactorily modeled analytically, a 1/10 size physical test model should be built and appropriately tested.

Update - 1987

Conclusion:

BWR/6 plants have complied with the NRC interim criteria for evaluation of free standing steel containment buckling. The issue is not applicable to the majority of Mark III containments which are reinforced concrete containments with steel liners.

Summary:

The NRC has developed interim criteria for evaluating free standing steel containment buckling for plants undergoing licensing review. All Mark III free standing steel containments must demonstrate compliance with these criteria. The majority of Mark III containments are steel-lined reinforced concrete and are thus unaffected by a buckling concern.

Supporting Information

Section III of the ASME Code provides specific guidance on the treatment of buckling of steel containment vessels with simple geometrics and loadings. More complex geometry and loading conditions are covered in ASME Code Case N284[1] which was approved by the ASME in 1980. The NRC was not completely satisfied with the code case and developed in 1981 a set of interim buckling criteria[2] for evaluating plants undergoing licensing review. All Mark III free standing steel containments comply with these criteria.

The NRC, however, considers these criteria as interim and is working with the ASME and the technical community to develop a code case consistent with these criteria.

References:

1. Case N-284, Metal Containment Shell Buckling Design Methods, ASME II, Div. I, Class MC, August 25, 1980.

2. NRC Interim Criteria for Evaluating Steel Containment Buckling, June 21, 1982.

3. Appendix 3F (Dynamic Buckling for Containment vessel) to General Electric Company BWR/6 Nuclear Island Design Standard Safety Analysis Repo t (GESSAR II) Docket No. STN 50-447.

7.3 DYNAMIC BUCKLING

Summary of Findings and Recommendations in Reed Report - 1975

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Update - 1987

Conclusion:

BWR/6 plants have
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Summary:

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References:

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... in Reed Report - 1975

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... delays. Uncertainty existed in calculating radiation so
... radiation exposure to personnel due to the Mark III
... thought that exposure was probably increased over previous
... commended that the pool be covered to limit personnel exposur
... radiation. It was also recommended that there be further
... the Mark III containment relative to personnel exposure con

Update - 1987

Conclusion:

... employed the A/E firm C.F. Braun to analyze expected radia
... issued the final results and recommendations to its customers
plants have accumulated 15 reactor years of operation as of Ju
pool design has not been a cause of plant availability loss
personnel exposure.

Summary:

Design modifications have been evaluated and analyses have
resolve concerns of personnel exposure and reduced plant av
Mark III containment. Resulting modifications and incre
calculational methods give assurance that potential expo
personnel will be well within acceptable limits.

Supporting Information

GE implemented engineering studies to better model fission
subsequent in-plant transport. These studies considered t
routine leakage to the suppression pool via safety/relief v
the most severe expected transient in which complete react
might occur. An analytical effort was also undertaken for G
engineering firm, C.F. Braun, to model the Standard Rea
(STRIDE) and to recommend to GE appropriate balance of plant
reduce the radiological impact of SRV discharges to the supp

C.F. Braun completed an initial evaluation in 1976 and a final evaluation, which included an improved GE fission product release and transport model, in 1977. GE reviewed these evaluations, transmitted preliminary results and recommendations to its customers in 1977, and transmitted final results and recommendations in 1978. The recommended design modifications are:

o Incorporation of a suppression pool cleanup system for the removal of iodine and particulate activity,

o Provide a "clean air" shower over personnel exit locks inside the containment, and

o Route the exhaust from the Residual Heat Removal (RHR) system to the drywell.

A detailed feasibility study for retrofitting a pool cover was conducted in cooperation with C.F. Braun. It was concluded that because of practical design problems, such as dynamic structural loading, and licensing issues, a retrofittable cover was not practical. Noise test and analyses were conducted which indicated that the noise from SRV discharge would be well below OSHA requirements, further reducing the need for a cover. SRV leakage has also been reduced from the assumed value of 100 lbm/hr to 20 lbm/hr. This should further reduce the background radiation level.

There is also a low-low set relief valve design feature which has been incorporated in the BWR/6 design which reduces the number of pressure excursions within the reactor pressure vessel. This will reduce the exposure for any given power isolation event.

In addition to the above analyses and design modifications the plant operator should be able to re-establish steam flow to the main condenser for a number of events which initiated an isolation transient. As a consequence of re-establishing steam flow, the cumulative radiological consequences should be reduced from an equivalent of 2.5 isolation transients per year to 1 per year.

Although the original design changes ensured potential exposures of less than one-fourth of 10CFR20 limits, these additional changes further reduce the potential dose to the Mark III personnel.

7.0 REGULATORY/LICENSING

7.5 PLANT ARRANGEMENT

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that GE had not specifically considered the potential need for increased provisions for security, sabotage prevention and handling of mixed-oxide fuel in the design of the BWR/6 Mark III containment and plant arrangement. It was believed that sabotage protection might, in the future, become a more stringent NRC requirement in nuclear power plant design in the United States (as it was in Europe) and it was recommended that provision for such protection be considered by GE.

Update - 1987

Conclusion:

- Studies of the specific procedures for personnel access into controlled areas of the plant, industrial security and sabotage protection have been considered by GE and discussed with the NRC, ACRS and A/Es. However, as specified in the governing regulatory guide, the utility/owner is responsible for plant security and sabotage requirements.
- GE's submittal to the NRC on the GESSAR docket in 1975 on design considerations for reducing sabotage risk has been reviewed and accepted by the NRC and no unresolved issues exist on plant arrangement.

Summary:

As specified in the NRC regulatory guides, the utility/owner is responsible for providing the procedures and equipment to meet plant security and industrial sabotage protection requirements. However, studies of specific procedures for personnel access into controlled areas of the plant, industrial security and sabotage protection have been considered by GE during the Mark III Standard Reactor Island Design (STRIDE) and have been discussed with the NRC, the ACRS and with Architect Engineers (A/Es). The result of these studies is that the STRIDE design provides adequate consideration for personnel access for radiation exposure control, security and sabotage. GE's submittal to the NRC, on the GESSAR docket, on design considerations for reducing sabotage risk has been reviewed and accepted by the NRC and no unresolved issues exist.

Supporting Information

The STRIDE design has been subjected to detailed evaluation by the NRC, A/Es, utility engineers, independent review committees and study groups. The Mark III design has been demonstrated by these engineering evaluations to meet the requirements imposed by government requirements for industrial security and sabotage protection. GE has submitted on the GESSAR docket Appendix 1F "Design Considerations Reducing Sabotage Risk." This document has been reviewed and accepted by the NRC with no unresolved issues remaining. Application of mixed-oxide fuel to the light water reactor will require a change in the present government policies regarding reprocessing facilities in the United States.

2. NRC Interim Criteria for Evaluating Steel Containment Buckling, June 21, 1982.

3. Appendix 3F (Dynamic Buckling for Containment vessel) to General Electric Company BWR/6 Nuclear Island Design Standard Safety Analysis Report (GESSAR II) Docket No. STN 50-447.

7.0 REGULATORY/LICENSING

7.4 MARK III RADIATION LEVELS

Summary of Findings and Recommendations in Reed Report - 1975

There was a concern that the uncovered suppression pool in Mark III might cause increased personnel exposure due to radioactive release inside the containment via relief valve discharge and that plant availability might suffer because of access delays. Uncertainty existed in calculating radiation source terms and in projecting radiation exposure to personnel due to the Mark III open pool and it was thought that exposure was probably increased over previous designs. It was recommended that the pool be covered to limit personnel exposure to steam, noise and radiation. It was also recommended that there be further study and analyses of the Mark III containment relative to personnel exposure concerns.

Update - 1987

Conclusion:

GE employed the A/E firm C.F. Braun to analyze expected radiation exposure. GE issued the final results and recommendations to its customers in 1978. Mark III plants have accumulated 15 reactor years of operation as of June 1987. The open pool design has not been a cause of plant availability loss nor increased plant personnel exposure.

Summary:

Design modifications have been evaluated and analyses have been performed to resolve concerns of personnel exposure and reduced plant availability for the Mark III containment. Resulting modifications and increased precision in calculational methods give assurance that potential exposures to Mark III personnel will be well within acceptable limits.

Supporting Information

GE implemented engineering studies to better model fission product release and subsequent in-plant transport. These studies considered the consequences of routine leakage to the suppression pool via safety/relief valves (SRVs) and of the most severe expected transient in which complete reactor depressurization might occur. An analytical effort was also undertaken for GE by the architect/engineering firm, C.F. Braun, to model the Standard Reactor Island Design (STRIDE) and to recommend to GE appropriate balance of plant changes which could reduce the radiological impact of SRV discharges to the suppression pool.

C.F. Braun completed an initial evaluation in 1976 and a final evaluation, which included an improved GE fission product release and transport model, in 1977. GE reviewed these evaluations, transmitted preliminary results and recommendations to its customers in 1977, and transmitted final results and recommendations in 1978. The recommended design modifications are:

- o Incorporation of a suppression pool cleanup system for the removal of iodine and particulate activity,
- o Provide a "clean air" shower over personnel exit locks inside the containment, and
- o Route the exhaust from the Residual Heat Removal (RHR) system to the drywell.

A detailed feasibility study for retrofitting a pool cover was conducted in cooperation with C.F. Braun. It was concluded that because of practical design problems, such as dynamic structural loading, and licensing issues, a retrofittable cover was not practical. Noise test and analyses were conducted which indicated that the noise from SRV discharge would be well below OSHA requirements, further reducing the need for a cover. SRV leakage has also been reduced from the assumed value of 100 lbm/hr to 20 lbm/hr. This should further reduce the background radiation level.

There is also a low-low set relief valve design feature which has been incorporated in the BWR/6 design which reduces the number of pressure excursions within the reactor pressure vessel. This will reduce the exposure for any given power isolation event.

In addition to the above analyses and design modifications the plant operator should be able to re-establish steam flow to the main condenser for a number of events which initiated an isolation transient. As a consequence of re-establishing steam flow, the cumulative radiological consequences should be reduced from an equivalent of 2.5 isolation transients per year to 1 per year.

Although the original design changes ensured potential exposures of less than one-fourth of 10CFR20 limits, these additional changes further reduce the potential dose to the Mark III personnel.

7.0 REGULATORY/LICENSING

7.5 PLANT ARRANGEMENT

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that GE had not specifically considered the potential need for increased provisions for security, sabotage prevention and handling of mixed-oxide fuel in the design of the BWR/6 Mark III containment and plant arrangement. It was believed that sabotage protection might, in the future, become a more stringent NRC requirement in nuclear power plant design in the United States (as it was in Europe) and it was recommended that provision for such protection be considered by GE.

Update - 1987

Conclusion:

- Studies of the specific procedures for personnel access into controlled areas of the plant, industrial security and sabotage protection have been considered by GE and discussed with the NRC, ACRS and A/Es. However, as specified in the governing regulatory guide, the utility/owner is responsible for plant security and sabotage requirements.
- GE's submittal to the NRC on the GESSAR docket in 1975 on design considerations for reducing sabotage risk has been reviewed and accepted by the NRC and no unresolved issues exist on plant arrangement.

Summary:

As specified in the NRC regulatory guides, the utility/owner is responsible for providing the procedures and equipment to meet plant security and industrial sabotage protection requirements. However, studies of specific procedures for personnel access into controlled areas of the plant, industrial security and sabotage protection have been considered by GE during the Mark III Standard Reactor Island Design (STRIDE) and have been discussed with the NRC, the ACRS and with Architect Engineers (A/Es). The result of these studies is that the STRIDE design provides adequate consideration for personnel access for radiation exposure control, security and sabotage. GE's submittal to the NRC, on the GESSAR docket, on design considerations for reducing sabotage risk has been reviewed and accepted by the NRC and no unresolved issues exist.

Supporting Information

The STRIDE design has been subjected to detailed evaluation by the NRC, A/Es, utility engineers, independent review committees and study groups. The Mark III design has been demonstrated by these engineering evaluations to meet the requirements imposed by government requirements for industrial security and sabotage protection. GE has submitted on the GESSAR docket Appendix 1F "Design Considerations Reducing Sabotage Risk." This document has been reviewed and accepted by the NRC with no unresolved issues remaining. Application of mixed-oxide fuel to the light water reactor will require a change in the present government policies regarding reprocessing facilities in the United States.

7.0 REGULATORY/LICENSING

7.6 MARK I AND MARK II CONTAINMENT LOADS

Summary of Findings and Recommendations in Reed Report - 1975

Because of feedback from operating plants and changing NRC requirements relating to hydrodynamic phenomena, the Mark I and Mark II containments were undergoing extensive additional analyses required by NRC. Based on the input received from utilities evaluating preliminary LOCA and/or SRV dynamic loading projections, it was anticipated that modifications and retrofits could be required by NRC as a result of these analyses. It was recommended that available experienced, knowledgeable personnel be marshalled for the direction and execution of work necessary to accomplish this substantial analysis program. It was also recommended that necessary planning be done to minimize plant unavailability caused by any retrofit work that might be required.

Update - 1987

Conclusion:

GE established a centralized technical program management group to facilitate and control the program activities. In 1975 Mark I and Mark II owners established technical programs for addressing questions posed by the NRC. NRC acceptance of the work is evidenced in NRC Safety Evaluation of the Mark I Short Term Program in 1977 and the Long Term Program in 1980. In 1981 the NRC issued a regulatory guide addressing LOCA-related loadings and in 1982 a regulatory guide addressing SRV loads for the Mark II containment. The NRC was intimately involved in each of these containment test programs as they progressed to a conclusion. BWRs have made or are in the process of making modifications required by the NRC.

Summary:

To facilitate and control the program activities, GE established a centralized technical program management group. In the spring of 1975, the NRC sent letters to all utilities with plants utilizing a pressure suppression containment requiring information on the basis for design to accommodate LOCA and SRV dynamic phenomena. In response to these NRC inquiries the Mark I and II owners established technical programs for addressing the questions being posed. These test and analysis programs and plant modifications are now complete.

Supporting Information

Mark I Containments

The Mark I Owners Group was formed in April 1975 and immediately undertook a Short Term Program to assure that the plants could be operated safely while a more rigorous re-evaluation was performed. The Short Term Program involved the use of data from 1/12 scale 2-D Mark I LOCA tests and current SRV analytical models to perform structural analysis of each plant to a set of interim structural acceptance criteria established with the NRC. Various groupings of plants with the same design configuration were used to facilitate the evaluation and testing tasks. The program was successfully completed in August 1976. The NRC issued their Safety Evaluation of this program in December 1977 (NUREG-0408).

The Long Term Program got underway before the end of the Short Term Program. It consisted of a variety of test and analytical model development tasks. Additional scaled tests (1/4) were performed to obtain data on the pool swell portion of the LOCA event, a full scale 3-D test facility was constructed to obtain data on the steam condensation phase of the LOCA event, a unique quencher device for SRV loads was developed and in-plant tests were performed. The data from these tasks were used to develop emperically based load definitions and/or analytical models for predicting loads (i.e., loads on submerged structures). The tests were also used to develop load mitigation devices (pool swell deflector) or physical modification concepts (shortened downcomers and downcomer bracing) for use by those plants requiring such action to meet the final structural design acceptance criteria established with the NRC. The final Long Term Program tasks were completed in 1981, however, the Final Load Definition Report (LDR) was issued in March 1979. This LDR contained all the loads and/or methods needed for the utilities to complete the design assessments for the individual plants. The activities that were conducted after the LDR was issued were considered confirmatory and primarily involved additional steam condensation tests in the full scale 3-D test facility and responding to NRC questions. Like the Mark III program, the NRC was intimately involved in the planning, execution, and evaluation of each task undertaken by the Mark I Owners Group. They issued a Safety Evaluation Report (SER) of the Mark I Long Term Program in July 1980 (NUREG-0661). The Mark I Long Term Program was completed with the issuance of Supplement 1 to this SER in August 1982.

Mark II Containments

Like Mark I, the Mark II Owners Group was formed in the spring of 1975. The initial technical activity was the generation of a report similar to the Mark III LDR, a report which would quantify the LOCA and SRV phenomena to be used by the utility's architect-engineers (A/Es) to calculate the plant unique loads on their respective containment and internal structures. Using available applicable test data and analytical methodologies, GE worked with a technical committee comprised of A/Es to develop the Dynamic Forcing Functions Report (DFFR) which was issued in October 1975. The Mark II Long Term Program was to be comprised of the test and analysis tasks that would be needed to confirm the adequacy of the DFFR.

The Mark II Long Term Program consisted of a variety of test and analytical model development tasks. A full scale single vent LOCA test facility representing a unit cell of a Mark II containment, a full scale in-plant SRV test in a foreign Mark II plant (the same used by Mark III program), several small scale multi-cell steam tests and a full scale multi-cell segment test in a foreign test facility form the primary source of data for the confirmation and/or refinement of the loads and methods presented in the DFFR. In addition to this testing, substantial efforts were undertaken to justify analytical methods, load combinations, load combination methodology, etc. As information was developed through the execution of the program tasks, the DFFR was revised to reflect the new interpretation of loads or methods. As with Mark III, the major civil structure changes that were required to assure a design satisfactory to the NRC were identified early in the program. As subsequent re-evaluations were performed, when load refinements were made, most changes that were subsequently incorporated were made to address piping and component requirements, not the containment structural integrity.

Like the other containment programs, the NRC was intimately involved in and frequently changed its requirements for the planning, execution, and evaluation of each task undertaken by the Mark II Owners' Group. In July 1981 they issued NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria." This document addresses the LOCA related hydrodynamic loadings. SRV loads for the quenchers used in both Mark II and III is addressed in NUREG-0802 which was issued in October 1982.

7.0 RFGULATORY/LICENSING

7.7 NRC SUBMISSIONS

Summary of Findings and Recommendations in Reed Report - 1975

The Reed study group concluded that GE's policy was one of complying with NRC regulations, but that submissions of information requested by the NRC were sometimes only partially respo::sive or inadequately supported. It was recommended that General Electric establish a practice to further substantiate design requirements in licensing documentation and to highlight accompanying programs or tests that were necessary to substantiate submissions.

Update - 1987

Conclusion:

GE established a specific Safety and Licensing Management Group to ensure the quality, timeliness and completeness of material transmitted to the NRC. GE also established a system of approvals to ensure thorough review of submitted material. This provided a check against incomplete responses and premature transmittal of information prior to verification. All responses to NRC have been satisfactory prior to licensing of each operating plant.

Summary:

It was not uncommon for NRC questions to be unclear and on occasion there would be misunderstandings as to their intent. It was also common for an adequate response to a question to trigger requests for further information that the NRC had not originally contemplated. Also, answers that were responsive with respect to current design status at the time made, could appear inadequate in later years as a result of changes in NRC requirements, design development or industry developments. In other situations, GE engineers and NRC engineers might differ on the degree of detail necessary on a given issue. Although misunderstandings or disagreements might occur, GE at all times sought to fully respond to and resolve questions with the NRC.

Several major organizational and procedural changes which have been implemented in the past several years have corrected the type of situation identified.

Supporting Information:

The licensing process is characterized by the NRC staff asking many questions and the plant owner (applicant) providing written responses. Some of these responses were provided by GE as input to the owner. Written responses, however, were required. In the early phases of licensing, many questions were only asked verbally at meetings. Later, questions were formally transmitted by AEC (NRC) to the applicant. Later still, review sessions were conducted prior to receipt of formal questions to discuss their content and assure understanding of NRC needed information. This formalization of the process with time reflected a recognition of two problems. One was that questions were sometimes unclear or broad based and thus impossible to answer fully without extensive work. Secondly, the response to questions often resulted in changes to the application (the PSAR) and thus loss of the question content and so at times it was not clear when the questioning period was concluded. In addition, regulatory changes resulted in new requirements to old issues.

The following steps have been accomplished in addressing concerns about NRC submissions:

1. <u>Formation of Specific Safety & Licensing Program Management Group</u>

   This group of senior engineering people was formed to ensure the quality, timeliness and completeness of material transmitted to the Nuclear Regulatory Commission. The Group covers four main areas:

   A. Preliminary and Final Safety Analysis Report documents.

   B. Regulatory guides and Nuclear Regulatory Commission or industry standards.

   C. Responses to Nuclear Regulatory Commission questions.

   D. Standard documentation; for example, General Electric Standard Safety Analysis Report, Final Design Approval.

   This group is in an "overview" position organizationally and it provides a check against incomplete response or transmittal of information that is premature from a design verification standpoint.

2. <u>Establishment of a System of Approvals to Ensure Thorough Review of Submitted Material</u>

   Generally, material submitted for Nuclear Regulatory Commission review undergoes many levels of checking. For example, the responsible design engineer who generates the response obtains his manager's approval. This response is then reviewed by the lead engineers in engineering and licensing, the project licensing engineer, the project manager and the customer or architect/engineer. This chain of reviews has proven to provide both a technical and project oriented evaluation to ensure a technically correct and responsive submittal.

7.0 REGULATORY/LICENSING

7.8 ANTICIPATION OF REGULATORY CHANGES

Summary of Findings and Recommendations in Reed Report - 1975

The Reed study group felt that in some instances engineering organizations were not making adequate plans to incorporate probable regulatory changes, even though the licensing organization was aware that changes were likely. It was recommended that GE adopt the practice of systematically identifying future regulatory requirements and establish a program to implement those requirements which were reasonable or inevitable in a timely and economical fashion.

Update - 1987

Conclusion:

GE established a Safety and Licensing Program Manager responsible for regulatory guides and NRC/industry standards. The BWR design was reviewed against anticipated new NRC regulatory guides to identify potential changes and these were incorporated where practical.

Summary:

A substantial effort has been implemented to review the BWR design against NRC regulatory guides in an effort to identify potential changes due to anticipated regulatory requirements. Specific actions, such as the establishment of a special Safety and Licensing Program Manager responsible for regulatory guides and NRC/Industry Standards, have been implemented to address the concerns identified.

7.0 REGULATORY/LICENSING

7.9 'ERIOD OF SAFETY OF UNATTENDED REACTOR

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that the current industry standard used by NRC since 1973 required that all necessary safety actions for the first 10 minutes of a design basis event be automatic. Alternate requirements (longer time) existed in other countries and it was anticipated that NRC would be establishing new requirements. Thus it was recommended that GE determine a logical basis for operator response times and establish an implementation plan for whatever the new requirement might be.

Update - 1987

Conclusion:

GE BWR designs satisfy the industry standard on response times issued in 1984 by the American Nuclear Standards Institute (ANSI). In addition, BWR plants are in the process of complying with new NRC rules that are more limiting in NUREG-0737.

Summary:

An industry standard addressing this item, ANSI ANS-58.8 - 1984 (formerly ANSI N660), "Time Response Design Criteria for Nuclear-Safety-Related Operator Action" was published in September 1984. This standard specifies new time criteria to replace the existing criterion of 10 minutes. The criteria contained in this standard establish timing requirements for determining whether the nuclear safety systems that mitigate the consequences of design basis events may be initiated or adjusted by (a) use of operator action or (b) automatic protection system. The nuclear-safety-related operator action response times in these criteria are based on simulator measurements of operator performance to various anticipated operational occurrences and accident situations. The data collected was reduced using statistical methods and a 95% confidence level was utilized to assign the criteria.

The time response criteria given in this standard include conservative time margins, time delays and other restrictions to provide an adequate nuclear-safety margin for the purposes of system and plant design and nuclear-safety evaluations. GE BWR designs satisfy these criteria.

The 10 minute duration is still the standard timing requirement for determining whether operator actions can be used to mitigate the consequences of design basis events. However, there has been more emphasis on reducing the dependence on operator actions. For example, NUREG-0737, Item II.K.3.18 requires automation of the ADS system for events without drywell pressure. For this rule, the 10 minute operator action time is not applicable. BWR nuclear plants are being modified to meet this new rule.

7.0 REGULATORY/LICENSING

7.10 SAFETY SYSTEM REDUNDANCY

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that current Nuclear Regulatory Commission requirements provided that, for a postulated loss-of-coolant accident, the emergency core cooling system must furnish adequate core cooling with the assumption that the most critical piece of equipment in the emergency core cooling system fails to operate upon demand or is otherwise unavailable for use. This was called the N-1 criterion.

The German and Swiss licensing authorities required that the most critical piece of equipment be considered out of service for maintenance at the time of the accident and that the next most critical piece of equipment fails to operate upon demand. This was called the N-2 criterion.

NRC was known to be considering possible new requirements that might be in the direction of the European ones. Thus it was recommended that General Electric make a full study of the possible need for assuming that the two most critical pieces of the emergency core cooling system equipment would be unavailable following a loss-of-coolant accident. It was recommended that this study would utilize actual plant experiences.

Update - 1987

Conclusion:

NRC has maintained its position that the N-1 criterion (the most critical piece of equipment in the ECCS fails to operate upon demand or would be otherwise unavailable to use) is the appropriate conservative licensing basis. U.S. BWR plants meet all the NRC requirements based on the N-1 criterion. GE-supplied BWR plants outside the U.S. also meet the applicable requirements of national regulatory authorities. Because of NRC's position, a study of the N-2 criterion was unnecessary.

Summary:

The Nuclear Regulatory Commission, through issuance of technical specifications, places restrictions on the length of time a plant may be operated with equipment out of service. Additionally, the specification places increased surveillance requirements on parallel systems during these periods. Meeting these operating restrictions is more desirable than installing additional equipment to cover infrequent out-of-service periods.

Studies performed by General Electric demonstrate that the emergency core cooling system meets its availability goals taking into account both the probability of a failure upon demand and the probability of a system being out of service at the time of a postulated accident. These goals assure meeting acceptable safety limits.

Outside the U.S., appropriate design modifications are made as necessary to meet applicable national regulations.

7.0 REGULATORY/LICENSING

7.11 CORE CATCHER

Summary of Findings and Recommendations in Reed Report - 1975

A small study of a core catcher was recommended even though it was recognized that there was a low probability of an NRC requirement for such a device. The purpose of the study was to be able to respond to any NRC inquiry on this subject.

Update - 1987

Conclusion:

Evaluations performed by GE and others have shown that the addition of a core catcher to the BWR/6 Mark III design was not cost effective in reducing plant risk in the event of a severe accident. A core catcher was not incorporated into the BWR/6 Mark III design nor is it required by the NRC.

Summary:

Evaluations by the General Electric Company and others have shown that core catchers are not cost effective and do not reduce plant risk in the event of a severe accident.

Supporting Information:

As a result of the TMI accident, an item was identified in the NRC's Advanced Notice of Rulemaking associated with evaluations of proposed design modifications. Core catchers were included in the NRC's list of modifications. However, the NRC has not recommended the implementation of a core catcher.

GE performed detailed evaluations of severe accident risks as part of the GESSAR II Probabilistic Risk Assessment. As part of this effort GE also considered the impact of potential design modifications, including core catchers. The GE analyses showed that core catchers were not cost effective and would not reduce plant risk in event of a severe accident.

The results of the GE analysis are consistent with other recent evaluations that indicate the negligible risk reduction associated with core retention devices. A sizeable study was recently completed for the Industry Degraded Core Rulemaking (IDCOR) Program*. The study involved an evaluation of a number of core catcher designs for the IDCOR reference plants (which include both a Mark I and Mark III BWR). This study also found negligible risk reduction for core catchers.

* IDCOR Technical Report 20.1 Core Retention Devices, July 1983.

7.0 REGULATORY/LICENSING

7.12 FUEL TRANSFER ACCIDENT

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that there had not been an evaluation made of the consequences and probable causes of a potential fuel transfer accident in the Mark III containment and it was recommended that such an evaluation be done.

Update - 1987

Conclusion:

Fuel transfer accident causes have been evaluated for the BWR/6 Mark III. Contingency tooling has been designed and evaluated which is capable of removing the fuel assembly container and its cargo of one or two fuel assemblies from a stuck fuel transfer carriage. Extensive testing and Failure Modes and Effects Analysis were performed to verify reliability and safety adequacy of the designs. In 15 reactor years of BWR/6 operation no fuel bundles have become stuck in the fuel transfer tube under actual operating conditions.

Summary:

The potential fuel transfer accident in the Mark III involves the carriage containing a fuel bundle becoming stuck in the fuel transfer tube. The Mark III design has been evaluated and it has been determined that provisions for such a stuck fuel event are adequate.

Supporting Information:

Should a loaded fuel carriage become stuck in the transfer tube for any reason, contingency tooling has been developed which is capable of removing the fuel assembly container and its cargo of one or two fuel assemblies from the carriage and the transfer tube. Under all conditions of a transfer, the fuel will always be immersed in water. Makeup water can be made available under any condition.

A life test was completed with the original electrical system and hydraulic power unit as well as a BWR/6 inclined fuel transfer system components. A life test requirement was imposed on the vendors before the equipment was shipped. Interface control documentation is in place to provide full system integration on all components. Also all equipment has been thoroughly tested during the site system startup test programs.

Each transfer tube access point is suitably interlocked and shielded for personnel protection. Area radiation alarms are provided. Intrusion into any of these areas immediately alarms the control room and deactivates the transfer system.

A Failure Modes and Effects Analysis has been completed and no open items remain.

7.0 REGULATORY/LICENSING

7.13 OFF-SITE RADIATION EXPOSURE

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that new regulations were being drafted regarding the limits to be set on the permissible whole body radiation dose at the site boundary. While the Reed study found that GE was working effectively to minimize radiation dose to off-site persons, a concern was expressed over the proper control of direct or scattered radiation from equipment such as the turbine. It was recommended that GE assure that external radiation be well below the anticipated limit of 60 mrem per year at the appropriate site boundary, and that GE work with plant architect-engineers and consortium partners on basic methods to achieve this objective.

Update - 1987

Conclusion:

A typical site boundary dose from direct radiation is about 10% of the allowable dose permitted in the EPA regulations. Site boundary radiation levels are continuously monitored to assure compliance with regulatory requirements.

Summary:

At the time of the Reed Report, the NRC had a draft regulation which intended to limit the radiation exposure at the site boundary to 60 mrem per year. This was widely known by all utility applicants at that time. In the years since the report was written, plant equipment and shielding arrangements have been evaluated, and measurements have been made at operating BWR plants. Also the EPA has issued 40CFR190, which includes the subject radiation sources. The dose limits established in this regulation are being used in existing plants' operations and new plant designs.

Supporting Information:

EPA regulation, 40CFR190 issued in January 1977, applies to most fuel cycle facilities. With regard to power reactors, it specifies that the annual dose, from all sources and facilities on a given site, to any off-site person should not exceed 25 mrem whole body radiation and 75 mrem to the thyroid gland. The limits selected are well within the recommendations of national and international expert technical groups and are considered to meet the intent that radiation exposure should be maintained "as low as practicable" or "as low as reasonably achievable."

In 1975 direct radiation measurements were made around the Duane Arnold and Cooper operating plants, with participation by technical experts from the Health and Safety Laboratory (HASL, then a part of ERDA), the EPA, NRC and GE. As interpreted for application for new large BWR plant equipment arrangement, the subject radiation from the turbine building would be expected to produce a dose of about 8 mrem per full power year at a typical site boundary 2000 feet away. This may be compared to natural background radiation of over 100 Mrem per year.

In practice, it is found that the contribution to direct radiation from equipment located in other BWR plant structures is insignificant in comparison to equipment in the turbine building. The actual radiation exposure at any off site location is a function of the site size, the turbine building location on the site, the number and orientation of multiple units at the site and duration of occupancy around the site. Several operating plants have added some shielding in or around the turbine building to insure conformance with the new EPA limit.

While site layout and shielding subjects are usually within the scope of work of the utility and its architect-engineer, GE provides information and consultation as necessary with regard to radiation source terms.

7.0 REGULATORY/LICENSING

7.14 PLUTONIUM IN TURBINE

Summary of Findings and Recommendations in Reed Report - 1975

It was stated that plutonium had been detected on the inside of BWR turbines. It was noted that migration and concentration of plutonium in the balance of plant was not well understood nor had it been systematically evaluated.

Update - 1987

Conclusion:

EPRI initiated a three-year program in 1976 to study this issue. Levels of plutonium detected in operating plant turbines were very low. In addition it was concluded that the migration of plutonium throughout the balance of plant does not constitute a health and safety concern. The results of this EPRI program are known to the NRC and this concern is closed with the NRC.

Summary:

In 1976 EPRI initiated a three-year program to determine the concentration of transuranics and other radionuclides in the solid low-level radwastes from nuclear plants. Results are reported in EPRI NP-1494, Research Project 613, issued in August 1980. As part of this study, crud scrapings from inside a BWR turbine that had been in operation since 1962 were analyzed. The levels of plutonium detected inside the turbine were very low. In addition, the plutonium content of the various sources of radwaste indicates that migration throughout the balance of plant does not constitute a health and safety concern.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.1 ENGINEERING MANPOWER

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that additional engineering manpower was needed in several areas, such as system design and analysis, materials, reload fuel design and core management and engineering support to BWR services. It was considered probable that Mark I and Mark II reevaluation activities would impose additional requirements on engineering. It was also noted that responsibility for carrying through product design, development, test and production was diffuse.

Recommendations were made for manpower increases in areas with potential for expanded work scope and for assignment of clear-cut responsibility and accountability in product design and development efforts.

Update - 1987

Conclusion:

Engineering staff was increased by 14 percent from 1975 to 1978. The management system was modified and computer tracking implemented. Issues were resolved as planned, with more than 90 percent of engineering tasks completed on schedule.

Summary:

Engineering manpower was increased 14 percent between 1975 and 1978, providing necessary manpower to apply to design issues. Peaks in manpower requirements have been managed by purchasing external engineering subcontract services. In addition, the engineering staff was reorganized to reflect the technical needs and to focus attention on improved quality and design issues. Upon completion of the BWR/6 design effort and the successful resolution of design issues, the engineering manpower was reduced from 1978 levels consistent with the reduction of backlog engineering work scope.

Management systems were developed to provide measurement and control of resource application. Periodic program reviews by top management assured that appropriate resources were allocated to meet customer requirements and GE's quality objectives.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.2 COMPUTER FACILITIES

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that computer capability was inadequate to accommodate existing and future needs. It was recommended that the current system be evaluated and that hardware be added to double computational capacity within two years. It was also recommended that the operational system be improved to eliminate turnaround delays caused by peripheral systems inadequacies.

Update - 1987

Conclusion:

The computer facility was expanded and updated. Capacity in 1987 is almost five times greater than 1975.

Summary:

New and expanded computer capabilities were developed which successfully supported automation of the design process, easily accommodated running more accurate design computer programs, and were satisfactorily been used to meet the requests from regulatory bodies for increased numbers and complexity of safety analyses.

The 1987 installation includes distributed processing networks with powerful interactive work stations, clustered Digital Equipment computers, and access to the GE Cray computer. This replaced much of the previous batch mode CDC and Honeywell computer capacity. The local network computer capacity provides 4.8 times the 1975 capacity, and access to the Cray computer.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.3 PROCURED EQUIPMENT CONTROL

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that there was insufficient in-house design expertise on the design of vendor-supplied equipment. It was difficult to control such designs and to monitor the materials and process aspects of their production. To improve availability and gain better control of vendor-supplied equipment, it was recommended that GE institute greater standardization, engineering selection and control of vendors, and that management make a commitment to reliability programs for procured equipment. It was also recommended that a program be initiated to acquire ownership of designs for critical valves and pumps.

Update - 1987

Conclusion:

Improved performance of procured products has been achieved through reorganization of engineering and QA and expanded qualification programs.

Summary:

The engineering function, in direct support of vendor-supplied equipment, was reorganized in March 1976. The organization was component oriented rather than system oriented. In this way there was a ready exchange of technology and experience associated with specific components. In addition, personnel were upgraded to improve technical capabilities at all levels.

Increased involvement by design engineering, development of a structural and disciplined engineering organization and the strengthening of qualification programs resulted in improved product performance and plant availability.

On critical components, GE reviewed design detail drawings in addition to the normal outline drawings and supporting stress reports required by Codes. Extensive qualification programs were completed. Engineering procedures were changed to require Quality Assurance to review and sign off on all materials requests prior to issue to ensure current definition of quality requirements.

Department procedures were rewritten to better define engineers' responsibilities and to control engineering changes.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.4 STANDARDIZATION

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that BWR/6 standardization was not being adequately addressed at the hardware level and that program direction to achieve greater standardization was lacking. The resulting multiple designs of fuel and reactor components created cost and control problems in manufacturing and made it more difficult to assure that licensing requirements were satisfied. Plant availability and capacity could also be affected by lack of standardization. It was recommended that a clear-cut program be instituted to establish a reference BWR/6 design, with the objectives of reducing the number of plant designs and improving plant availability and capacity.

Update - 1987

Conclusion:

Attention of management and engineering was focused on standardization during the BWR/6 final design period. A Change Control program was emphasized.

Summary:

An organizational component was created to direct an increase in the standardization emphasis throughout the organization and to focus management and engineering attention on utilizing standard system and hardware designs. This component was maintained from August 1975 through January 1978, and during this period several programs were implemented. In January 1978 the organization was dissolved since it had accomplished its objective and the majority of the requisition plants had progressed to a design stage where most of the equipment decisions had been made.

Supporting Information:

A computerized standardization report was created to identify the top 1800 design documents for the BWR/6 standard plant and for all of the BWR/6 requisition plants. The BWR/6 standard design was granted Preliminary Design Approval by the NRC in December 1975. This report identified the system designs and equipment designs which existed and focused attention on those designs that had significant variation between plants. Using this report, lead system engineers and component designers created action plans to complete the Standard Plant Design, reduce the natural trend toward custom design of individual plants and promote application of the standard designs to all requisitions. Specific action items were identified for individual designers. In addition, the report was periodically produced and released to line managers to be used as one measure of engineering performance.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.5 TEST FACILITIES

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that additional investment in testing was needed to assure equipment design qualification before installation and operation in order to improve plant reliability. It was recommended that additional facilities be developed to test new equipment, provide information on equipment performance and identify any areas where reliability could be improved. Attention was directed to flow induced vibration testing and hydraulic testing of components, as well as to testing of control rod systems and the containment pressure suppression system.

Update - 1987

Conclusion:

Action to upgrade and expand component testing has been accomplished. A number of major testing facilities were built and testing programs were developed and completed. Equipment designs have been tested and qualified. (See also Topics 4.1 - Stress Corrosion Cracking, 7.2 - Mark III Dynamic Loads and 7.6 - Mark I and II Containments.)

Summary:

Major testing facilities, programs developed and completed: High Flow Hydraulics, Fuel Transfer, Control Rod Drive handling, Control Rod Drive Test, Recirculation Flow Control Valve, Main Steam Isolation Valve, Safety/Relief Valve, Reactor Water Cleanup Pumps, Pipe Test, Containment Test, and BWR Services Training.

Supporting Information:

Steps taken to upgrade and expand testing have included:

1. Construction of a test facility which is a 60 degree sector of a full-size BWR/6 core and associated internals. This facility, the High Flow Hydraulic Facility, became operation in 1978 and was used to evaluate vibration of BWR/6 internals. This evaluation has allowed elimination of vibration problems prior to startup of a BWR/6.

2. Construction of a separate facility to test the BWR/6 inclined fuel transfer system was completed in 1976. This system transfers fuel from the fuel storage pool in the fuel building to the upper fuel storage pool of the reactor building. Over a two year period the facility has been used to perform a complete qualification of the entire system prior to startup of the lead BWR/6. An actual production system was tested before shipment to a reactor site.

The test verified that a production system performed as intended. Following the qualification tests, an accelerated, 40-year life reliability test was performed when the system was cycled 12,000 times to check component interfaces and long-term performance and operational characteristics. Such tests allowed improvements in the design which will help insure more reliable system operation.

3. Construction of a separate facility to test the control rod drive handling equipment was completed in 1978. The facility was built to simulate the undervessel portion of the BWR/6 with the goal of reducing CRD replacement time, and hence radiation exposure, during CRD maintenance. Each set of production equipment was tested prior to site delivery.

4. Modification of the Control Rod Drive Test Facility to allow testing a new concept in control rod drive operation. This concept, called "ganged rods," originated with BWR/6 and tests were performed to qualify the new control system operation.

5. Construction of a separate facility to test the recirculation flow control valve. This test facility, completed in 1978, using actual flow control valves, determined the performance characteristics of the valves under simulated flow conditions and determined the life of various moving parts of the valves. 7000-hour tests (corresponding to four years of expected operation) verified operation within performance specifications.

6. Construction of a test facility to study the cause of leakage across the main steam isolation valve (MSIV) seat was completed in 1978. Past MSIV leakage has required maintenance in the field which has contributed to plant unavailability. Tests on the main steam isolation valve in the new facility have allowed General Electric to determine the causes of leakage and to develop a tool design to alleviate the problem.

7. Modification of a facility to allow testing of safety/relief valves became operable in 1978. The test objective was to study the growth of leakage rates over long time periods and to evaluate materials for safety/relief valve seats which would minimize erosion and, therefore, improve valve reliability. 5000 hour tests provided leakage rate data and examination of material samples resulted in the best combination of disc and nozzle seat materials.

8. Modification of a test facility to allow testing of reactor water cleanup pumps to determine pump failure mechanisms. Pump design changes have resulted and test operation is being monitored to determine the affect of design changes on pump operation.

9. Construction of a BWR Services Training Facility in 1980 serving a dual role as both a testing and training complex. The only facility in the world capable of full-scale simulation of nuclear plant refueling and maintenance activities, it acts as a proving ground for developing and testing of various tools, servicing procedures and equipment installation techniques in a radiation-free environment.

10. Pipe test facility constructed to allow simultaneous testing of a large number of pipes with diameters of 4" to 16" under carefully controlled temperature, stresses, and water chemistry conditions.

11. Numerous containment test facilities. These included full scale tests as well as scale model tests and allowed all pressure suppression phenomena to be thoroughly evaluated.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.6 QUALITY ASSURANCE

Summary of Findings and Recommendations in Reed Report - 1975

The staffing and organizational status of quality assurance in engineering had not allowed the effort to be fully effective. Increasing governmental requirements were putting added burdens on the quality assurance operation, and there was an unsatisfactory balance between engineering design operating work and design assurance work. Communications problems were compounding the situation. It was recommended that the quality assurance function be upgraded across the Division and that their responsibilities for the total scope of product design be affirmed. It was also recommended that the quality assurance component review engineering procedures for simplicity, clarity and consistency; develop training courses to familiarize designers with the procedures; and furnish an overview of the design verification process. General Managers were encouraged to explore means for resolving communications problems.

Update - 1987

Conclusion:

Regular QA audits were increased and a positive response in engineering communications was achieved. Quality functions were reorganized, reporting level and staffing increased, and procedures were revised and implemented.

Summary:

The engineering quality functions were reorganized, reporting level was increased, and the staffing level was increased several fold. Procedures which define the engineering quality assurance program have been rewritten to greatly strengthen the program. Design assurance activities are stressed and supported by engineering management at all levels. A formal system to assure awareness and application of changing regulatory requirements has been implemented. The audit program is accepted constructively by the engineering organization, assuring management that the upgraded quality program is being implemented throughout the engineering functions.

Supporting Information

A Quality Assurance and Reliability organization has been established and reports at a high enough level so that it can conduct design assurance activities with independence from the line design organizations, and so that its position clearly affirms that its design assurance activities apply to the total scope of engineering design/development activities.

The Quality Assurance (QA) groups within the Quality Assurance and Reliability organization are assigned responsibility for managing a comprehensive engineering quality assurance program which includes development and documentation of the engineering quality system, training and indoctrination of personnel to the quality assurance system practices and procedures, and auditing to ensure adequacy of and compliance with the quality assurance system. The directly applied quality assurance resources increased several-fold in the 1975-1980 time to accommodate the increased level of business. The size of the organization has since decreased due to the decrease in the number of backlog plants.

A Design Review group formed within the Quality Assurance and Reliability organization has contributed significantly to the total impact on design assurance work through its own independent reviews of design adequacy and by fostering application of formal, disciplined design reviews throughout the responsible design groups as well.

The Engineering Operating Procedures (EOPs) which define the engineering quality assurance program have been completely rewritten to both simplify them and strengthen control over engineering work. For example, significant controls have been added in the areas of materials, computer codes, test control and procurement of engineering services. In addition, an Engineering Internal Audits function has been established within the Engineering Services organization to focus on engineering compliance with EOPs.

The system for assuring that engineering personnel are aware of changing governmental design requirements has been formalized by creating specifications of Regulatory Requirements and Industrial Standards (which include General Electric Product Safety Standards and General Electric alternate methods for meeting Regulatory Guide requirements). These specifications are controlled within the General Electric design and change control system and are formally applied to to plant projects. Any project unique requirements are also identified in the controlled system and are available "over-the-counter" to engineering personnel affected by them.

The engineering quality assurance audit program has been expanded significantly, with approximately 10 comprehensive formal audits conducted per year. In addition, approximately 4 comprehensive formal audits are specifically focused for compliance with EOPs. Findings by auditors are received constructively and a positive attitude toward quality throughout the engineering organization, together with strong program support from management, have led to the resolution of previous communication difficulties.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.7 SYSTEMS DESIGN ORGANIZATION

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that there was not an adequate systems engineering organization and that procedures for BWR systems design reviews needed improvement. It was recommended that a Systems Engineering Function be established and implemented to delineate system requirements. It was also recommended that an independent and rigorous BWR Systems Design Review discipline be established to provide comprehensive evaluation of each generic BWR design at critical points in the program.

Update - 1987

Conclusion:

Lead systems engineers were established and they provided strong guiding influence on BWR design.

Summary:

The BWR work breakdown structure was reexamined and the product was subdivided into its 31 functional systems. A Lead System Engineer was then appointed to take primary responsibility and authority for the correct and complete design and the quality of performance of each of the systems which make up the BWR. The overall product performance and the organizational development and functional direction of the Lead System Engineering group were assigned to a newly created "composite office" consisting of Senior Engineers.

Documents and procedures required for design review have been fully developed and are in use.

Supporting Information:

The composite office and Lead System Engineers provided a strong guiding influence on the BWR design. In addition to developing more active interaction with plants in the pre-operational and startup phases and with operating plants, they created a complete set of System Design Specifications and a Composite Plant Specification which localized all product level and systems design requirements into one place. These documents were used as a basis for design review and design verification work and were an efficient guide for all design work on BWR/6. A series of thorough system design reviews were also conducted to assure that the systems would perform their intended functions.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.8 FIELD EXPERIENCE OF DESIGN ENGINEERS

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that design engineers were isolated from field construction and operating plant problems. It was recommended that a program be established to increase opportunities for hardware and design engineers to obtain closer contact with on-line plant operations.

Update - 1987

Conclusion:

The interchange of assignments between design and operating engineers improved design knowledge in the field and familiarity by design engineers with field issues.

Summary:

Actions taken have fallen into three categories:

- o Creation of an operating engineers' organization to place permanent senior GE engineers at operating reactors.
- o Increased use of engineers to do needed on-site work and to bring back useful experience for design input.
- o Improvement of communication systems for bringing operating experience into the design organization.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.9 RELIABILITY PROGRAM

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that, while BWR performance in availability and capacity was competitive, improvements were needed. It was determined that there was no organizational component responsible for across-the-board reliability/ maintainability management, nor was there an overall program specifically focused on this area. Goals of 85% availability and 75% capacity factor were seen to be reasonable and achievable for the BWR/6 Mark III. It was recommended that the reliability engineering function be upgraded and organized as a separate entity and that a highly visible, group-wide program be established to set goals and implement them on a measurable basis and an expedited schedule. It was recommended that a formally structured approach in design and development be undertaken to achieve high plant reliability.

Update - 1987

Conclusion:

Each area of concern identified was addressed; a program manager was assigned to each major activity, test facilities and programs were implemented, computerized data storage and retrieval was begun; and a formalized organization was put in place.

Summary:

GE actions taken to upgrade BWR reliability can be separated into two phases. Specific, relatively short-term actions were put in place to assure that plants being designed and delivered would operate at a level of reliability that would be satisfactory to our customers. The framework of a long-term program was also put in place to continue to improve the operating performance of BWRs. Reliability analysis of the BWR/6 Mark III design has shown that the proposed reliability/availability goals are achievable.

Supporting Information:

The short-term program was defined and implemented through the following steps:

1. Design areas of concern, from a reliability standpoint, were identified by the responsible engineering organization.

2. Programs for improvement were defined for each area.

3. A Corporate commitment was made to carry out the design and development, to provide added test facilities and conduct qualification tests, and to implement the resulting improvements into the backlog hardware.

4. A program manager was assigned for each major activity and detailed plans were developed and implemented.

These programs have produced results that have been reflected in the hardware.

The long-term program begun at the same time consists of the following elements:

1. A computerized system was put in place to store and retrieve data on the causes of plant outages and forced power reduction.

2. GE has located a Senior Operations Engineer at BWR plants to improve feedback on plant performance (see Item 8.8).

3. A Reliability Engineering organization was established. The Availability Engineering group within that organization analyzes the operating plant performance and feeds back performance data to the design organizations. They also assist the design organization with reliability studies in support of improved designs.

4. Reliability improvements, as they are developed and qualified, are implemented into new hardware and, in many cases, operating plants.

5. Design review activities have been expanded significantly.

As a result of the overall reliability program, the capacity factor of operating BWR plants has improved significantly.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.10 EXTERNAL SUPPORT OF DEVELOPMENT

Summary of Findings and Recommendations in Reed Report - 1975

It was recommended that R&D programs with other organizations should be promoted and established. Other organizations would include licensees, utility customers, U.S. and foreign governments, etc.

Update - 1987

Conclusion:

Increased support from outside agencies was achieved. A BWR Development Board was established to improve the exchange of technical information with licensees. BWR owners' groups were also established.

Summary:

There has been a substantial increase in cooperative research and development programs with DOE, EPRI and NRC; with BWR owners groups; with GE licensees, and with foreign BWR vendors.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.11 AUXILIARY POWER SYSTEM INTEGRATION

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that power supplies for nuclear safety systems were highly reliable systems with redundant independent distribution and that they had very small impact on plant availability (.005% compared with 1% attributable to all power supplies). Non-safety power supplies had a minor impact on total plant unavailability - less than 1% of 27.6% total plant unavailability. It was recommended that additional coordination of interface needs be developed between GE and the responsible customer/A-E design organizations in order to foster possible increases in non-safety power system availability. It was also recommended that responsibility for power supplies for GE systems be centralized.

Update - 1987

Conclusion:

Improved information exchange resulted in incorporation of latest requirements and latest improvements.

Summary:

Responsibility for power supply for GE systems was centralized within the GE organization and the overall system responsibility shared jointly among customer, A/E and GE. Exchange of information assured incorporation of the latest requirements and design improvements into power supply systems in order to increase availability.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.12 FUEL CYCLE COST AND FUEL PERFORMANCE

Summary of Findings and Recommendations in Reed Report - 1975

Three areas were identified as having the potential for reducing fuel cycle cost and improving overall fuel performance. It was recommended that the following approaches be considered:

1. In-core fuel management strategy which would avoid movement of fuel from a relatively low power zone into a relatively higher power zone during refuelings.

2. Six-month refueling intervals.

3. Use of spent fuel in place of natural uranium in the core blanket of BWR/6 initial cores.

Update - 1987

Conclusion:

Preconditioning (PCIOMR) resolved concern about fuel shuffling. No utility interest in six-month cycles was expressed. Use of discharged fuel as a blanket proved to be impractical.

Summary:

The approaches identified have been studied to determine their potential for providing fuel cycle cost savings and improving fuel performance. Conclusions of this work were:

1. Limiting the number of fuel bundles moved from a low power location to a relatively higher power location during refueling was to reduce PCI failures. If the non-barrier fuel is preconditioned, the movement of fuel to any location is acceptable. Preconditioning of non-barrier fuel is now standard practice. For barrier fuel preconditioning is not necessary.

2. The objective of six-month fuel cycles rather than annual was to reduce fuel cycle cost. However, the availability of the nuclear system increases with longer fuel cycles, reducing total power generation costs. This effect outweighs benefits of fuel cycle cost savings from short cycles. An increasing number of utilities are now extending to 18 and 24-month fuel cycles. There is no utility interest in six-month fuel cycles.

3. The use of low exposure discharged fuel in the initial cores of BWR/6s could result in fuel cycle cost savings. However, the practical problems of obtaining, shipping and handling appropriate discharged fuel bundles have outweighed fuel cycle cost benefits.

8.0 ENGINEERING QUALITY AND COST IMPROVEMENTS

8.13 ENGINEERING/MANUFACTURING COMMUNICATION

Summary of Findings and Recommendations in Reed Report - 1975

The large geographical separation between the Nuclear Engineering Division located in San Jose, California and the manufacturing facilities located in both Wilmington, North Carolina and Memphis, Tennessee was identified as a concern. It was recommended that a study be done to determine the best division of engineering and manufacturing personnel among the various facilities.

Update - 1987

Conclusion:

Alternatives were studied, and improvement in communications and organizational reporting relationships changed in lieu of relocation of people. Follow up of actions taken led to the conclusion that adequate liaison was established.

Summary:

While problems of geographic distance from suppliers are recognized, GE believes the primary need is to maintain a close relationship among all elements of the engineering organization. For this reason, engineering capability is concentrated at the San Jose site. In the specific instance of the Wilmington facility, measures have been taken to ensure that close contact is maintained. These measures include manufacturing facility engineering offices reporting to San Jose, joint management meetings, participation by San Jose engineers in resolving manufacturing site problems and efficient telephone system communications. Due to a decline in shop loading at the Memphis reactor vessel manufacturing facility, GE no longer maintains a business agreement with CBI Nuclear and our engineering office in that facility has been discontinued.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.1 C&I FIELD CHANGES

Summary of Findings and Recommendations in Reed Report - 1975

A concern was expressed over the number of on-site changes required in C&I equipment due to shipment of components prior to completion of · nal design. It was noted that implementing these changes in the field resulteu in QC manpower becoming too involved in installation and required a lengthy documentation cycle. It was recommended that procedures and practices be implemented in C&I to substantially reduce field changes and to reduce the time required to document those changes when they were necessary.

Update - 1987

Conclusion:

Programs were established and implemented to address these concerns.

Summary:

A number of programs were established to address the concerns identified in the Reed Report. These programs were effective in reducing C&I field changes in both design and procurement areas and streamlining the documentation process for necessary changes. Programs were developed to further improve implementation and tracking of necessary changes.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.2 PRODUCTIVITY AT REACTOR SITE

Summary of Findings and Recommendations in Reed Report - 1975

The Reed study group recognized areas that were opportunities to improve construction efficiency at reactor sites.

Update - 1987

Conclusion:

Although GF's role at construction sites was quite limited compared to the utility and principal constructor(s), GE took steps to improve efficiency within its scope of responsibility.

Summary:

The following improvements were implemented:

1. A formal system was used to track the status of implementation of required field changes at the construction sites.

2. An audit was conducted in 1978 of cleanliness practices at the construction sites. Subsequently, the instruction on cleaning and cleanliness control was revised to improve its implementation.

3. Wherever practicable, installation of reactor internals into the pressure vessel was transferred from the construction site to the vessel manufacturer's plant for improved fabrication control and quality assurance.

4. Installation instructions for GE-supplied hardware and technical direction at the plant sites were reviewed, revised and modified based on the continuing experience gained during the construction of plants.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.3 COFRENTES CONSTRUCTION - U.S. DESIGN SCHEDULE

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that the construction schedule of the Cofrentes (Spain) BWR/6 was following immediately behind the design schedule of the BWR/6 standard plant. There was a concern that the construction of this plant might be impacted by unresolved licensing issues. It was recommended that construction be scheduled to avoid major changes resulting from possible regulatory requirements.

Update - 1987

Conclusion:

GE successfully supported the construction schedule of the Cofrentes plant.

Summary:

In January 1984, the preoperational test program was completed at Cofrentes and the customer submitted a request to the Spanish Regulatory Agencies for initial fuel loading. In May, an emergency/evacuation drill was successfully completed. On July 24, 1984, the plant received the license to start initial fuel loading. The first fuel bundle was loaded August 6, 1984 and fuel loading was completed August 21, 1984. On September 5, Spanish authorities authorized initial heatup to 5% power. The 5% power testing was completed on October 1 and authority was obtained for 20% testing. Initial turbine generator synchronization occurred on October 14. Twenty percent (20%) testing was completed October 29. Testing up to 50% power (test condition 2) was started October 30. The plant was put into commercial operation in 1985.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.4 TECHNICAL DIRECTION OF INSTALLATION SCOPE

Summary of Findings and Recommendations in Reed Report - 1975

A concern was expressed that there were ambiguities in the scope of technical direction to be provided by GE in the installation of equipment. It was recommended that a clearer definition of GE's responsibilities be developed.

Update - 1987

Conclusion:

Contract language for GE technical direction is supplemented by a formal presentation by GE to the utility staff one year prior to the start of work. There have been no significant difficulties over the scope of technical direction to be provided by GE. GE's scope of work is now complete for all U.S. construction sites except one.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.5 ARCHITECT ENGINEER (A/E) INFORMATION INPUT

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that GE project management did not always have adequate access to A/E drawings and schedules. It was recommended that this needed to be more clearly specified in contracts.

Update - 1987

Conclusion:

A system was successfully implemented which assured that GE had adequate access to A/E drawings and schedules.

Summary:

Required A/E interface information is now contractually provided to GE through a set of controlled documents known by numbers prefix "A50" (Customer Supplied Information) on the project's Master Parts List. This includes kickoff information needed for the PSAR, as well as specific interface information for each system for which General Electric has design responsibility. Customers have accepted this approach and use the A50 procedure to provide GE with interface information.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.6 PROJECT MANAGEMENT INFORMATION SYSTEM

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that plans were under way to streamline and update the current project information system to provide "real time" cost and schedule measurement and control for better project control. It was recommended that this effort be expedited.

Update - 1987

Conclusion:

GE implemented the improved system.

Summary

GE implemented an up-to-date computer system for cost and schedule control on project work. This Nuclear Program Control (NPC) System coordinated the financial and schedule requirements for engineering, procurement and manufacturing. The basic elements of NPC became operational in 1977.

The work assignment portion of the NPC data base was improved to provide overnight updates as opposed to previous twice monthly updates. This enhanced GE's capability to handle short cycle tasks.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.7 CONTROL AND INSTRUMENTATION MANUFACTURING SPACE

Summary of Findings and Recommendations in Reed Report - 1975

It was determined that more space was required for C&I Manufacturing and it was recommended that this space be provided in San Jose.

Update - 1987

Conclusion:

The manufacturing space was acquired and the work completed.

Summary:

A 100,000 square foot facility near the headquarters of GE in San Jose was leased in 1976 for C&I Manufacturing. In 1978, it was expanded to a total of 150,000 square feet. This facility provided adequate space for the assembly and testing of control room systems. The work was completed and shipped and this facility was retired.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.8 BACKLOG PLANTS

Summary of Findings and Recommendations in Reed Report - 1975

It was recommended that GE examine opportunities and cost benefits for renegotiating backlog orders in order to simplify commitments or to incorporate standard plant features on a block basis.

Update - 1987

Conclusion:

Limited changes were negotiated to simplify commitments and assure the reliability of supplied equipment.

Summary:

Within the limitations of contractual commitments, GE did renegotiate some limited changes to simplify design and assure that equipment which was supplied met high standards of quality and reliability.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.9 CHANGE CONTROL

Summary of Findings and Recommendations in Reed Report - 1975

It was noted that a Change Control Board had been organized to monitor and authorize design change and to assume responsibility for appraising the overall effect of design changes on all projects. It was felt that it was too early to assess the effectiveness of the Board at that time.

Update - 1987

Conclusion:

The Change Control Board proved to be an effective means of monitoring and authorizing design changes.

Summary:

The Change Control Board became the recognized body to authorize changes from the Engineering organization. Board membership included all functions in GE. The change control process assured that change recommendations were packaged to identify all impacts and all implications of the change.

9.0 PROJECT MANAGEMENT EFFECTIVENESS

9.10 INVESTMENT EMPHASIS

Summary of Findings and Recommendations in Reed Report - 1975

It was recommended that future investments should concentrate on availability/capability improvements as opposed to shop capacity.

Update - 1987

Conclusion:

GE made substantial investments in engineering facilities and training facilities aimed at availability/capability improvements (See Items 6.2 and 8.5).

Summary:

Major investments have been made in engineering test facilities to improve the BWR technology base and to improve product performance. Some of the major facilities include the Pipe Test Lab, High Flow Hydraulic Facility, Valve Test Training Facility, Steam Sector Test Facility, the Inclined Fuel Transfer Test Facility, the Equipment Environmental Qualification Test Facilities, Control and Instrument Training & Diagnostic Facility, Under Vessel Training Facility, the Maintenance Training Facility and the Tulsa Simulator Facility for BWR/6 Operator Training. Investments have also been made in the Wilmington manufacturing lab to provide capability to upgrade and better control fuel product quality.

10.0 FUTURE PRODUCT OFFERING

Summary of Findings and Recommendations in Reed Report - 1975

Several Reed Report recommendations were made relative to future product offerings, including a small development effort for the next generation of BWR design.

Update - 1987

Conclusion:

GE continued to evaluate future product offerings which ultimately led to an Advanced Boiling Water Reactor design.

Summary:

GE joined with several foreign firms to complete a feasibility and plant definition study for an advanced design in the 1978-1979 time frame. Subsequently, GE worked under contract with Tokyo Electric Power Company (TEPCO) to complete a technical evaluation including preliminary engineering for the new design in 1985. In 1987 TEPCO announced its intention to build two plants in Japan utilizing the advanced design with GE supplying the nuclear boiler, initial fuel and turbine generator. The units are scheduled for commercial operation in 1996 and 1998.

APPENDIX 1

REED REPORT REFERENCES TO (UPDATE REPORT)

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* No response provided.

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APPENDIX 2

RECOMMENDATIONS

	REED REPORT		UPDATE REPORT
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	59	R-5	3.2
	59	R-6	3.2
	60	R-7	3.4
	61	R-8	3.4
	61	R-9	3.3
	61	R-10	3.3
	62	R-11	3.3
	63	R-12	3.6
	64	R-13	3.5
	65	R-14	3.1
	65	R-15	3.5
	66	R-16	3.4
	66	R-17	8.12
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	89	R-7	7.2
	90	R-8	7.2
	91	R-9	7.2, 8.1
	92	R-10	7.4
	93	R-11	7.3
	94	R-12	7.6
	94	R-13	6.2
	96	R-14	8.3

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	101	R-2	2.10, 1.6
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	106	R-5	4.2
Production, Procurement and Construction	107	R-1	8.4
	108	R-2	6.3
	109	R-3	4.5
	109	R-4	3.4
	110	R-5	3.4
	110	R-6	9.10
	111	R-7a	8.4
	111	R-7b	2.6,2.7,2.8,8.3
	112	R-7c,d	8.3
	113	R-73	6.2
	113	R-7f	8.9
	113	R-8	8.3
	114	R-9	2.6
	115	R-10	9.1
	116	R-11	9.7
	116	R-12	8.13
	117	R-13	9.2, 6.2
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Quality Control Systems Overview	121	R-1	8.6
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Management/Information Systems	125	R-1	8.7, 8.9
	126	R-2	8.4
	126	R-3	8.8
	126	R-4	1.1
	127	R-5	9.3
	127	R-6	9.1
	128	R-7	9.4
	128	R-8	9.5
	128	R-9	9.6
	129	R-10	8.9
Regulatory Considerations	131	R-1	7.7
	132	R-2	7.8
	133	R-3	1.5
	134	R-4	7.9

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	136	R-6	6.4
	136	R-7	2.7, 2.8
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	138	R-10	7.10
	139	R-11	2.1
	140	R-12	7.11
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	144	R-3	6.2, 8.9
	145	R-4	6.2

APPENDIX 3

SUBTASK REPORTS

FINDINGS

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	15	4.1.6	8.1
	16	4.1.7	8.2
	18	4.2.1	1.1
	19	4.2.2	1.1
	20	4.2.3	1.1
	22	4.2.4	5.1
	24	4.2.5	5.1
	26	4.2.6	2.1
	28	4.3.1	5.1
	29	4.3.2	5.1
	30	4.3.3	1.1
	31	4.3.4	1.1
	32	4.3.5	7.1
	33	4.3.6	5.2
	34	4.3.7	3.1
	35	4.3.8	2.15
	36	4.4.1	1.2
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	41	4.4.5	7.5, 7.12
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	25	IV.B.1.b	6.1, 6.3
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	31	IV.B.2.a	3.2
	32	IV.B.2.b	3.2
	33	IV.B.2.c	3.4
	35	IV.B.2.d	3.4
	37	IV.B.3.b	3.3
	39	IV.B.3.c	3.3
	41	IV.B.3.d	3.6
	43	IV.B.4.a	3.5
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	48	IV.B.4.d	3.4
	49	IV.B.4.e	8.12
	50	IV.B.4.f	3.7
	50	V.A	4.5
	53	V.B	2.3
	54	V.C	3.8
Electrical, Control and Instrumentation	7	1	5.2
	10	2	2.15
	12	3	5.3, 3.6
	14	4	2.4
	15	5	2.5
	26	1.2	2.17
	27	1.3	2.6
	29	1.4	5.4
	32	3.2	2.5
	34	4	2.5
	35	5	*
	36	6	2.5, 9.7
	40	7	2.5
	43	8	8.11
	60	9	8.9, 2.5
	67	10	
	69	11	8.7
Mechanical Systems and Equipment	5	3.1.1	7.2
	11	3.1.2	7.2, 7.3
	13	3.1.3	7.2
	14	3.1.4	7.3
	15	3.1.5	7.4
	16	3.1.6	7.2
	17	3.2.1	7.6
	18	3.2.2	7.6
	19	3.3.1	2.13
	21	3.3.2	2.6
	24	3.3.3	2.14
	25	3.3.4	2.7
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	28	3.3.6	2.9
	30	3.4.1	1.4
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	44	3.6.1	8.3

* No response provided.

Section	REED REPORT Page	REED REPORT Number	UPDATE REPORT Section Number(s)
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	19		1.5, 4.1
	22		4.6
	23		1.5
	26		2.2
	28		2.16
	30		2.16
	31		8.3
	33		2.6
	35		2.11
	36		4.5
	39		2.3
	40		1.5
	42		2.13
	43		8.3
	46		4.2
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	51		4.6
	52		4.2
	53		8.6, 9.9
	54		4.7
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	56		4.4
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	58		9.6, 4.2
	60		4.2
	63		1.5
	68		1.5, 7.4
	72		1.5
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	81		1.5
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	88		2.12

* No response provided.

Section	REED REPORT Page	REED REPORT Number	UPDATE REPORT Section Number(s)
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	4	3	8.13
	5	4	4.5
	6	5	9.7
	7	6	3.4
	8	7	6.3
	9	8	3.4
	11	9	8.3
	14	10	2.6
	15	11	8.3
	16	12	6.1
	17	13	9.2
	19	14	9.1
Quality Control System	ES-1	1	8.6
	ES-1	2	8.6
	ES-1	3	8.9
	ES-2	4	8.6
	ES-3	5	8.6
Management/Information Systems	4	1.1	8.7, 8.9
	6	1.2	8.4, 5.1
	7	1.3	8.8
	8	1.4	1.1
	9	1.5	9.3
	10	2.1	9.1
	12	2.2	9.4
	13	3.1	9.5
	14	3.2	9.6
	15	4.1	8.9
Regulatory Considerations	11	2.1.1	7.7
	13	2.2.1	7.8
	15	2.2.2.1	7.13
	16	2.2.2.2	1.5
	18	2.2.2.3	7.9
	19	2.2.2.4	7.5
	21	2.2.2.5	6.4
	23	2.2.2.6	4.1
	24	2.2.2.7	2.7, 2.8
	24	2.2.2.8	7.5
	26	2.2.2.9	1.5
	27	2.2.3.1	7.10
	28	2.2.3.2	2.1

* No response provided.

	REED REPORT		UPDATE REPORT
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	32	2.2.4.3	2.10, 1.6
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	5		8.9
	7		8.9
	9		5.1, 8.9, 3.2, 2.7 2.8, 4.1
	10		8.9
	11		6.2
	14		8.4
	18		8.9
	22		8.9
	36		8.9, 5.1, 3.2
	48		8.9
	50		8.9
	96		8.9
	98		5.1, 4.1, 8.9, 3.2, 2.7, 2.8
	116		6.2, 8.9
	122		6.2
	130		8.4
	137		8.4

* No response provided.

APPENDIX 4

SUBTASK REPORTS

RECOMMENDATIONS

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	49	5.2.5	1.1
	49	5.2.6	3.3
	49	5.2.7	7.2
	50	5.2.8	7.4
	50	5.3.1	8.7
	53	5.3.2	10.0
	53	5.3.3	8.9
	54	5.3.4	8.1
	54	5.3.5	1.1
	54	5.3.6	1.3
	55	5.4.1	8.2
	55	5.4.2	5.3
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	27	3	3.2
	28	4	8.12
	31	5	3.2
	32	6	3.2
	33	7	3.4
	35	8	3.4
	38	9	3.3
	38	10	3.3
	39	11	3.3
	41	12	3.6
	43	13	3.5
	45	14	3.1
	47	15	3.5
	48	16	3.4
	49	17	8.12
	50	18	3.7
	51	19	4.5
	53	20	2.3
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	11	2	2.15
	13	3a	5.3
	13	3b	3.6
	14	4	2.4
	16	5	2.5
	26	1.2	2.17
	29	1.3	2.6
	33	3.2	2.5
	34	4	2.5
	41	7	2.5
	44	8	8.11
	65	9	8.9
	68	10	
	71	11	8.7
Mechanical Systems and Equipment	5	3.1.1	7.2
	11	3.1.2	7.2
	13	3.1.3	7.2
	14	3.1.4	7.3
	15	3.1.5	7.4
	16	3.1.6	7.2
	17	3.2.1	7.6
	18	3.2.2	7.6
	20	3.3.1	2.13
	23	3.3.2	2.6
	25	3.3.3	2.14
	26	3.3.4	2.7
	27	3.3.5	2.8
	28	3.3.6	2.9
	30	3.4.1	1.4
	42	3.5.3	6.2, 8.9
	45	3.6.1	8.3
Materials, Processes and Chemistry	7	V.A	4.1
	12	V.B	1.5
	18	V.C	2.10, 1.6
	23	V.D	4.2
	27	V.Ea	2.16
		V.Eb	4.3
	28	V.Ec	4.2
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	22		4.6
	23		1.5
	27		2.2
	29		2.16
	30		2.16
	31		8.3
	33		2.6
	35		2.11
	39		6.2
	30		2.3
	40		1.5
	42		2.13
	43		8.3
	46		4.2
	50		4.2, 8.5
	51		4.6
	52		4.2
	53		8.6, 9.9
	54		4.7
	55		4.2
	56		4.4
	58		*
	59		9.9
	60		4.2
	65		1.5
	69		1.5
	73		1.5
	79		1.5, 4.1
	83		1.5
	87		2.12
	90		2.12
Production, Procurement and Construction	2	1	9.10
	3	2	8.4
	4	3	8.13
	5	4	4.5
	6	5	9.7
	7	6	3.4
	8	7	6.3
	9	8	3.4
	11	9	8.9, 8.3, 2.6, 2.7, 2.8
	14	10	2.6
	15	11	8.3

* No response provided.

Section	REED REPORT Page	REED REPORT Number	UPDATE REPORT Section Number(s)
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	17	13	9.2
	19	14	9.1
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	ES-5	2	8.9
Management/Information Systems	4	1.1	8.7, 8.9
	6	1.2	8.4, 5.1
	7	1.3	8.8
	8	1.4	1.1
	9	1.5	9.3
	10	2.1	9.1
	12	2.2	9.4
	13	3.1	9.5
	14	3.2	9.6
	15	4.1	8.9
Regulatory Considerations	13	2.1.1	7.7
	14	2.2.1	7.8
	15	2.2.2.1	7.13
	18	2.2.2.2	1.5
	19	2.2.2.3	7.9
	21	2.2.2.4	7.5
	23	2.2.2.5	6.4
	23	2.2.2.6	4.1
	24	2.2.2.7	2.7, 2.8
	25	2.2.2.8	7.5
	26	2.2.2.9	1.5
	28	2.2.3.1	7.10
	29	2.2.3.2	2.1
	30	2.2.4.1	*
	31	2.2.4.2	*
	33	2.2.4.3	2.1, 1.6
	35	2.2.4.4	7.11
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	12		6.2
	15		8.4
	19		8.9
	49		8.9, 6.2
	116		6.2, 8.9
	123		6.2
	131		8.4

* No response provided.

STATUS REPORT OF

POTENTIAL SAFETY ISSUES DISCUSSED IN THE

1975 NUCLEAR REACTOR STUDY (REED REPORT)

Prepared for Dr. Bertram Wolfe
Vice President and Chief Scientist
Nuclear Energy Operations
General Electric Company

June 1987

FOREWARD

The purpose of this Status Report is to review the items of potential safety significance discussed in the Nuclear Reactor Study, a 1975 General Electric Company report which assessed the preliminary design of General Electric's BWR/6 reactor and associated systems. The study was directed by Dr. Charles E. Reed, and is frequently referred to as the Reed Report.

This Status Report has been prepared at the request of Dr. Bertram Wolfe, Vice President and Chief Scientist of General Electric's Nuclear Energy Operations, in response to several newspaper reports which suggest that there may have been safety issues discussed in the Reed Report which were not properly resolved or otherwise addressed. The findings were then reviewed with Dr. Wolfe and compiled in this Status Report. The final conclusion of this Status Report is that all such items have been resolved in the twelve years since the Reed Report was completed. The report does not discuss any safety issues that may have arisen or been resolved since the Reed Report was written in 1975.

Status Report of Potential Safety Issues Discussed in the <u>1975 Nuclear Reactor Study (Reed Report)</u>

One of the many strengths of the General Electric Company is its depth of expertise in almost every technical field. This expertise is frequently called upon to perform independent reviews of major GE technical projects. Company experts not connected with the project are formed into a technical committee with the charter to examine the project in order to identify and explore areas of risk and to provide guidance and perspective to company management. On rare occasions such reviews lead to major restructuring or abandonment of a project. In most cases the committee's insights result in emphasis or reemphasis on specific technical areas, programs to address committee concerns, and/or an exploration of new approaches suggested by the committee in its findings.

The purpose of the Reed study was to assure that GE's new BWR/6 product line of reactors would operate with the highest achievable level of quality and reliability. The study was conducted at an early stage in the design process of the BWR/6 in order to permit the recommendations of the task force to be included in the final design of the equipment to be furnished to GE's customers. Consistent with the practice throughout the nuclear industry, the BWR/6 design included the newest product advances for safety and efficiency with the detailed engineering work to be completed prior to the completion of the construction phase. Because of the long duration of nuclear plant projects, advanced concepts could be made available prior to completion of detailed design. This was true of the BWR/6 where the first U.S. unit went into commercial operation in 1985, ten years after completion of the Reed Report.

The Reed task force made critical comments about a number of areas of BWR/6 project and design approaches. They made numerous recommendations for improvement of the BWR/6, many of which were based on programs already underway. These recommendations were reviewed by General Electric management which, where appropriate, strengthened existing programs and initiated new programs to implement the recommendations.

The BWR/6 plants in operation have proven the BWR/6 to be a very successful design. These plants have shown performance above the average performance of earlier BWR designs. Their availability (percent of the time during a year the plant is available to generate power) has been 80% compared to a BWR fleet average of about 67%. The investment GE made in implementing the recommendations of the Reed Report task force played an important role in this success.

The purpose of a review such as the Reed Study is to forcefully present committee conclusions and guidance to the responsible management. Many of the key areas involve matters of judgment, and a number of conclusions by the committee may, in fact, be incorrect. The intent is to provide timely information candidly presented for evaluation and action by the responsible management. In order to encourage both candor and timeliness, studies such as the Reed Study are generally maintained as company proprietary documents with no external circulation and limited circulation within the company. Nevertheless, as described below, the Reed Report has received significant reviews both inside and outside of General Electric; in particular matters potentially bearing on reactor safety have been examined in detail.

The Reed Report was completed in 1975. The Report was not a safety study as noted by the Reed study group in the Report itself, which included the following statement:

> The Nuclear Reactor Study Group concentrated on reviewing opportunities for improvement in the availability and capability factors of the BWR plants. Although in the course of the Study Group's review, nuclear safety aspects were considered, this Study was not a safety review. However, the Study Group found no reason to believe that applicable safety requirements are not being met for operating BWR plants or will not be met for future BWR plants.

Since the Report reviewed all aspects of the BWR/6 design, however, General Electric's Nuclear Safety and Licensing organization reviewed the report in detail to assure that any safety issues which might have been discussed were being properly addressed by GE and the NRC. That review resulted in the identification of 27 safety-related items. These items were than assessed and it was determined that all of the items were previously known and were being properly addressed in the continuing design of the BWR/6 or in other programs. Additionally, it was determined that all items were previously known to the NRC or did not require that formal notification be given to the NRC.

In 1976 proceedings before the Joint Committee on Atomic Energy of the Congress of the United States (JCAE), Dr. Reed described the Report. At that time, two senior members of the NRC's technical staff also reviewed the entire Reed Report in detail at GE's offices. The NRC advised the JCAE that this review confirmed that the Report did not identify new safety issues, nor were there instances of significant safety concerns which had not previously been reported to the NRC.

In December of 1977, Congressman Dingell, Chairman of the Subcommittee on Energy and Power of the House Interstate and Foreign Commerce Committee, requested that the NRC review safety issues again and advise him of the status of any safety issues addressed in the Report. In response to a request from the NRC, GE prepared a status report on the 27 issues which had been previously identified. In preparing that status report, the reviewers combined four of the prior issues into two issues because they were essentially the same items. Thus, the final number of issues was 25. The GE review concluded that there were no safety issues which had not been resolved or which were not otherwise being addressed in programs of which the NRC was aware. NRC subsequently completed its review of these issues and reported to Congressman Dingell that all issues had either been resolved or acceptable solutions were available and the issues would be resolved in ongoing NRC programs or in the appropriate plant licensing proceedings.

Beginning in 1979, portions of the Reed Report were provided to intervenors and NRC's Atomic Safety & Licensing Boards where questions were raised about safety issues discussed in the Report. Additionally, the NRC retained the copy of the full Reed Report which was provided in one of these proceedings in 1979. The NRC has that copy in its possession today.

In all of the reviews and proceedings described above, the determination was made that there were no items of potential safety significance mentioned in the Reed Report which had not been resolved or which were not otherwise being properly addressed.

The remainder of this Status Report addresses each of the 25 items of potential safety significance previously identified and updates the status of those issues today.

1. DEGREE OF COMPLETION OF BWR/DESIGN

Item

The Reed Report noted that the BWR/6 design in many instances was incomplete.

Discussion and Status

The BWR/6 was still in the early stages of design at the time of the Reed Report, which was one of the important factors in performing the study at that time. The licensing process is based on the premise that construction permits are issued by the Nuclear Regulatory Commission without complete or final detailed design information. The initial application for a construction permit is accompanied by the Preliminary Safety Evaluation Report ("PSAR") which pursuant to NRC regulations provides the "preliminary design of the facility," 10CFR 50.34(a)(3). Later, following issuance of the construction permit, the Final Safety Analysis Report ("FSAR") is filed in conjunction with the operating license application. In 1975, BWR/6 plants were in the process of seeking construction permits based upon preliminary designs. The General Electric BWR design practice being followed at the time of the Reed Report established the preliminary design based on experience, engineering judgment, and analytical sensitivity studies.

The design of the BWR/6 has been completed for some time. Any plants incorporating this design have received licenses only after Nuclear Regulatory Commission review of their respective FSARs which document pertinent final design information.

Thus, based on the above, this matter was always in conformance with regulatory requirements.

References

1. General Electric Standard Safety Analysis Report (GESSAR): Docket No. STN 50-550, Final Design Approval 7/27/83, Amendment 1 8/9/85, Amendment II 9/22/86.

2. Grand Gulf Nuclear Station Updated Final Safety Analysis Report, Docket No. 50-416, Operating License 6/16/82.

3. River Bend Final Safety Analysis Report, Docket No. 50-458, Operating License 11/20/85.

4. Perry Nuclear Power Plant 1&2 Final Safety Analysis Report, Docket No. 50-440, Operating License 3/18/86.

5. Clinton Power Station Unit 1, Final Safety Analysis Report, Docket No. 50-461, Operating License 4/10/87.

2. AMOUNT OF MARGIN BETWEEN DESIGN CALCULATIONS FOR THE CORE AND OPERATING LIMITS

Item

The Reed Report noted that the BWR/6 core nuclear and thermal hydraulic design margins were not large enough at the preliminary design stage to cover all of the following factors:

a. Method uncertainties and new information from continuing qualification of design methods;

b. Future requirements arising from completion of system design;

c. Allowances for variation in operator requirements such as fuel failures, load following, reactor outages, etc., and

d. Adaptation of the generic design to particular contractual requirements, manufacturing requirements and other plant sizes.

It was also noted that one or some combination of the following consequences might result:

a. Power derating up to 15% during a portion of the cycle because of excessive power peaking required for operating flexibility or in order to continue to meet operating limits at rated power with high void coefficients and design scram reactivity values.

b. Reactivity shortfall, because of need for end-of-cycle maneuvering allowance, or possible requirement for a more conservative scram performance at end of cycle. In equilibrium cores, significant reactivity shortfall was thought probable.

c. Regulatory delays because of the need to show ability to meet licensing limits.

Discussion and Status

During the completion of the BWR/6 design, a number of planned improvements in the preliminary design were made to assure that there would be more than adequate margins. The methods used for the establishment of operating limits for the BWR/6 core design have been fully qualified and, as part of the NRC licensing process, operating thermal limits have been established. Compliance with the resulting operating thermal limits specified in the Technical Specifications for each plant assures sufficient safety margins. The stated concerns of power derating of up to 15% and significant reactivity shortfall were operational considerations and never represented a safety concern.

2. AMOUNT OF MARGIN BETWEEN DESIGN CALCULATIONS THE FOR CORE AND OPERATING LIMITS (Continued)

Based on over 15 reactor years of BWR/6 operating experience to date, more than adequate design margin has been demonstrated. In fact, BWR/6 has so much margin that improvement programs permitting higher power levels and greater operating flexibility than originally sought have been licensed by the NRC and plants are operating in these modes now.

In summary, core operating margins are in fact larger than anticipated in the early design analysis.

References

1. NUREG-0151, USNRC, Safety Evaluation Report, GESSAR-251 Nuclear Steam Supply System Standard Design, Dock No. STN 50-531, March 1977, Section 7.6.4, Recirculation Pump Trip System, page 7-27.

2. NEDE-24011-P, "Generic Reload Fuel Application," Section 5.2.2.3, Rod Withdrawal Limiter System, page 5-23.

3. NEDO-24154, Vol. 1 & 2 , "Qualification of the One-Dimensional Core Transient Model for BWRs," dated October 1978.

4. NEDE-24154-P, Vol. 3, "Qualification of the One-Dimensional Core Transient Model for BWRs", dated October 1978.

5. (Same as Reference 1), Fast Scram, page 7-13.

3. IMPACT OF COLD SHUTDOWN MARGIN OF BWR/6 CORE DESIGN

Item

The Reed Report noted a concern over whether the then current BWR/6 calculated cold shutdown margin for the equilibrium core would be adequate to demonstrate that the licensing requirement was met for shutdown margin with a stuck rod. The concern was that this might cause early shutdown and unavailability while taking corrective action. Further, the calculated value for cold shutdown margin for BWR/6 did not include the bias factor which might be obtained from field data or partially burned gadolinia cores and took credit for crud buildup.

Discussion and Status

The design margin for shutdown reactivity has traditionally been 1% delta K/K. The design strikes a balance among economic, technical and operating considerations. As experience with gadolinia cores is accumulated, the data is factored into the design predictions being used for current designs. As a design is completed, the specific enrichments and gadolinia concentrations and distributions are specified to assure adequate cold shutdown margin. The GE design requirement is still 1% delta K/K. The initial cores usually have more margin than this. Burnup characteristics of gadolinia have been monitored in operating plants and adjustments have been made to the design codes as appropriate.

Crud buildup is not taken into account in the initial clean core calculation but is included in the equilibrium cycle when crud becomes a factor. Ongoing data from gadolinia cores will continue to be factored into the design to assure adequate shutdown margin for equilibrium cores.

The final design is described in detail in each plant's Final Safety Analysis Report. This report has been reviewed and approved by the NRC prior to issuance of the operating license. Based on over 15 reactor years of BWR/6 operating experience the ability to meet and exceed the shutdown margin requirements has been demonstrated. Considering all BWR core designs to date, shutdown margin demonstration tests have always been successful.

This issue has been satisfactorily resolved with the NRC.

4. IMPACT OF END-OF-CYCLE SCRAM REACTIVITY INSERTION RATE IN CORE FULL POWER LIFE

Item

The Reed Report discussed the importance of improved end-of-cycle scram reactivity. Shortfall problems and several possible solutions under consideration within General Electric included possible use of a system called prompt relief trip, a device to trip the recirculation pumps, and fast scram. It was also pointed out that derating of plant power was an alternative to implementation of one or more of these possible solutions.

Discussion and Status

Technical Specification limits on steady state minimum critical power ratio and on many other parameters are issued by the Nuclear Regulatory Commission based upon analyses performed by General Electric on systems currently installed in BWRs. Compliance with these limits by BWR operators ensures that all operation takes place under conditions that have been analyzed for safe operation. Compliance with the Technical Specification limits on minimum critical power ratio assures safe operation. Derate, if needed, would be applied rather than exceeding these limits.

For GE's BWR/6 product line plants, a more rapid insertion of the control rods was accomplished by design modifications to the scram system. For all product lines, measured scram rate and recirculation pump trip provide adequate end-of-cycle scram reactivity margin.

The experience gained from over 15 reactor years of BWR/6 operation shows large margins to safety limits and that more than adequate end-of-cycle scram reactivity is available. Plant derates have not been required and, in fact, BWR/6 plants have been licensed and operated at power levels greater than those originally sought.

This issue has been satisfactorily resolved with the NRC.

Reference

1. NUREG-0151, Safety Evaluation Report, GESSAR-251 Nuclear Steam Supply System Standard Design, Docket No. STN 50-531, March 1977, Section 4.2.4, Functional Design of Reactivity Control Systems, pages 4-7 to 4-11.

5. LONG TERM EFFECT OF RADIATION ON CORE INTERNALS

Item

The Reed Report noted uncertainties in then-current estimates of radiation and corrosion degradation to BWR/6 core internals which might affect lifetime of these components. The concern was that core internals might have to be replaced prior to the end of the 40-year design lifetime to provide assured structural integrity for continued operation.

Discussion and Status

The potential for radiation and corrosion degradation to core internals has always been a design concern for all nuclear power reactors, and the BWR/6 design also had to address this issue.

The core internals, (i.e., core support structures) of the BWR/6 have been evaluated to determine if there are areas where radiation degradation could present a potential service lifetime of less than the design life of 40 years.

The uncertainties associated with radiation degradation of core support structures are related to neutron fluence induced property changes which potentially occur in stainless steel, the material used in the components.

If the calculated fluence values used for design life extend into the range where material properties change rapidly with further increased fluence values, a potential for shorter component design lifetime could exist. With this consideration the design of the components in the core support was based on available neutron fluence data, and analyses which demonstrated that the components would have sufficient design margin. In addition, evaluations of radiation degradation potential for stainless steel material continue to be performed as additional neutron fluence data are provided from planned surveillance programs and examinations of operating components.

In the evaluation of BWR/6 core support structures, two areas were identified where the neutron fluence levels could be greater than the concern threshold of 5×10^{20} neutrons per square centimeter. The areas identified were the central region of the top guide and the mid-plane of the shroud.

Although it is extremely unlikely that significant structural degradation could occur, the consequence of gross failure of the identified components was evaluated to assure that safe shutdown of the plant could be achieved. Analyses indicate that there is sufficient margin in the hydraulic control rod drive to insert rods to achieve shutdown even

5. LONG TERM EFFECT OF RADIATION ON CORE INTERNALS (Continued)

with channel interference or loss of positive interchannel spacing. In addition, the Standby Liquid Control System installed in all BWR plants is designed to achieve safe shutdown in the event the control rods become inoperable.

In conclusion, analysis indicates that BWR core internals should last their design lifetime and continued evaluation of potential degradation of reactor components is part of the ongoing in-service inspection programs carried out at all U.S. nuclear plants, both BWRs and PWRs.

This item has been satisfactorily resolved with the NRC.

6. DEGREE OF PROOF OF ACCURACY OF TRANSIENT DESIGN METHODS

Item

The Reed Report noted that the then current method used to calculate BWR transients, although comprehensive in their treatment of the overall Nuclear Steam Supply System, were approximate in their treatment of the reactor core. They were also essentially untested with respect to their treatment of the central features of the system, the coupled nuclear hydraulic process. In view of these uncertainties, the concern was that the allowed margins being used at the preliminary design stage might not be sufficient to assure acceptability and capability.

Discussion and Status

General Electric undertook a study and review plan with the Nuclear Regulatory Commission related to core design and physics evaluation methods. The results of these studies were transmitted to the Nuclear Regulatory Commission via topical reports.

The Reed Report suggested that further model verification be performed. Since the time of the Reed Report, pressurization transient tests have been conducted in two operating BWRs and stability tests have been conducted in four other operating BWRs. These tests were performed under controlled conditions. In parallel with these tests a more sophisticated transient stability computer model known as ODYN has been developed and verified with this plant data. Analyses have demonstrated that the margins included in the present model and the input values used with the model provided sufficient margin to accommodate the coupling of nuclear and hydraulic effects.

The computer model being used and its application technique have been reviewed and approved by the NRC. There are currently no open issues with the NRC concerning verification of models used to analyze BWR operational transients.

This issue has been satisfactorily resolved with the NRC.

References

1. NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the GE BWR," February 1973 and Amendments 1 and 2.

2. NEDO-21506, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," January 1977.

3. Letter dated October 25, 1977, from E.D. Fuller (GE) to Denwood F. Ross (NRC), Subject: General Electric Proposal for Changes in Licensing Basis Transient Model.

4. NEDO-24154-A, Vol. 1, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," August 1986.

7. IMPACT ON FUEL INTEGRITY OF REDUCED MODERATOR TEMPERATURE DUE TO EQUIPMENT FAILURE

Item

The Reed Report noted the following: the loss of feedwater heating or the accidental activation of the high pressure core spray or high pressure coolant injection systems could produce power transients long enough to cause increased leakage from the fuel. There was a concern that these transients might be more severe in this regard than pressurizat on transients because they last long enough for pellet clad interaction failure to occur.

The study also stated that the loss of feedwater heating also presses GETAB* limits. To stay within these limits in BWR/4, there was concern that the scram flux trip point might have to be reduced to 115% of full power from its then-present setting of 120%; the use of a thermal power monitor in conjunction with this tighter trip setting might be needed to avoid false scrams; the transient might lead to a significant loss of capacity factor in the future, since conditioning to overpower levels might violate thermal limits. The need for preconditioning arises from the possible long-term duration of these transients (10 minutes or more) which would allow pellet/clad interaction to occur.

Discussion and Status

At the time of the Reed Report, some BWRs were operating at less than rated capacity due to fuel failures partially attributable to a phenomenon known as pellet clad interaction. A study was carried out to determine the safety significance of pellet clad interaction in a BWR. The basic conclusion which resulted from this study is that, while fuel failures induced by pellet clad interaction remain a commercially undesirable event they do not compromise plant safety.

An extensive long-term fuel development program has been successfully completed to address the concern of pellet cladding interaction. This resulted in a barrier fuel design which is not susceptible to cladding perforation due to pellet cladding interaction under any expected operating conditions. Many BWRs have already installed this barrier fuel design. Barrier fuel has been approved by the NRC and essentially eliminates the probability of pellet clad interaction.

For all plants the core operating thermal limits are established considering loss of feedwater heaters as well as other specified

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* General Electric Thermal Analysis Basis...a standardized methodology for calculating thermal hydraulic performance of the core.

7. IMPACT ON FUEL INTEGRITY OF REDUCED MODERATOR TEMPERATURE DUE TO EQUIPMENT FAILURE (Continued)

operational transients. Compliance with these core operating thermal limits specified in the Technical Specifications assures sufficient safety margins.

This issue has been satisfactorily resolved with the NRC.

References

Letter dated November 10, 1976 from G.G. Sherwood (GE) to Victor Stello, Jr. (NRC), Subject: Information Concerning Pellet Clad Interaction.

General Electric Standard Safety Analysis Report (GESSAR), Docket No. STN 50-550, Section 15.1.1.

8. PERFORMANCE OF RELIEF VALVE AUGMENTED BYPASS SYSTEM

Item

The Reed Report noted that the scram insertion requirements for plants designed for relief valve augmented bypass had not yet been achieved.

Discussion and Status

This Relief Valve Augmented Bypass System was deleted from the design of General Electric's boiling water reactors.

This is considered a closed issue.

9. IMPACT OF HYDRODYNAMIC PHENOMENA ON CONTAINMENT DESIGNS

Item

The Reed Report noted that all BWR containment types were undergoing extensive additional analysis required by the NRC to evaluate structural adequacy. The Report also noted that this work would include Mark I, II and III containment designs and might result in modifications to existing Mark I and II containments.

Discussion and Status

At the time of the Reed study, plant operating experience and changing NRC requirements led to concern regarding the structural significance of three hydrodynamic phenomena: relief valve air clearing, high temperature condensation and pool swell. Subsequent to the Reed Report an additional load, condensation-oscillation, was identified.

Since the Report was completed, extensive test programs required by the NRC were completed for all three containment types . Owners groups were formed to determine what actions were required to assure that the design of the containments would meet the NRC requirements. Extensive testing and analyses were performed and the results were reviewed by the NRC. Load definition reports for Mark I, Mark II and Mark III containments were then approved by the Nuclear Regulatory Commission. Containment designs were modified in the light of these load definition reports and certain changes were made in containment structures in accordance with NRC requirements.

This issue has been satisfactorily resolved with the NRC.

References

1. NEDO-21888, "Mark I Containment Program Load Definition Report," Rev. 2, November 1981.

2. NUREG-0661, "Mark I Containment Long Term Program," July 1980.

3. NEDO-21061, "Mark II Containment Dynamic Forcing Functions Information Report," Rev. 4, November 1981.

4. NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria," August 1981.

5. GE Company Document 22A7007, "General Electric Standard Safety Analysis Report," App.3B, February 25, 1982.

6. NUREG-0978, "Mark III LOCA Related Hydrodynamic Load Definition," August 1984

10. RADIATION EXPOSURE FROM REMOVAL OF STEAM DRYER/SEPARATOR ASSEMBLY

Item

The Reed Report noted that the impact on BWR/6 maintainability and availability resulting from potential personnel radiation exposure from handling the dryer/separator assembly needed to be accounted for in the design.

Discussion and Status

Total dose to the plant workers is controlled administratively. The BWR/6 Mark III configuration has been modified to minimize as much as possible radiation exposure to plant workers including features such as underwater transfer and storage of equipment, (i.e., dryers, separators, etc.) These actions reduce exposure rates to onsite workers. Experience in operating BWR/6 plants indicates that personnel exposures are well within regulatory limits.

This issue has been satisfactorily resolved with the NRC.

11. LEVEL OF TESTING OF THE MARK III CONTAINMENT

Item

The Reed Report noted that in the initial Mark III test set up, the simulated drywell and boiler were 1/3 scale while the suppression pool section was full scale. Since these three components did not match each other with regard to size, the application of the test results to the full-scale system required analysis. The then current Mark III test setup was 1/3 scale throughout; however, no dimensional analysis had been performed and in the opinion of the review team, the ability to extrapolate the test results to full scale might be questioned.

Discussion and Status

At the time of the Reed Report, preliminary testing had been performed for Mark III containments at a semi-scale facility. The initial testing that had been performed was to aid in the calculation of containment pressures. The scaling factors were sufficiently well-known that the specified values were conservative.

The load definition report for the Mark III containments has been approved by the Nuclear Regulatory Commission considering results from full-scale and semi-scale confirmatory testing completed prior to the issuance of the first BWR/6 operating license. Where semi-scale testing was performed, scaling studies were also performed. The entire testing program, including the use of scaling factors, was reviewed with the NRC.

This issue [illegible] been satisfactorily resolved with the NRC.

Reference

1. GE Company Document 22A7007, "General Electric Standard Safety Analysis Report," App.3B, February 25, 1982.

2. NUREG-0978, "Mark III LOCA Related Hydrodynamic Load Definition," August 1984.

12. PRESENCE OF DETECTABLE PLUTONIUM INSIDE BWR TURBINE

Item

The Reed Report noted that plutonium has been detected on the inside of BWR turbines. The concern was that migration and concentration of plutonium in the balance of plant was not understood nor systematically evaluated.

Discussion and Status

Experience in BWR operation indicates that essentially all of the plutonium formed in the reactor fuel stays there. Analyses of reactor water show that plutonium concentration is negligible. In fact, the concentration in the reactor water is so low as to be less than 1% of the permissible drinking water concentration limit set by federal regulations. These trace quantities are removed by the reactor water purification system. The resins from this system are removed periodically and disposed of at a licensed waste disposal facility.

Analysis has demonstrated that the water to-steam decontamination factor in a BWR reactor is approximately 1,000, and thus the potential concentration in condensed steam is 1/100,000 of the permissible concentration in drinking water. Therefore, it is doubtful that such a source could result in a significant plutonium deposition. Based upon this evaluation and experience at operating BWR/6 plants, it has been concluded that there is no safety concern for plutonium in reactor water.

This issue item has never been an NRC concern.

13. IMPACT OF SUPPRESSION POOL SLOSHING ON MARK III CONTAINMENT

Item

The Reed Report noted that testing associated with preliminary Mark III designs had not been completed and that it was necessary to demonstrate that these design changes made adequate provision against a phenomenon known as suppression pool sloshing.

Discussion and Status

Seismic slosh testing was performed as part of the containment testing program and it has been shown that seismically induced waves were insignificant. These tests and evaluations were reviewed by the NRC.

This issue has been satisfactorily resolved with the NRC.

Reference

NEDE-21069-P, "Mark III Containment Seismic Slosh," November 1975.

14. EVALUATION OF FUEL TRANSFER ACCIDENT IN MARK III CONTAINMENT

Item

The Reed Report noted that an evaluation addressing the potential for a fuel transfer accident in the Mark III containment had not been accomplished as of that time.

Discussion and Status

This type of potential accident analysis is a standard design practice and was planned prior to the time of the Reed Report.

The Mark III containment contained a new fuel transfer concept. The design of this system was in a preliminary stage and a complete evaluation had not yet been completed. The evaluation was subsequently completed, the concept tested, and the system is now operational at BWR/6 plants.

This concern has been satisfactorily resolved with the NRC7.

Reference

1) NUREG-0124 (Supplement 2 to NUREG-75/100, "Safety Evaluation Report related to the preliminary design of the GESSAR-238 Nuclear Island Standard Design," Docket No. STN 50-447, USNRC, January 1977, Section 15.3, page 15.1.

2) NUREG-0979, "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design," Docket No. 50-447 dated April 1, 1983.

15. IMPACT OF CORE DESIGN AND LICENSING CRITERIA ON BWR CAPACITY

Item

The Reed Report contained a table which identified a number of major problems affecting the BWRs capacity factor.

Discussion and Status

The Reed Report noted a number of major areas which had a significant impact on BWR availability. The items listed were fuel densification, new emergency core cooling system criteria, end-of-cycle scram reactivity, fuel channel wear, feedwater sparger cracking, offgas, recirculation system pipe cracking and preconditioning interim operating management recommendations (PCIOMR) for fuel.

The NRC was aware of each of these items at the time of the Reed Report. The impact on capacity factor represented an operational concern and not a safety concern because appropriate operating limitations ensured that all safety limits were observed. Specific responses to these items are discussed in other parts of this document, e.g., see Item 4 on scram reactivity.

There are no unresolved issues with the NRC in these areas.

As noted in the introduction, the BWR/6 has an availability higher than prior BWRs, and none of the items of concern to the Committee have impacted on its availability or capacity factor.

References

1. NEDE-21282-)-A, "General Electric Densification Program Status," March 1978.

2. NED-20566-P, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with Appendix K."

3. NEDE-21354-P, "BWR Fuel Channel Mechanical Design and Deflections," September 1976.

4. NED-21811, "BWR Feedwater Nozzle/Sparger, Final Report," March 1978.

5. NEDC-20994, "Peach Bottom Atomic Power Station Units 2 and 3, Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations," September 1975.

6. NED-21156, "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration," January 1976.

7. NEDO-21000-1 and 2 (Volumes 1 and 2), "Investigation of Cause of Cracking in Austenitic Stainless Steel Piping," July 1975.

16. ADEQUACY OF DESIGN PROCEDURES TO ASSURE COMPLIANCE WITH LICENSING CRITERIA.

Item

The Reed Report noted a concern that then current practices lacked a systematic basis for assuring that the BWR/6 design conformed to applicable licensing requirements.

Discussion and Status

Product safety standards, regulatory guide implementation positions, and regulation implementation specifications were established to assure that the applicable licensing requirements were documented and easily accessible for use by the engineers and designers of BWR systems and components. Verification of design compliance to determine conformance of the BWR systems and components with the applicable requirements was completed in the finalization of each plant design.

This program enhanced the established basis which was already in place for assuring that the BWR/6 design conformed to the applicable licensing requirements.

Implementation of the new system has resolved this concern. The NRC has granted BWR/6 operating licenses and has audited the GE system from time to time.

17. CONSISTENCY OF DEGREE OF VERIFICATION OF CALCULATIONAL MODELS

Item

The Reed Report noted a concern that there was a lack of a consistent program for verification of calculational models.

Discussion and Status

The General Electric Quality Assurance System has been improved since the time of the Reed Report. It establishes a consistent program for assuring verification of calculational models. Design reviews and independent design verification of inputs are performed as part of the verification program. Several NRC audits of the quality assurance system on calculational models have been performed in the last 10 years. These have confirmed the adequacy of the procedures as applied to calculational models.

The NRC only accepts the use of qualified models for licensing (FSAR) application. The successful licensing of BWR's in the USA validates that qualified models meeting all NRC requirements were developed and are in place.

This issue has been satisfactorily resolved with the NRC.

References

1. NEDE-21254-P, "BWR Fuel Channel Mechanical Design and Deflections," September 1976.

2. NEDE-10801-P-A, "Core Spray and Bottom Flooding Effectiveness in the BWR/6", February 1977.

3. NEDE-10958-PA, "General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," January 1977.

4. NEDE-20231-PA, "Emergency Core Cooling Tests of an Internally Pressurized Zircaloy-Clad, 8x8 Simulated BWR Fuel Bundle," April 1976.

18. POSSIBILITY OF CONTROL ROD BINDING DUE TO FUEL CHANNEL CREEP

Item

The Reed Report noted a concern that the binding of control rods was a possibility resulting from a phenomenon known as channel creep.

Discussion and Status

Extensive analyses and testing of the fuel assembly channels have been performed to demonstrate that the normal operation of the control rods will be affected before there is a significant degradation in scram time. This would permit timely correction of any problem. The results of these tests and analyses were reported to the Nuclear Regulatory Commission and are under review. In addition, individual plant technical specifications require periodic scram time testing of all control drives. These tests would identify any possible channel creep that could interfere with proper drive operation.

In addition a Service Information Letter (SIL) number 320 dated December 1979 was issued to all BWR owners advising them on channel shuffling rules to minimize irradiation exposure and thus minimize cladding creep.

The NRC has not imposed any new requirements on operating plants as a result of this item. Presently GE is considering this item closed although no NRC conclusion has been published.

Reference

NEDE-21354-P, "BWR Fuel Channel Mechanical Design and Deflections," September 1976, Amendment No. 1, February 1977; Amendment No. 2, July 1977.

19. COMPLIANCE OF DESIGN WORK AND REVIEWS WITH WRITTEN PROCEDURES

Item

The Reed Report noted that some design reviews and audits were not in conformance with procedures.

Discussion and Status

This item stems from an internal General Electric audit. Such audits are a part of the General Electric Quality Assurance System. It is expected that these QA audits will find occasional inadequacies in procedures and non-compliance with procedural requirements which will necessitate corrective action. For the audit in question no safety concerns were identified as a deficiency.

The internal audit program is an ongoing effort which continues to find occasional deviations and need for quality system definition improvements. In a few instances potentially reportable deficiencies have been identified through such audits but in general this is not the case. General Electric internal audit reports are available to and have been inspected by the Nuclear Regulatory Commission and General Electric customers from time to time.

This is considered satisfactorily resolved with the NRC.

20. ABSENCE OF AVAILABILITY GOALS IN DESIGN PROCEDURES

Item

The Reed Report noted that instances of nonconformance with General Electric procedures have occurred, some involving issues that are basic to the achievement of design integrity and which affect plant availability.

Discussion and Status

The Reed Report was particularly concerned about the adequacy of BWR availability goals in design. A concern was identified that the present allocation of engineering resources does not provide for an optimal balance in engineering design goals between availability and safety. It was particularly noted that availability goals did not, in may instances, exist in current procedures.

Several programs were developed to collect field data for analysis and feedback into the BWR/6 design process. Availability goals were identified and assigned to key systems design personnel. Performance against goals was recorded and fed back to the responsible design management. Availability trend reports and forecasts were periodically prepared for systems engineers and design management. The COMPASS field data collection system was strengthened in terms of applied personnel and data-reporting requirements. Engineering procedures for reliability were reviewed and revised. Follow-up audits for compliance to procedures were conducted. Corrective actions were written for any non-compliance and corrective actions were completed.

This item was never an NRC concern.

21. SEISMIC CAPABILITIES OF 8X8 FUEL SPACER

Item

The Reed Report noted a concern that the 8x8 fuel bundle spacer design might not have adequate seismic capability to meet future anticipated NRC seismic requirements.

Discussion and Status

Seismic testing of the fuel assembly spacer was completed and reported to the Nuclear Regulatory Commission. The 8x8 spacer was demonstrated to meet all safety and licensing requirements. Therefore, no items of safety significance were identified with respect to this item.

This item is considered satisfactorily resolved with the NRC.

Reference

1. NEDE-21175-P, "BWR/6 Fuel Assembly, Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," November 1976.

2. NEDE 21175-3 P-A, "BWR Fuel Assembly Evaluation of Combined Safe Shutdown & (LOCA) Loadings "Amendment #3.

22. EXTENT OF LIFE OF POSITION SENSOR IN TRAVERSING IN-CORE PROBE SYSTEM

Item

The Reed Report noted a concern that there were readout problems in the design of the traversing in-core probe (TIP) system.

Discussion and Status

The Reed Report identified higher than actual readouts of radial power asymmetry and potential TIP system life-limiting phenomena resulting from operational demands on the system.

The traversing in-core probe system is an operational system with no safety system inputs. Failure of this system could lead to a loss of plant capacity but does not represent a safety concern. System reliability enhancements like a new tip drive mechanism, a new indexer, new purge are control and a new NUMAC drive control unit are available. The new Gamma TIP probes resolve the asymmetry readout problems while also demonstrating longer life and improved accuracy.

This system and its enhancement are not part of any open NRC issues.

23. RADIATION LEVELS OUTSIDE BIOLOGICAL SHIELD AND DRYWELL

Item

The Reed Report noted a concern that unexpectedly high radiation levels outside the biological shield and outside the drywell were observed in some operating BWR plants. It was felt that these radiation levels were unacceptable because they would create difficulties in maintaining and servicing the affected areas of the plant.

Discussion and Status

These observations were made during the startup testing of certain early BWR plants. The affected plants were modified to resolve these shielding problems. Radiation levels at operating plants have not exceeded NRC limits for normal operation. Plant procedures assure that radiation exposure levels for workers do not exceed NRC limits.

This item is considered satisfactorily resolved with the NRC.

24. STRESS CORROSION CRACKING IN DRESDEN 1 CONTROL RODS

Item

The Reed Report noted that evidence from the Dresden 1 plant boron carbide rods indicated that external stress corrosion cracking might be an additional contributor to limited control rod life times.

Discussion and Status

Cracking of the boron absorber tubes in the control rod blades at Dresden 1 had been observed. Examination of the control blade absorber rods was performed and a determination made that there was no loss of boron contained in the absorber rods even though a single through-wall crack was identified in one of the rods that received detailed metallurgical examination.

These results prompted an analysis of the potential safety implications of absorber rod cracking and loss of reactivity worth. Plant technical specification and verification procedures require a demonstration that the reactor can be maintained sub-critical in the cold condition with the highest worth control rod completely out of the core, and, by a specified reactivity margin. Such a demonstration, in a condition shown to be one of the most sensitive to a loss of worth, must be performed prior to each reactor cycle. Only the loss of boron carbide from a significant number of tubes would provide a major reduction in reactivity worth. Since helium pressure buildup from boron depletion is the major source of stress in these tubes, the large difference in boron burnup in the tubes across the wing of the control blade (tip tubes in a new blade will absorb about 70% more neutrons per unit time than the blade average; twice as many as the center-most tube), and the variation in boron burnup among the blades in different core locations, makes a sudden massive loss of boron most unlikely. The study concluded that the times over which the cracking of a number of absorber rods might occur would be long in comparison to the duration of the reactor cycle. Thus any loss of reactivity worth would be revealed by a cold shutdown margin test before any such loss could jeopardize safe shutdown of the reactor.

At the time, the Region 3 Inspection and Enforcement Office of the Nuclear Regulatory Commission was informed of the occurrence of these failures. Copies of General Electric reports were made available for Nuclear Regulatory Commission inspection.

Since the data from the Dresden plants became available, additional control rod inspections at four BWRs has indicated that cracking of control rod absorber tubes does occur and could result in some loss of boron carbide with a consequent reduction of control rod life. GE has thus revised downward the expected life of a control blade. All BWR plants have been informed about the revised control rod lifetimes and

24. STRESS CORROSION CRACKING IN DRESDEN 1 CONTROL RODS (Continued)

GE has developed new Duralife control blades which incorporate new tube materials and hafnium to eliminate cracking while extending blade life. These new control blades are going in future plant reloads.

There is no open safety issue with the NRC on this item.

25. PEAK PRESSURES IN ATWS CALCULATIONS FOR BWR/3 PLANTS

Item

The Reed Report noted that, under certain assumed adverse conditions, pressures in the 1600-1650 psig range had been calculated for certain BWR/3 plant transient events.

Discussion and Status

The Reed Report commented on calculations for BWR/3 plants which show possible peak pressures in the range of from 1600-1650 psig for anticipated transients without scram (ATWS) events. The report further noted that since these calculated anticipated transients without scram overpressures would occur at elevated temperatures when the reactor pressure vessel materials are in a ductile state, they would be expected to have no serious damage consequences on the vessel. However, it was further observed that analyses should be performed to verify this expectation. A recent analysis for a BWR/3 plant, using state-of-the-art analytical tools, calculates a peak pressure less than the values identified above.

General Electric has developed design modification alternatives intended to comply with new NRC requirements issued since the Reed study was prepared. NRC has given generic approval of these alternatives under the ATWS Rule. These alternatives have been selected by the plant owner as appropriate for each BWR type and class. Each plant will receive NRC approval of the design selected to comply with the ATWS Rule requirements.

Solutions have been identified and are being implemented. This issue is satisfactorily resolved with the NRC.

References

1. NEDO-20845, "Anticipated Transients Without Scram: Study for the Pilgrim Nuclear Power Station," March 1975.

2. NEDO-20846, "Anticipated Transients Without Scram: Study for the Monticello Generating Plant," March 1975.

3. NEDO-20847, "Anticipated Transients Without Scram: Study for the Nine Mile Point Nuclear Power Station, Unit 1," Revision 1, May 1975.

4. NEDO-20847, "Anticipated Transients Without Scram: Study for the Oyster Creek Nuclear Power Station," March 1975.

5. NEDO-20847, "Anticipated Transients Without Scram: Study for the Dresden 2, 3, and Quad Cities 1, 2 Nuclear Power Stations," Revision 2, September 1975.

25. <u>PEAK PRESSURES IN ATWS CALCULATIONS FOR BWR/3 PLANTS</u> (Continued)

6. NEDO-20847, "Anticipated Transients Without Scram: Study for the Millstone Nuclear Power Station, Unit 1, Docket Number 50-245," March 1975.

7. Letter, Gus Laines (NRC) to T.A. Pickens, Chairman, BWR Owners' Group on "Acceptance for Referencing of Licensing Topical Report NEDE-31096-P, Anticipated Transients Without Scram; Response to NRC ATWS Rule, 10CFR50.62", dated October 21, 1986.

"1975
NUCLEAR
REACTOR STUDY"

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NUCLEAR REACTOR STUDY

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NUCLEAR REACTOR STUDY

EXECUTIVE SUMMARY

INTRODUCTION

OBJECTIVE OF STUDY

The Nuclear Reactor Study was a highly technical study with the objective of determining the basic requirements for implementing the Nuclear Energy Division's (NED) quality strategy through continuing improvements in the availability and capability of Boiling Water Reactor Nuclear Plants (BWR's). This strategy is predicated on the view that leadership of the BWR in these characteristics represents the greatest opportunity for reducing the Utility customer's power generation cost, with resulting lower power costs for industry and for the ultimate consuming public. The Study included review of the broad range of opportunities for development of BWR leadership in all aspects of availability and capability across the entire range of design, development, manufacturing, construction and operation.

It should be recognized that the NED scope of supply is generally limited to the main components of the nuclear steam supply systems and the fuel. The performance of BWR plants therefore involves not only the GE scope of supply, but also relationships with the Architect/Engineering profession, component suppliers, contractors and the Utility plant operators. The ability of NED to improve total plant availability and capability is strongly influenced by its relationships with these other participants.

This page is part of the
1975 "Nuclear Reactor Study".
See the 1987 "Update Report"
for current status.

DEFINITIONS

Quality, as used in this report, refers to the capability of the entire BWR plant to produce electrical energy (kilowatt hours) as needed by the Utility customer, including response to load demands without unplanned shutdowns, and without power derates. The term Availability describes the fraction of time within a given period that the plant is available for operation, though not necessarily at its full capacity. The plant Capability Factor is the fractional amount of energy the plant is capable of producing over a period of time as compared to the energy that would be produced at full capacity and availability. If there are no derates or other operating restrictions, the Capability Factor would be numerically equal to Availability. Capacity Factor is the fractional amount of energy the plant is actually called upon to produce over a given time period. Since nuclear plants are used for base load, they are generally operated at capacity factors close to their capability.

SAFETY ASPECTS

The Nuclear Reactor Study Group concentrated on reviewing opportunities for improvement in the availability and capability factors of the BWR plants. Although in the course of the Study Group's review, nuclear safety aspects were considered, this Study was not a safety review. However, the Study Group found no reason to believe that applicable safety requirements are not being met for operating BWR plants or will not be met for future BWR plants. Thus, findings and recommendations contained in the Study Group's report about the need for further work, evaluations or increasing margins pertain to improving availability and capability factors and not to any safety requirement. It must be recognized that one cannot rule out future events, such as inadvertent spills, small leaks in piping, etc., which might lead to loss of overall plant availability or capability. While such events are not considered unsafe to the general public, the tendency by some people to relate them to safety may continue due to the high visibility of nuclear power.

SCOPE OF STUDY

The Study concentrated on the Nuclear Reactor including Controls, Instrumentation, Protective System, Steam Bypass and Regulating Systems, Auxiliary Systems and Controls, Containment, as well as selected Tools and Equipment, and the Uranium Dioxide (UO_2) fuel activities. Other pertinent aspects of the Study included the important related activities which can affect plant availability and capability, including:

1. Extended scope offerings such as STRIDE (Standard Reactor Island Design).
2. Design, manufacture and operation of GE-furnished equipment, and
3. Methods of doing business including design processes, procurement practices, organizational responsibilities and interfaces, and the extent of offerings and standardization.

THE REPORT AND ITS STRUCTURE

Section I of the report describes the establishment and membership of the Nuclear Reactor Study Group and summarizes the meetings which were held over the seven-month period of the Study. The Study included many in-depth investigations, and different Group members studied specifically assigned topics and issues in detail. All findings and recommendations were evaluated by the entire Study Group. These findings and recommendations are included in Section II. Major findings and recommendations are presented briefly in this Executive Summary.

Appendix A provides a description of the BWR designs and their evolution, as background information. Biographies of Study Group participants make up Appendix B.

NUCLEAR ENERGY DIVISION PROGRAMS AND ACKNOWLEDGEMENT

The Nuclear Reactor Study Group is indebted to NED personnel for their conscientious and professional response to the many inquiries, as well as for the background briefing sessions, all of which required considerable extra effort. The findings resulted, in most cases, from interviews with, and presentations by NED personnel.

In many areas, problems noted here were recognized by NED, and in many cases corrective programs of varying degrees of intensity are underway (e.g., fast scram development, Mark III testing, etc.). Recommendations were solicited from NED personnel, and many constructive ones were made, including some that were adopted in the final recommendations. The Nuclear Reactor Study Group acknowledges and appreciates this support. All interpretations and the final recommendations are those of the Nuclear Reactor Study Group.

MAJOR FINDINGS

The introduction of light water nuclear reactors for the peaceful purpose of power generation initially was confronted by challenging cost targets. Well-established fossil fuel power plants had reached a high stage of maturity and fossil fuel costs had decreased to their lowest historic levels. This encouraged nuclear plant designers to achieve the lowest plant and fuel costs.

This early environment for the development of nuclear energy has been greatly changed by the oil embargo and the rise in imported oil prices. The nuclear fuel cycle now has a significant cost advantage over the fossil fuels -- more than enough to offset higher nuclear plant costs. The margins are now sufficient to encourage further capital investment in nuclear plants for the worth of future energy production savings. This opens up the potential for many development, design and proof activities that can increase plant availability and, particularly, plant capability and capacity factors. It was in recognition of this opportunity that NED adopted its objective of increased plant availability and capability. The new objective, resulting from the new environment, introduces many new derivative opportunities and goals for plant and fuel which were reviewed by the Nuclear Reactor Study Group.

The importance of nuclear plant availability and capability is illustrated by the magnitude of their equivalence in terms of fuel cost or plant capital cost. It is estimated that a one percentage point improvement in availability is worth about 3¢/million BTU in fuel cost, or approximately $18,000,000 in capital cost for a 1000 MW plant (1980 dollars). Hence, it is important to improve availability by improvement in component durability and reliability and by reduction in the time required for preventative or corrective maintenance. It is also important to minimize operating restrictions so as to maximize plant capability.

FINDINGS, CONT.

After careful investigation, it is the considered opinion of the Nuclear Reactor Study Group that a great deal of excellent technical work has been accomplished by the Nuclear Energy Division (NED), and it is recognized that the existing BWR/6 design has received more thorough technical examination than any of its predecessors (BWR/1 through 5). Many actions are underway to further improve the design and the design process for both the reactor and the containment.

However, the technical demands of this new technology, its rapid growth and long cycles, and the difficulty of obtaining test data, place a demand on this business for which there are no equals in the General Electric commercial business history. It is concluded that the existing BWR/6-Mark III design will fall short of achieving availability and capability such as those characteristic of the General Electric Large Steam Turbine Generator (LSTG). LSTG products have evolved slowly over many years and have had the benefit of a great deal of field testing with a steady evolution in design leading to their present high level of availability and capability, which is viewed as establishing a standard of performance regarded as an ideal to be approached by all large scale power generation equipment.

More specific findings are as follows:

I. Operating Plants

a. A great deal of engineering and management effort is necessarily devoted to solving operating problems in existing plants, and to the attendant customer and government agency relations. Problems have involved equipment, piping, materials, fuel, deratings, operating procedures and the length of refueling and maintenance shutdowns.

FINDINGS, CONT.

b. Experience with operating nuclear plants indicates that the operating availability of Boiling Water Reactor nuclear plants does not differ significantly from that of Pressurized Water Reactor plants. Both have been at an availability plateau of about 70% for several years with no significant change. Large fossil fired steam plants have also been experiencing an availability of about 70%. Hence, BWR nuclear plants are now operating at competitive levels of availability. It is anticipated, however, that the nuclear plants and large fossil plants will improve in availability with maturity. It is important that the BWR meet or exceed this improvement trend.

c. The capacity factor of operating nuclear plants has been about 55%. While the utilities would like to run the nuclear plants at full capacity when they are available, this has not been possible. Plants have been operated at reduced power because of regulatory requirements or owner option during some parts of the operating cycle. Capability and capacity factors have also been reduced by the need to limit the rate of change of power in order to optimize fuel element service life.

d. NED has been pursuing an active approach to providing product service for operating plants. Continued efforts for further improvement of product service and management methods for reducing the length of the refueling/maintenance outage could have a significant near-term effect on improving plant availability. Experience in this activity would also be applicable to improving BWR/6 design, operation, availability and capability.

FINDINGS, CONT.

e. Even with improved product service, continued improvements in fuel and in equipment durability and reliability will be required if higher levels of availability and capability are to be achieved in operating plants and in future BWR/6 plants.

2. BWR/6 Plants

a. The existing BWR/6 Mark III design is incomplete and several important technical problems are still unresolved.

b. The BWR/6 Mark III designs have been arrived at by a process of evolution and reaction to competitive offerings and regulatory requirements. Designs were pushed forward with large jumps in rating in a complex, rapidly evolving technology. The result is a lack of integration in the design. Design integration requires either a very slow, long evolutionary process, for which there has not been time, or the conceiving of the product as a unit, which will itself be an iterative and somewhat lengthy process.

c. Some fuel clad leakage can be expected during the life of the fuel, which may require occasional power derating.

d. One must expect continuing fuel problems, power derates during part of the operating cycle, less than expected fuel burnup, operating limitations, earlier than anticipated replacement of some components, licensing problems, etc. No sudden infusion of resources will correct this situation quickly. An approach to the ideal of LSTG availability-capability standards will take time.

FINDINGS, CONT.

e. Measures for partially or completely correcting some of the anticipated plant availability and capability problems in BWR/6 are definable now, and can be applied to at least some of the units already sold. Indeed, some of the problems must be solved for all BWR/6-Mark III units, even the earliest. Solution or mitigation of some of the remaining plant availability and capability-related problems requires development of additional technology.

3. Standard BWR/6

a. A clear-cut program needs to be established with responsibility for achieving a STANDARD or REFERENCE BWR/6 design that will provide increased performance margins.

b. Historically, a high degree of standardization has not been achieved in the utility industry; however, the need for standardization is greater than ever for nuclear plants and the industry environment is more favorable. Such standardization has not as yet been achieved on the 42 BWR/6 units on order, nor on any of the previous plants. Since the design is not yet complete, standardization can proceed by incorporating improvements in successive groups or blocks of plants. If utilities further delay construction schedules, the latter members of the series need not be current models, but might be an improved design to be developed in the next 4-5 years.

FINDINGS, CONT.

4. System Design

   a. Some of the engineering design margins are considered insufficient to ensure full power operation reliably throughout plant life. Reactor power will probably be limited to less than nameplate rating (to as low as 85%) at certain parts of the core operating cycle.

   b. Several calculational models are not as yet based on sufficient experimental data, perhaps resulting in a potential 5% to 10% loss in rated output. (Not additive to the above.) A comprehensive program for improved verification of calculational models needs to be implemented.

   c. A 90% availability goal is not realistic, and a goal expressed in such terms is not meaningful, since it does not express the need of the customer to have a plant which can be run at full capacity whenever desired. An availability goal of 85% and a capability factor goal of 75% would be more realistic at the present time. A Division-wide, high-visibility, availability/capability improvement program will be required to achieve these goals.

   d. A separate Systems Engineering Function and procedures for overall BWR Systems Design Reviews are needed.

5. Mechanical & Electrical Systems

   a. In spite of large amounts already expended on engineering development, additional investment in engineering development and test would be required to ensure a product approaching LSTG quality.

FINDINGS, CONT.

b. Additional facilities are needed to assure full scale life testing and design qualification prior to installation and operation of certain important plant components.

c. The recirculation flow control valve design is still incomplete, and this valve is not as yet proven by sufficient testing. Commitment to installation in operating plants must be made while life test information is being obtained.

6. Containment

a. The Mark III containment has several, unresolved design problems. Some structural problems may be present in the existing Mark I and II designs.

b. The need for large numbers of safety/relief valves may decrease availability and capability due to some potential radiation problems inside the containment where personnel must have access. REliief valve actuation would require rapid evacuation of this area and may limit access for some time after relief valve actuation. There is also the possibility for inadvertent valve actuation or failure to reclose, which can adversely affect availability and capability.

7. Materials & Processes

a. The NED BWR materials and chemistry effort needs to be increased. Important added tasks for a substantial number of additional professional people have been identified.

FINDINGS, CONT.

b. Reactor internals are exposed to high energy neutron fluxes; some material, such as the lower grid, is located in areas where loss of ductility later in life may require replacement, which would be very difficult in the current design, due to the radiation levels involved.

c. Selective replacement of 304 stainless steel, which is used extensively, may be required due to the risk of stress corrosion cracking.

d. The buildup of radiation levels as plants get older could impede maintenance and service, and eventually require time-consuming and costly decontamination. This occurs in Pressurized Water Reactors (PWR's) as well. Government-sponsored R & D programs can be expected to be addressed to this type of problem in the future.

8. Management

a. Responsibility for carrying through hardware design, development, production and test is diffuse. Personal responsibility should be assigned for key components and systems.

b. The existing organizational structure needs modification to increase the influence of material expertise on management decisions.

c. Design engineers need more exposure to current field construction and operating plant problems.

d. Engineering support of construction and procurement of plants and components needs to be increased.

RECOMMENDATIONS

The losses in capability and availability in existing plants are due to a variety of causes which have been identified. Since the BWR/6 Mark III is considerably different from previous designs, it is to be expected that several additional factors will tend to adversely impact its availability and capability, and the Study Group has attempted to identify these. The recommenda.ions that follow are expected to be helpful in increasing the availability and capability of both existing and new plants.

SHORT TERM RECOMMENDATIONS

1. Operating Plants

   Place the highest Division priority and level of management attention on improving the availability and capability of plants now in operation or near operation. Provide for spin-off from the operational plant improvement activity into the standard plant design. Conversely, provide for backfit of standard plant improvements into operating and near operational plants.

2. Refueling & Maintenance Time

   Develop more rapid refueling and maintenance procedures. Study and document step-by-step procedures and tools necessary to reduce the time required during a refueling and maintenance period. The objective should be to cut the outage time by at least 50%. Establish a task force of experienced manufacturing engineers and field service personnel comprised of NED, Installation and Service Engineering (I&SE) and Manufacturing Engineering Consulting Service (MECS) personnel to perform this task.

RECOMMENDATIONS, CONT.

3 Product Service

NED should establish a BWR nuclear service business operation with the following priority of objectives (note relation to recommendation 4):

a. Establish and maintain an improved system of reporting in detail all component failures and causes of nuclear plant shutdowns and operating incidents leading to loss of plant capability.

b. Provide BWR owners a broad range of services that will enhance their utilization of nuclear plants and improve overall plant availability and capability. This service should be supporting of and function cooperatively with I&SE.

Examples of important services of immediate value to customers would be the development of methods for improvement and simplification of the Preconditioning Interim Operating Management Recommendation (PCIOMR), development of improvement in refueling and maintenance scheduling, and development of radioactive waste handling and management procedures, as well as development of plant decontamination procedures.

4. Field Information Feedback

In connection with recommendation 3, strengthen the overall NED network for collecting, analyzing and disseminating data and experience from operating plants for feedback into the evolving design process in the manner of the elaborate LSTG system.

5. Systems Engineering Function

Establish and implement a Systems Engineering Function to delineate System requirements. Establish an independent and rigorous BWR Systems Design

RECOMMENDATIONS, CONT.

Review discipline that provides a comprehensive and independent evaluation of each generic BWR design at critical points in the program

6. Availability/Capability Planning & Control

Set realistic availability and capability improvement goals and establish a Division-wide program to achieve them on a measurable basis and expedited schedule, by assignment of specific goals to hardware components and responsible individuals and organizations. The goals should contain assigned numerical reliability improvement targets for each element of total Nuclear Steam Supply System (NSSS) hardware. At each level in each organization component, establish the methods to achieve the assigned goals, the schedule of required accomplishment, and the techniques to measure and report the degree of achievement.

7. Cooperative R & D

Promote and establish cooperative R & D programs with other organizations conducting R & D work relevant to BWR technology, including licensees, utility customers, U.S. and foreign governments, etc.

8. Six-Month Refuel Interval

Immediately study the feasibility and benefits of going to six-month refueling cycles on all BWR models with the objective of:

a. Reducing fuel cycle costs
b. Permitting more frequent scheduled maintenance and reducing forced outages
c. Increasing core operating margins
d. Improving control margins.

This recommendation could be applied to future BWR/6 commitments, to BWR/6's in the pipeline, and backfitted to existing plants.

RECOMMENDATIONS, CONT.

9. Design Control-Purchased Equipment

Take steps to acquire and develop more engineering competence and design expertise pertaining to outside purchased hardware. Implement a procurement policy calling for the development of vendor relations that provide for NED engineering review and approval of design details and materials of critical purchased components, even though this may increase the cost of these items.

10. Materials & Processes

a. Establish a unified Materials and Processes component with responsibility for:

(1) Creating a Materials and Processes Laboratory organization at San Jose with responsibility for experimental work and testing related to Materials with priority emphasis on Stress Corrosion Cracking, Water Chemistry, Plant Radioactive Contamination and Decontamination;

(2) Strengthening the Materials and Processes effort in manufacturing at Wilmington; and

(3) Imposing an added discipline on Materials and Processes control in design, procurement, construction and field operation.

b. Establish a program to minimize radioactive contamination within plant systems and to simplify decontamination.

c. Increase the level of effort in materials engineering development and design.

RECOMMENDATIONS, CONT.

11. Product Responsibility

Assign clear-cut product responsibility and accountability to specific individuals for components and systems in place of the present diffuse responsibility and accountability assignments. This responsibility should extend from design through development, test, production, construction and field operation, including vendor as well as in-house designs.

12. Training of Design Engineers

Establish a rotational training program to increase the "real world know-how" of hardware and design engineers.

INTERMEDIATE TERM RECOMMENDATIONS

13. Standard BWR/6

Adopt and install new objectives, procedures and management systems to achieve an optimum degree of standardization in phases by introducing design changes in successive blocks of plants in backlog and future orders. Develop a Standard or Reference BWR/6 design in parallel with ongoing design activities. The design objective should be to reduce the number of different plant designs, to improve plant availability and capability and to provide a basis for introducing block design changes. Assign full responsibility for design of the standard plant to a qualified leader with a dedicated design team.

14. Upgraded BWR/6

Exhaustively study the feasibility and cost benefit aspects of upgrading the present product line by using the next larger vessel size, more fuel

RECOMMENDATIONS, CONT.

and higher flow capability for each power rating (reduction in core power density by about 20%); this study should include, where practical, present backlog orders.

15. Backlog Plants

a. Examine opportunities and cost benefits for renegotiating backlog orders in order to simplify commitments or to incorporate standard plant features on a block basis.

b. Delay as long as possible the commitment of developmental hardware to backlog plants. Prepare contingency plans for substitution of backup hardware.

16. Future Orders and Scope

Initiate a product offering and scope study for future offerings to define the optimum standardization scope of supply and performance.

17. Verification of Calculational Models

Obtain additional experimental data to check calculational models.

a. Arrange for more instrumentation during startup testing and selected phases of subsequent operations.

b. Investigate cost/benefits of additional component testing. Encourage Government agencies to expand related experimental studies.

c. Investigate possible use of a Utility-owned reactor for experimental verification of key nuclear calculational models and fuel proof testing.

RECOMMENDATIONS, CONT.

d. Review consistency of predictions of new models vs. models previously in use.

e. Include evaluation of all models for accuracy and applicability in Design Reviews.

18. Fuel

a. Fuel programs should be implemented and accelerated to reduce future failure rates. These programs should include:

(1) Continuation of work on fuel failure mechanisms.

(2) A program to design and test a reduced fuel duty design such as a 9 x 9 array, provided that analyses confirm overall benefits over 8 x 8, at least for present plant commitments.

(3) Continuation of work on "barrier approach".

(4) Development of a fine motion control rod design as a backup to the barrier and reduced fuel duty programs.

(5) A program to improve the effectiveness of the PCIOMR procedure.

b. Make provisions for relicensing the General Electric Test Reactor (GETR).

c. Arrange for Remote Materials Laboratory capability to accommodate full length fuel rods and cask handling.

19. Design Qualification and Test

a. Embrace the discipline of consistently providing for the resources and time cycle needed to accomplish adequate design qualification and endurance testing before making a decision to include in a commercial offering a new unknown hardware design which is important to plant reliability.

RECOMMENDATIONS, CONT.

b. Establish, or arrange for the use of, additional experimental facilities for:

(1) Full scale flow induced vibration testing of major components under realistic conditions.

(2) Hydraulic testing of all components such as safety/relief valves, flow control valves, jet pumps, etc.

(3) Containment/Pressure Suppression System Testing.

(4) Control rod system performance and life testing (including fast scram and slow motion control rods).

20. Containment

Make a reassessment of the Mark III containment concept for future offerings.

21. External Factors

Increase attention to identifying, recognizing and preparing for the technical impact of external factors (including social, political, economic and regulatory).

LONG TERM RECOMMENDATIONS

22. BWR Test Bed

Arrange for acquis .ion or use of a BWR test bed in which new kinds of fuel can be tested and in which nuclear and thermal hydraulic measurements can be made in actual operation.

RECOMMENDATIONS, CONT.

23. Advanced BWR Design

Establish a small, separate group of highly competent individuals with instructions to produce a sound preliminary design of a superior BWR standard power plant for offering to the industry in approximately 1980. The results of the design studies of this group may be expected to have short to intermediate use as inputs to decisions regarding changes in existing designs.

DETAILED FINDINGS AND RECOMMENDATIONS

The foregoing summary has included most of the major types of findings and recommendations. Attention is also directed to Section II which sets forth many more detailed findings and recommendations directed to providing technical solutions for principal availability and capability problem areas and concerns.

ESTABLISHMENT OF NUCLEAR REACTOR STUDY

In October, 1974, Mr. R. H. Jones, Chief Executive Officer of General Electric, requested Dr. C. E. Reed, Senior Vice President, Corporate Studies and Programs, to make a highly technical study of the Boiling Water Reactor Nuclear Power System with the objective of determining the requirements for implementation of the Nuclear Energy Division "quality strategy", with particular emphasis on the BWR/6 product line.

Reasons for Study

The first GE BWR over 300 MW was placed in commercial operation in December 1969. By the end of December 1974 there were 20 such plants in operation. As the number of large nuclear power plants in operation has increased, the electric power generation industry has become more aware of the economic value of high availability/capability factors. While the present availability/capability of nuclear power plants isabout the same as that of large fossil plants, it is becoming evident that any increase above present performance will provide sizable benefits for the user. These benefits have increased in the last two years, as the increases in fossil fuel costs, construction costs and costs of capital all tended to amplify this effect. For instance, a 1 percentage point improvement in availability or capability is worth 3.0c/MBTU in fuel cost, or approximately equivalent to $18 M in capital cost for a 1000 MW plant (1980 dollars). This can be compared to about $780 M capital cost for the whole plant, and $75 M for the Nuclear Steam Supply System, also in 1980 dollars. While in the past such considerations have generally not been included in the competitive evaluations on which the utilities based their equipment selection, evidence is now being accumulated that shows that relative availability/capability can and should become an important consideration. GE's experience in the Large Steam Turbine Generator (LSTG) area has shown that leadership in availability/capability will be recognized by the utilities. The adoption of the "quality strategy" by NED was a decision to give major emphasis to the achievement of similar leadership in the Nuclear Reactor field.

Membership of Study

To conduct this study, Dr. Reed established a Task Force including the following individuals, chosen for their particular expertise and background (see Appendix B for biographical data):

Chairman:

C. E. Reed* Senior Vice President, Corporate Studies and Programs

Members:

W. W. Beardslee, Manager-Manufacturing Engineering Consulting
R. H. Beaton, Vice President and General Manager, Energy Systems and Technology Division, Power Generation Business Group
A. M. Bueche* Vice President, Corporate Research and Development
K. P. Cohen* Consulting Scientist, Nuclear Energy Products Division
B. R. Doyle, Manager, General Accounting Operation, Corporate Finance Staff
C. W. Elston, Manager-Steam Turbine-Generator Operational Planning
M. C. Hemsworth, Manager-Consulting Engineering, Aircraft Engine Bus. Group
T. H. Lee* Manager-Group Strategic Planning Operation, Power Generation Business Group
H. E. Stone, Manager-Nuclear Energy Operational Planning, Nuclear Energy Div.

Advisory Consultants:

L. C. Harriott, Manager-Engineering Consulting Service
M. C. Leverett, Manager-Nuclear Safety Assurance, Nuclear Energy Division
J. F. McAllister, Staff Executive-Product Quality
S. Neal, Manager-Applications and Communications, Corporate Research & Development
M. P. O'Brien, Consultant to General Electric Company
E. Schmidt, Consultant to General Electric Company
A. M. Weinberg* Consultant to General Electric Company
J. F. Young* Vice President and Staff Executive, Technical Resources

Staff:

A. W. Robinson, Staff Executive-Corporate Studies and Programs
S. Thornton, Consultant, Corporate Consulting Service

* Member- National Academy of Engineering

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Meetings

The Study was paced by a series of eleven meetings of the whole Study Group while, in between formal meetings, individual members proceeded with separate interviews, discussions with experts, visits, reviews of drafts, and in-depth studies of assigned topics.

The eleven formal task force meetings covered the following summary agendas:

#1 - November 11, 1974 - Fairfield, Connecticut

- Objectives and organization of Study
- Introduction of members
- Initial NED background briefings

#2 - December 4 & 5, 1974 - Fairfield, Connecticut

- NED briefings
- Discussion with R. H.Jones, Chief Executive and W. D. Dance, Executive Officer
- Preliminary identification of topics for depth studies.

#3 - December 18, 19, 20, 1974 - San Jose, California

- Product Quality briefings (corporate, LSTG, Naval Programs, Apollo)
- NED briefings continued
- Initial identification of areas of concern

#4 - January 6 & 7, 1975 - Clearwater, Florida

- Selection and assignment of topics for depth studies
- Final NED briefings

#5 - January 22, 23, 24, 1975 - San Jose, California

- Listing of identified problems/concerns
- Status report on assigned depth studies

- Informal discussion with selected NED managers

- Task Force "brainstorming" session

#6 - February 13 & 14, 1975 - Fairfield, Connecticut

- Briefings: Aircraft Engine Group, Marine System Programs, Marine Steam Turbine Program

- Informal discussion: T. O. Paine, Senior Vice President (formerly Group Executive - Power Generation Group)

- Briefing: Relationships between M&P Labs and supported design function in GE.

- Progress reports on assigned depth studies

#7 -- February 27 & 28, 1975 - Schenectady, New York

- Preliminary findings and recommendations from depth studies

- Informal discussion: A. E. Schubert, President, Allied Gulf Nuclear Services (formerly Vice President and General Manager, Nuclear Energy Division)

- Briefings on GE Installation and Service Engineering

#8 - March 11 & 12, 1975 - Schenectady, New York

- Review of findings and recommendations

- Informal discussion with M. F. Kent, formerly Vice President and General Manager, Power Generation Sales Div.

#9 - April 2, 3, 4, 1975 - Fairfield, Connecticut

- Final review of findings and recommendations

#10 - April 30, May 1 & 2, 1975 - Fairfield, Connecticut

- Review depth studies updated on the basis of Meeting #9 comments

- Review preliminary drafts of study report

- Identify key findings and recommendations for inclusion in the Executive Summary

#11 - May 20 & 21, 1975 - Fairfield, Connecticut

- Final review of drafts of study report to obtain Task Force members' comments and concurrence.

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Altogether, the Task Force held twenty-five full days of meetings, many of them supplemented by evening meetings. Topics singled out for depth studies were the following:

A. Nuclear Systems

B. Fuel

C. Electrical, Control and Instrumentation Systems

D. Mechanical Systems and Equipment

E. Materials, Processes and Chemistry

F. Production, Procurement and Construction

G. Quality Control Systems Overview

H. Management/Information Systems

I. Regulatory Considerations

J. Scope and Standardization

In the following Section II, Study findings and recommendations are grouped under the same headings.

NUCLEAR SYSTEMS

GENERAL FINDINGS AND RECOMMENDATIONS

1. Nuclear System Commitments

Because of the long lead time required to get field experience on reactor performance, the light water reactor industry has been obliged to commit and design improvements more on the basis of theoretical analysis than on experimental data, as is usual engineering practice for short cycle products. Many of the difficulties in current BWR's were caused by the discovery of new phenomena, or the impositions of new regulatory requirements, which were not recognized at the time of commitment.

As fossil fuel costs and all capital costs escalate, past emphasis on nuclear power plant marginal economics will be supplanted by emphasis on plant reliability* availability and capability.

Recommendations

a. A formally structured approach should be taken in design and development for high plant reliability*

b. Future designs and design changes should not be committed until engineering design and testing are much further advanced than has been traditional in the industry.

c. Design margins in excess of required safety margins should be substantial.

d. Provision should be made in design for future regulatory actions.

e. No hasty changes should be made on plants now in the pipe-line.

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* Reliability is defined throughout as the absence of forced outages. Numerically reliability = 1 - forced outage rate.

2. BWR Quality Leadership Strategy

Finding

Existing BWR system designs do not fully support the BWR quality leadership strategy. The BWR/6 design, and BWR/2-5 plants, are close to operating limits in many areas. Probable consequences - aside from fuel effects - are some combination of:

a) a few percentage point losses of availability,

b) plant deratings for equilibrium cores up to 15 during the last quarter of their cycles,

c) occasional forced outages from containment contamination, for repair of relief valves and other plant components, for modification of the suppression pools or the steam lines leading thereto,

d) occasional burnup shortfalls of 1000-2000 MWD/T,

e) licensing difficulties, and

f) an adverse impact on public image.

These will affect the early BWR/6's, and also all the BWR/2-5's, especially those with 7 x 7 fuel.

While the behavior of PWR's has not been fully analyzed, the principal near term prospect for a successful strategy of superior reliability would appear to be the difficulties of the competitors in the steam generator area.

Recommendations

a) The present product line should be upgraded, by using the next larger vessel size and flow capacity for each power rating (reduction of core power density by 20). In conformity with Recommendation 1 above, this recommendation should receive detailed engineering study before commitment. Commitment should not occur before mid-1976. The reduction of core power density will

(1) Reduce fuel duty and thus most of the incentive for 9 x 9 fuel for future plants.

(2) Increase core operating margins.

(3) Increase availability and capability.

(4) Reduce void coefficient and severity of transients.

(5) Increase NSSS and containment costs.

(6) Reduce the overall cost of nuclear power.

b) Develop rapid refueling procedures and schedule 6-month refueling intervals, which will

(1) Reduce fuel cycle costs.

(2) Permit more frequent schedule maintenance and reduce forced outages.

(3) Increase core operating margins.

(4) Improve control margins.

This recommendation should be evaluated on future BWR/6 commitments, for BWR/6's in the pipeline, and for backfit to existing plants.

c) Inaugurate preliminary design and development of a superior quality, long-range standard BWR (BWR/S) for introduction in 1980 or later.

(1) Specifications per Table I (see Specific Recommendation 4).

(2) Temporary preliminary design team, insulated from pressures of producing backlog.

(3) Review of results after two-year design study.

(4) Decision to proceed dependent on results of study and progress of development, and on business circumstances in 1978.

5) Study may be used as a guide to the direction of evolution of BWR/6's.

SPECIFIC FINDINGS

F-1 NRC Impact on BWR Design

NED makes insufficient advance provision in design or design margin to reduce the impact of expected NRC regulations.

F-2 BWR System Design

The BWR system design in many instances is incomplete. The design process could be improved by more complete integration of functional requirements, design bases (methods and models), and design criteria. Pressure to meet schedules has resulted in omission of some trade-off studies and has resulted in a requirement for larger overall design margins.

F-3 BWR Reliability/Availability/Maintainability Design Program

The NED design program to improve Reliability/Availability/Maintainability, outside of the safety area, is largely undefined and unimplemented. Action in this area is not yet consistent with a BWR quality leadership strategy.

The plant availability goals which have been set seem unlikely to be met. These goals have not been adequately evaluated statistically to assure they will provide the BWR a competitive edge in reliability/availability/maintainability.

F-4 NED Manpower Resources

The manpower resources of NED are insufficient to implement the BWR quality leadership strategy in the face of other demands to keep operating plants running, produce the backlog orders and respond to regulatory initiatives.

-5 Computational Facilities

NED computer capability is inadequate to automate the design process, to accommodate longer running, more accurate design programs, and to meet anticipated requests from regulatory bodies for increased numbers and complexity of safety analyses. This lack of computer capability and procedures leads to pressure on design margins as more complex, finely tuned designs are developed.

F-6 Nuclear Design Methods

Reactor core design methods have to cope with an unusually complex coupled nuclear and thermal-hydraulic system. NED has applied considerable effort and ingenuity in meeting this problem, despite being hampered by a computer with a smaller capability than competitors (who have a less complex system to deal with).

Nuclear design methods have evolved rapidly in the recent past to correct discrepancies between predictions and field data (which have reduced original design margins), and to strengthen identified weaknesses in calculational models. Qualification and verification of these improved methods is under way, but is not yet complete. The qualification and verification efforts may uncover additional discrepancies that could further change design margins. The qualification and verification efforts are hampered by the lack of an instrumented test facility representative of a larger nuclear and thermal-hydraulically coupled BWR against which design methods can be verified.

F-7 Thermal-Hydraulic Design Methods

NED has long had major efforts in thermal-hydraulics. The ATLAS loop is the largest heat transfer facility of its kind in the U.S., and GETAB is based on over 6000 data points. Hydraulic design methods are being continuously improved. The present design tools are of variable precision and accuracy, and are verified to different degrees by experimental data. Because of the great complexity of the thermal-hydraulic phenomena, both steady-state and transient, further variance between design predictions and experimental data may occur. Transient test facilities are still limiting in obtaining experimental data.

F-8 Design Methods Qualification

In principle, operating reactors provide a continuous stream of nuclear performance data against which analytical design models could be checked. The actual value of this data for checking specific models is limited because of inadequate scope or precision of plant operating instrumentation for scientific measurements. Nevertheless, such data have been usable and have led to significant improvements in our models.

Nuclear and thermal-hydraulic design methods are generally qualified separately by special tests and computations. Coupled nuclear and thermal-hydraulic data are necessary for the qualification of design methods for high power density BWR's, like BWR/6. However, coupled data are less available, and when available, are less accurate than the uncoupled data. The greatest need for verification is in the BWR-unique area of strong nuclear and thermal-hydraulic coupling.

F-9 BWR/6 Core Design Margins

In their recent redesign of the BWR/6 core, the NED core design team has succeeded in restoring some design margins. This design is calculated to meet all requirements, including fuel cycle cost and plant output, with a nominal 10% margin on LHGR. Nevertheless, BWR/6 core nuclear and thermal-hydraulic design margins are not large enough to cover all of the following factors:

a) Methods uncertainties and new information from continuing qualification of design methods.

b) Future requirements arising from completion of system design.

c) Allowance for variation in operator requirements such as fuel failures, load following, reactor outages, etc.

d) Adaptation of the generic design to particular contractual requirements, manufacturing requirements, and other plant sizes.

One or some combination of the following consequences may occur.

a) Power derating up to 15% during a portion of the cycle because of excessive power peaking required for operating flexibility or in order to continue to meet operating limits at rated power with high void coefficients and design (or "D") scram curve. (Most likely to occur progressively during the latter quarter of the cycle and more probable for reload than for initial cores.)

b) Reactivity shortfall, because of need for and of cycle maneuvering allowance, or possible requirement for more conservative scram curve at end-of-cycle. In equilibrium cores exposure shortfalls of up to 400 MWD/T are possible and some shortfalls of 1000-2000 MWD/T are probable.

c) Regulatory delays because of the need to show ability to meet licensing limits.

It is presently anticipated that BWR/4 and 5 designs not yet released will utilize the BWR/6 core design. Hence, conclusions stated above generally apply also to these plants.

F-10 BWR/2-5 Core Design Margins

BWR/2 through BWR/5 equilibrium cores were not designed or analyzed in detail at the time of initial commitment. Recent reload core design and equilibrium core analyses show some margins insufficient to sustain full reactor power for warranted fuel exposure.

One or a combination of the following consequences may occur for BWR/2 through 5 cores.

a) About one half of the reload cores may be derated from 5 to 20 (higher value for some of the BWR/4 reloads) in order to satisfy the ECCS MAPLHGR limits. Such derating would apply only over the period of time that the reactors contain the old 7 x 7 fuel bundles (the next four years) and would generally not be at maximum derate for more than one quarter of the cycle.

b) Derating up to 20 may be required in the last quarter of the operating cycle for many of the BWR/4 and 5 reload 8 x 8 cores to satisfy transient limits with the high void coefficients and the design (or "D") scram curve, unless plant hardware fixes (retrofits) in combination with hoped for improved computational models can demonstrate adequate accommodation of the unfavorable void coefficients and scram curve.

c) Power derating, burnup deficiencies (predicted for many of the cores), and the regulatory delays discussed under BWR/6 also generally apply to BWR/2-5 reload 8 x 8 cores (considerably more to BWR/4 and 5 than to BWR/2 and 3) and, to a lesser degree, to BWR/4-5 initial 8 x 8 cores.

F-11 Degradation of Core Internals

Uncertainties in current estimates of radiation and corrosion damage to BWR/6 core internals do not provide assurance of 40 years service lifetime. Core internals may have to be replaced earlier to provide assured structural integrity for continued operation. Replacement of permanently installed core internals would result in substantial BWR downtime because of high radiation levels and difficult access.

F-12 Void Coefficient/Relief Valve Problem

The main cause of potential BWR transient severity is the high reactivity worth of the steam voids. This applies to BWR/2, 3, 4, 5 and 6, with severity increasing in BWR/4 and 5 and again in BWR/6. The consequences are aggravated by the low thermal inertia of the steam system. Engineering has responded to this trend of increasing severity by providing protection against the transients through substantial additions of relief valves, trip circuitry, and fast scram blade drives, all of which are of novel design and require high reliability. Transient control involves use of the pressure suppression pool and the containment during expected plant operating conditions, such as turbine trips. A consequence of this is the potential contamination of the Mark III containment by discharge of radioactivity during relief valve operation.

F-13 End-of-Cycle Scram Reactivity

The curves of reactivity vs. control rod position of BWR/2 through 6 are unfavorable for end-of-cycle equilibrium cores (either with UO_2 reloads or MO_2 reloads) as a result of the high reactivity in voids. One possible, but undesirable, way to achieve acceptably rapid reduction of power on turbine trip during the last part of a cycle would be to reduce power and thereby allow some blades to be kept in the core. To avoid such derating, design changes are required; fast scram, prompt relief trip, recirculation pump trip, additional relief valves, and/or reduced relief valve settings. All of these tend to reduce availability and increase licensing difficulties.

F-14 Transient Design Methods

The current methods used to calculate BWR transients, although comprehensive in their treatment of the overall nuclear steam supply system, are approximate in their treatment of the reactor core. They are furthermore essentially untested with respect to their treatment of the central feature of the system, the coupled nuclear-hydraulic process. In view of these uncertainties, the allowed margins are not sufficient to assure acceptable availability and capability. As more accurate methods are developed and applied (and such developments are in progress), hardware requirements may increase and call in turn for an unknown degree of further backfitting or else derating.

The method used to determine stability margins would normally be considered both good and conservative. However some observed phenomena which could relate to reactor stability are not completely accounted for:

a) Sizable resonant fluctuation in the fission rate,

b) Decrease in fuel rod thermal time constant with Pu build-up.

F-15 Extent of Verification of Transient Analysis

The available tests of the transient model to date have been satisfactory. They have not provided a critical test of its least certain features, namely, representation of the void coefficient, and of the delayed scram reactivity curve.

Tests involving more extreme conditions in operating plants and comparisons with mult-dimensional nuclear/thermal/hydraulic transient models are needed to verify better the adequacy of the design model now in use.

F-16 Anticipated Transient Without Scram (ATWS)

Regulatory requirements call for backfit modifications to operating plants. While most of the work can be done during a regular refueling outage, an additional few percent of unavailability may occur during the year of modification.

F-17 Flow Control Range

The operating range of the BWR/6 recirculation flow control systems, nominally 75% to 100% of reactor power on the 100% power control rod pattern line, is less than utility load following requirements, although this range was accepted by the customer. The range originally estimated for BWR/3 was 50%. For equilibrium cores at EOC, the expected stable range is of the order of 40%; for BWR/4-5 the corresponding figures are 35% originally estimated, 25% currently estimated. The restricted flow control range reduces operating flexibility, and requires more frequent control rod movement, which tends to increase fuel failures. Fuel pre-conditioning will take longer than in earlier BWR's which have a wider flow control range. Accommodating xenon transients will also require more control rod motion.

F-18 Flow and Temperature Induced Transients

The loss of feedwater heating or the accidental activation of the High Pressure Core Spray or High Pressure Core Injection Systems could produce power transients long enough to cause increased leakage from fuel. These transients may be more severe in this regard than pressurization transients because they can last long enough for pellet-clad interaction failure to occur.

F-19 Special Problems with REVAB Plants

Scram insertion requirements for plants designed with Relief Valve Augmented Bypass (REVAB) have not yet been achieved. No solution has yet been identified. Delays of six months to modify REVAB systems and components may occur for plants scheduled to start up in 1975-1976.

F-20 Analytical Models and Margins for the BWR ECCS

The lack of thoroughly verified models for BWR ECCS has led to the imposition by the NRC of highly conservative model assumptions and design limits, which have resulted in 5 derates for BWR/3 facilities. Further derates of up to 10 are possible, at least until the 7 x 7 fuel is replaced by 8 x 8 fuel.

F-21 BWR Containment Design Status

Because of phenomena recently discovered, all BWR containment types (Mark I, II and III) are undergoing extensive additional analyses to evaluate structural adequacy. As a result of these analyses, Mark II as well as Mark I are likely to be redesigned and retrofitted.

The Mark III containment will have more severe operating and maintenance problems than Mark I and II because small accidents and normal relief valve operation will discharge steam and radioactivity and cause high noise levels within the containment. This will impact on BWR/6 availability (2% per year). In addition, the phenomenological problems associated with Mark III (such as suppression pool sloshing, steam-air bubble vibration, relief valve blowdown forces, LOCA blowdown forces on the containment, radioactivity contamination carryover throughout the containment) are likely to require substantial additional testing to clarify the applicable functional requirements.

F-22 Seismic Design Position and Capability

The seismic design responsibilities are diffused and dispersed within NED and between NED and A/E. Upgrading of NRC requirements on seismic design, coupled with the current lack of GE control over A/E designs could result in major redesign efforts and possible retrofitting in plants already constructed.

F-23 Radiological Contamination

The uncovered suppression pool of Mark III causes Mark III to be more susceptible than previous designs to the loss of availability due to present occupational dose limits and a fortiori to more stringent regulations which are anticipated. Mark I and Mark II designs may also be affected by increased difficulty in performing maintenance and backfit if required.

Design features such as PRT and REVAB and bottom entry core monitors, as well as pressure relief for abnormal events, lead to increased radioactivity in the containment. Source terms provided by GE for radioactivity release are subject to large uncertainties which may lead to unfavorable outcomes in terms of plant availability and reliability. Controls on sources of induced radioactivity are not adequately specified in design.

F-24 Other Accident Considerations

There is no GE action plan for industrial security or sabotage with regard to the Mark I and II containments, and such action for the Mark III is considered to be the A/E and customer responsibility.

The following events are presently schedule to receive further consideration before a firm position is established for GESSAR: (1) ATWS, (2) Major and minor fires, (3) Liquid radwaste tank failure, and (4) Off-gas (hold-up portion) failure. There are presently firm positions in GESSAR for (1) Fuel assembly insertion error, (2) Recirc pump seizure, (3) Cask drop, (4) Spent fuel, (5) Control rod drop, (6) Pipe breaks - inside and outside containment, (7) FW piping break. Of these events, there are likely to be new requirements imposed on cask drop, control rod drop, pipe breaks - outside containment, and fuel handling.

Some Mark III detailed designs do not appear to have adequate provision against suppression pool sloshing.

SPECIFIC RECOMMENDATIONS

R-1 Upgrade Future BWR/6's

Increased design margin in new proposal offerings, and those in recent sales that can be renegotiated, can be attained in the near term (that is, mid-1976) without extensive BOP redesign by upgrading current NSSS designs by one pressure vessel size. This would result in the following three standard offerings.

Bundles	Approx. RPV dia. (in)	Thermal Rating (MWt)	Gross NSSS Rating (3-1/2" Hg) (MW)
864	251	3579	1212
748	238	2894	977
624	218	2322	786

The object is to reduce power density from current BWR/6 offerings from 16-24%. This change would make available more flow per unit power to reduce the void coefficient and attendant transient response problems, increase RPV volume to decrease the magnitude of system pressure rise from turbine trip of MSIV closure, reduce fuel linear heat generation rate to decrease fuel failure and PCIOMR penalties, increase ECCS margins, increaze core operating and shutdown margins, and thus improve NSSS capability to attain high capacity factor, and provide margins to meet future requirements. Current design work on piping layouts in the containment and other A/E interfaces could be completed and utilized with little change. Detailed design studies are required to establish the optimum NSSS rating and/or RPV size.

It is expected that such design changes would add to NSSS and containment capital costs, which could be offset by an increase of capacity factor of 3 .

It is recommended that this proposal be evaluated in detail by NED and be implemented if the detailed evaluation bears out the benefits identified in the preliminary study.

R-2 Reduced Refueling Interval and Rapid Refueling

The development of rapid refueling methods for BWR's would permit the utilization of shortened refueling intervals with the attendant potential for reduced void coefficients, increased core operation margins, and reduced fuel cycle costs. Successful development and application of rapid refueling to BWR/4's, 5's and backlog BWR/6's would be an alternate to partial plant derates for those utilities willing to renegotiate their current fuel cycle contracts to permit six month refueling. It is expected that a significant present worth advantage in fuel cycle costs per plant could be gained with six month refueling intervals.

Application of six month fuel cycles to future BWR/6 proposals would provide added margins to assure plant performance while offsetting the increased capital cost for upgraded design with further reduced fuel cycle costs. In addition, some positive availability benefits between refueling is expected because of slight improvements in void coefficient and more rapid removal of failed fuel.

It is recommended that NED promote the early application of reduced refueling intervals and accelerate the development of rapid refueling.

R-3 BWR/6 Standardization

BWR/6 standardization should be based on the upgraded BWR/6 recommended in R-1 above. Portions of these designs may be utilized on backlog BWR/6 plants which are not upgraded.

Planned product improvement should be an integral part of BWR/6 standardization. Improvements to the standard plant designs may be added as block changes to future product offering at the proposal stage, but only after the proposed improvement has been fully designed and tested.

R-4 Superior Quality BWR

Upgrading future proposals on pressure vessel size, and standardization on the upgraded design stand a good chance of providing a BWR/6 availability advantage relative to competition, and will remove pressure to develop a "new product" on a crisis basis. These short term actions may not, however, produce the truly superior BWR which is needed for successful pursuit of product quality leadership strategy on a long term basis.

It is recommended that effort be applied in mid-1976 to the design of such a superior quality BWR/S. Design objectives for BWR/S should be established along the lines suggested in Table I. The preliminary design effort for BWR/S should be performed by a group insulated from the pressures of producing the backlog and should be targeted for completion by mid-1978.

Development should be done in parallel with preliminary design by the existing (or expanded) Development Engineering Section. Review of the preliminary design, and appraisal of the development results

TABLE 1

SUGGESTED DESIGN OBJECTIVES FOR SUPERIOR QUALITY BWR

1) Economically viable system.

2) Sufficient margin in core design to comfortably accommodate methods uncertainties, operating flexibility, instrumentation uncertainties, manufacturing tolerances, regulatory changes, new phenomena, etc., without derate during any portion of fuel cycle.

   Parameters to be investigated:

   - Core Specific Power
   - Steam Volume
   - Steam Pressure and Pressure Drops
   - Recirculating Flow Rate
   - Reactivity in Voids
   - Flow Control Range
   - Fuel Subdivision
   - Channel Design
   - Inactive Water Volume
   - Hexagonal vs. Square Lattices
   - Refueling Cycle Length
   - Burn-up Optimization
   - Control Rod Characteristics

3) Sufficient margin in core design to permit makeup of both initial and reload cores from 6 standard bundle designs, regardless of operating history or fuel management plan.

4) Excess design margins in all areas to provide for new phenomena, regulatory requirements, improved design methods, etc.

   - ECCS design margins sufficient to meet projected evolution of design bases and criteria
   - Seismic design margins sufficient to meet projected evolution of design bases and criteria

5) Fuel failures at end of life not to exceed 0.2% of bundles.

   - Elimination of PCIOMR operating constraints.

6) Ability to ride out turbine trip without relief valve actuation.

   - Radiological contamination of the containment during blowdown precluded by design.
   - Radiological doses to plant operators minimized in accordance with ALAP criterion.

7) Containment design based on functional requirements verified by test data.

   - Steam-air bubble vibration
   - Relief valve blowdown forces
   - LOCA blowdown forces
   - Suppression pool sloshing

8) Match plant availability and capability to system needs.

   - Refueling outage 15 days or less.
   - Load following/frequency control as defined by utility systems study.

9) Sufficient hardware standardization to permit manufacture of selected items to inventory in GE and vendor manufacturing facilities, and maintenance of spare parts inventories.

10) Backup scram system.

11) Sabotage protection consistent with projected regulatory requirements.

12) Design to prevent long-term build-up of radioactivity.

13) Consideration in the design of the eventual need for plant decommissioning and site reuse.

This page is part of the 1975 "Nuclear Reactor Study". See the 1987 "Update Report" for current status.

should take place during the second half of 1978 (development results are unlikely before the end of the year). Provided the decision is made to proceed, detailed system design should then be performed by the Systems Design Section (q.v.) and detailed hardware design by other Engineering sections as appropriate. Detailed design should be targeted for completion (including component development) by 1981. The BWR/S design, before introduction, can serve as a guide to evolutionary improvements in existing product lines. If feasible, specific features could be introduced through a managed series of block change improvements, each of which should be thoroughly tested, designed and verified prior to commitment.

R-5 Qualification of Core Methods

It is recommeded that a special test and analytical program be devised to be carried out in an actual, modern BWR at end-of-cycle conditions to test and qualify steady state and transient core design methods and correlations. Such a program should incorporate core loading, instrumentation, and operation optimized for the collection of data rather than power production. Important data to be obtained include the measurement of individual channel flow and temperature and void distribution under steady state and transient conditions. probably using specially instrumented bundles. Other data would include cold critical rod positions and clumped rod withdrawal critical patterns versus exposure, Gd reactivity worth versus exposure, fuel isotopics versus exposure, core power distribution, Xe transient reactivity distribution, and local power measurements for different enrichment and burnup UO_2 and PuO_2-UO_2 bundles. Instrumented tests such as turbine trip with 10-30% bypass, all blades out at 75% power should be conducted up to technical specification limits.

It is estimated that such tests would take 20-40 days of power reactor time dedicated to the tests and significant restriction on reactor operation for at least a year. Special instrumented bundles and equipment would be required.

R-6 9 x 9 Fuel Development

If 9 x 9 fuel were selected geometrically similar to 8 x 8, it would have a larger void coefficient, and would be more unstable because of a shorter thermal time constant. Design modifications to substantially reduce the void coefficient are required to balance these effects. It is recommended that commitment of backfit 9 x 9 fuel be contingent upon thorough analysis and adequate thermal-hydraulic and mechanical testing of the proposed 9 x 9 design.

9 x 9 fuel for new proposal offerings should be accompanied by a reduction of core power density and reoptimization of the core pumping system capacity.

R-7 Full Scale Testing of Suppression Pool

Previous testing of the pool swell phenomenon in the San Jose test facility was done with "one-third" sized boiler and drywell simulated vessels. It is recommended that "full-scale" experiments be reperformed with "full-scale" boiler and dry well. This, together with the 1/3 scale test already planned, will establish an improved basis for scaling of subsequent tests.

R-8 Mark III Suppression Pool Cover

It is recommended that the Mark III suppression pool be covered with removable (hinged for LOCA) covers and ventilated to further limit personnel exposure to steam, noise, radiological contamination or other adverse effects following relief valve actuation. In addition, the Mark III design should be reviewed to determine whether there are other separations of functions which are advisable such as separating safety related equipment within the containment for protection and to facilitate compliance with maintenance and operating requirements.

The Mark III suppression pool should be provided, at the first catwalk level, with an isolation decking light enough to be lifted off in a restrained manner (hinged) during a LOCA blowdown. The principal purpose of this isolation decking would be to prevent radiogas release from the pool into the containment when a safety/relief valve operates. Sub-deck ventilation should be provided to carry off any radiogases escaping from the pool during or after safety/relief valve actuation. Another functional requirement for the sub-deck would be to provide acoustic devices that would effectively suppress the noise associated with relief valve blowdown.

A review of the Mark III containment design should be made to determine if functional aspects of the containment design (such as seismic, radiological contamination, noise contamination, primary and secondary containment functions, on-line operating or service functions, etc.) may be separated and made independent of each other.

R-9 BWR System Design Section

It is recommended that a BWR System Design Section be established with BWRSD to set BWR system design requirements, perform trade-off studies to identify the optimum system configurations and performance characteristics, and accomplish system designs. This Section should be totally responsible for establishing design requirements at the overall plant and major system levels, and allocating these requirements to at least the subsystem level to ensure that overall design integration is achieved. The requirements should include, but not be limited to, topics as follows:

Performance
Operability
Transients
Accidents
Seismic
Safety
Availability/Capacity
Reliability
Maintainability

Since the basic strategy is that of a product that is required to have superior availability/capacity, the trade-off studies will measure the design against reliability/availability/maintenance characteristics. These trade-off and optimization studies and results need very early coupling to economic trade-off studies, the formulation of a viable marketing posture and a strategy backed by convincing appraisals of plant capability. The utilities and the Architect/Engineers have to be convinced that more conservative designs, which may cost more are the desired work-horses for power generation based on projected improvement in availability/capacity.

The trade-off studies should include the study of the total plant (not just the NSSS) and this means addition of resources or reliance on an A/E under contract to GE.

The outcome of the trade-off and optimization studies should be the development of clear and definitive system design specifications. Key attention has to be focused on specifications for systems and subsystems which identify reliability/maintainability which will form the base for lower level specifications for components, parts, etc.

The specifications on reliability must encompass all operational characteristics of the system in addition to those systems which are installed for safety purposes and which already have NRC requirements imposed on their function.

The work of this Section will require heavy emphasis on reliability and maintainability which will require a Subsection devoted to these studies and analyses.

R-10 <u>Advance Engineering</u>

Recommendation R-4 above calls for the development of a BWR/S design, based on present understanding of BWR technology, not necessarily [illegible] BWR/2's and since preserved, and executed by a group insulated from the pressures of producing the backlog. It is recommended that this group be a separate, not necessarily permanent, organization, of independent-minded, experienced engineers. The choice should be of generalists, rather than specialists. The skills required should be similar to those in the System Design Section (q.v.); like the S.D.S., access to the special skills and knowledge in the balance of Engineering will be needed to supplement their own resources.

R-11 BWR Reliability/Availability/Maintainability Design Programs

Recommendation R-9 above indicated the need for a distinct organization for system engineering which would include "Reliability Engineering" on a subsection level.

This action is deemed necessary to attract and staff this function (Reliability Engineering) with quality engineers. The present style of organization largely depends on and credits the individual engineer with the responsibility for reliability and maintainability. Heavy responsibilities in other sectors hinder in developing skill in these areas.

It is logical to expect that reliability begins at the system level and requirements and specification for systems and subsystems are expected to flow from "Reliability Engineering" to components responsible for hardware design. In turn, "Reliability Engineering" will be expected to secure data on hardware components and verify that its specifications at the system level are satisfied. To accomplish this responsibility, it will be necessary to have the authority to accomplish the required tests and analyses of tests at the component, subsystem and/or system level. This serves to insure verification of expected performance.

"Reliability Engineering" is also expected to provide maintenance specification and criteria for implementation at the hardware level that is consistent with reliability/availability requirements.

"Reliability Engineering" is to be responsible for accumulation, analysis and distribution of component and system failure history.

It assumes responsibility for corrective actions and programs at the system, subsystem or hardware level.

R-12 Core Management and Reload Fuel Design Function

For the next several years, the NED quality image will be determined by the assistance of Field Service personnel, principally during shutdowns, and, in between shutdowns, on the behavior of the reload fuel and the quality of core management. The findings highlighted the difficulties involved in meeting operating limits and burnup targets with reload cores for BWR/3 through 6 of the current design. Many of the other findings have recognized the need for improvements in the Field Services activity. A parallel uprating of reload core design and core management functions is recommended. This recommendation is a reversal of previous BWRO policy, which sought to reduce GE participation in the core management business and reduce fuel reloads to a standardized product. This policy is believed to be ultimately desirable, but perhaps 5-10 years premature. Specifically it is recommended that personnel of the Nuclear Engineering Design and Operation Subsection (of FD&SA) be increased by about fifteen professionals. NED should recognize that it is important to stay in this business at least for the present, and devise means to make it profitable.

The field information and practical experience developed by this Subsection are important inputs to improvements in design. The Subsection should be given explicit responsibility for seeing that fuel design improvements, based on their perceptions, are made.

R-13 Close Couple Thermal-Hydraulic and Nuclear Engineering Design

Improving the prediction of the steady state and transient performance of a modern, high power density BWR requires unique and extensive interrelation and feedback between the disciplines of heat transfer, fluid flow, and nuclear engineering.

It is therefore recommended that the design code and method development staffs involved with thermal-hydraulics, nuclear engineering, and system dynamics be integrated into one subsection charged with the responsibility for developing integrated design codes and methods.

R-14 Seismic Design Function

Considering the seismic design requirements escalation and the A/E evaluation process, it is recommended that the seismic design function be centralized and up-graded to subsection level to assure that the BWR seismic requirements are properly identified, analyzed, applied and implemented.

R-15 NED Computational Facilities

Make a decision whether to retain the Honeywell computer system or to replace it with an advanced IBM or CDC system. If the decision is made to stay with the Honeywell computer, begin implementation promptly of increased hardware capacity, aiming for a doubling of current capacity in two years. Alternately provide the doubled capacity with a more efficient IBM or CDC system. Improve the operational system so that large problems requiring the use of many peripheral units do not routinely suffer turnaround delays.

Such additional capability is needed to perform initial and reload core design computations for an increasing number of operating plants, to automate design to more effectively use engineering personnel, to facilitate the development and use of more sophisticated models to increase design reliability, to provide for routine analysis of effects previously ignored, and to respond to new licensing and code requirements.

R-16 Fast Scram Qualification

The development of a faster scram drive should proceed on the course now being followed by BWRSD with emphasis on selecting the final drive parameters so as to provide a highly reliable component.

Because of the urgency of fast scram development, it is likely that full-scale production will be scheduled before field feedback on initial factory production is available. Extra testing during the development period is desirable to assure satisfactory performance in the field. It is recommended that additional fully prototypical drive testing facilities be built to test four drives and blades at temperature, pressure, clearances, and appropriate water quality.

FUEL

GENERAL FINDINGS - PRESENT SITUATION

A Fuel Action Plan was initiated in 1971 in response to the Beaton Task Force report. The plan had four main thrusts as follows:

1) Fuel Action Plan

   a) Implement interim operating management recommendations (IOMR's) to reduce pellet clad interaction (PCI) fuel leakers.

   b) Introduce short-term fuel improvements as follows:

      (1) Improved 7 x 7 fuel (shorter pellets, chamfered pellets, thicker cladding, higher clad annealing temperature, getters and better drying).

      (2) Reduce peak power with an improved 8 x 8 fuel incorporating improvements identified in improved 7 x 7 fuel.

   c) Improve analytical capabilities to evaluate technological alternatives for long-range fuel design.

   d) Improve long-term fuel design. With major assistance from Corporate Research and Development (and the Fuel Research Scientific Task Force), identify failure mechanisms, decrease fuel failure susceptibility, and integrate system and fuel design (rod power and power change) to enhance fuel performance.

2) Status

   The new 7 x 7 fuel was first loaded into an operating plant in the spring of 1973. Twenty bundles were examined and showed no failures after about

8500 MWD/T* exposure. Exposure of 330 bundles has reached 5000 to 8000 MWD/T; no failures have been detected to date, but none were expected based on projected failure correlation. Confirmation of the improvement achieved is expected late in the spring of 1976, when exposures will reach the 10 - 12,000 MWD/T level.

The first 8 x 8 fuel was loaded into an operating plant in the spring of 1974. First assessment of the value of the improvement will not be available under September 1976. Substantial information on pellet-clad interaction leakage will not be available until the spring of 1977 when exposure reaches 10,000 MWD/T for BWR/2-3. BWR/4 data will be available spring 1978.

The first implementation of Interim Operating Management Recommendations in 1971 was known as the 10-10-10. It was not effective. Preconditioning, known as PCIOMR** was implemented in July 1973. It has been difficult for operators to implement, but where followed, it has been beneficial. This fuel action plan has been continued aggressively in the last four years. Significant gains have been made. But progress is handicapped by the long development cycle involved in evaluating technical alternatives, and the very long operating cycle necessary for final proof.

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* MWD/T = Megawatt-Days/Ton

** Preconditioning Interim Operating Management Recommendations impose a slow power ramp when raising fuel duty on fuel previously operated at lower fuel duty.

GENERAL RECOMMENDATIONS

Broadly, two primary thrusts are recommended:

1) <u>Improve PCIOMR In Operating Reactors</u>

The only option to early reduction of fuel failure costs, and to improvement in the business image of product reliability is to improve the effectiveness of PCIOMR.

An NRC* official recently indicated all PWR vendors now recommend some form of preconditioning to "preclude pellet-clad interaction", including use of modest ramp rates on initial power increases, after refueling, and after extended periods of operation at lower power levels, just as GE does. In addition, NRC is now exploring incorporation of PCIOMR in the technical specifications. It is therefore imperative that efforts be increased to define as soon as possible a set of operating recommendations that are clearly understandable to plant operators, are effective against fuel failures, are easily implemented by use of BWR control systems and process computers, and minimize the negative impact on capacity factor.

2) This is the prime strategy of the longer range fuel action plan. Because it is crucial to success of the availability strategy and to the business, it should be the pivotal program -- the critical path development -- to compress the time needed to prove prospective designs, and to prepare in parallel for production.

For this purpose, it is recommended that two co-equal strategies be pursued with one back-up strategy. The co-equal strategies should be

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* NRC = Nuclear Regulatory Commission

carried out as critical path developments to the point of production commitment when a choice is to be made between them and the backup. These strategies are:

a) Barrier fuel strategy with potentially high leverage on plant economics and performance.

b) Reduced fuel duty strategy based on more available technology with earlier determination of feasibility but less attractive economics and performance.

c) Fine motion drive backup strategy as a means of assuring preconditioning yet achieving the fast scram requirements of BWR/6 and related regulatory trends.

SPECIFIC FINDINGS AND RECOMMENDATIONS - NEAR TERM

Improvement of PCIOMR

F-1 Finding

Failure of fuel already in operation can be significantly reduced by increasing PCIOMR effectiveness.

R-1 Recommendation 1

a) Assign individuals with recent startup experience to prepare revised instructions for plant operators.

b) Improve process computer software.

c) Assign resident engineer skilled in PCIOMR operation to each of six to eight sites that constitute major portion of fuel risk during next few years.

d) Explore improvement of preconditioning techniques.

e) Maintain PCIOMR flexibility in technical specifications.

Reduction of Refueling Time

F-2 **Finding**

a) Reactor availability can be substantially increased through reducing time required for refueling, when it is on the critical path.

b) Reload fuel will often be needed on relatively short notice if fuel failures cause premature shutdown; otherwise availability will be sacrificed. NED contingency planning for short-term fuel delivery needs continuing study.

R-2 **Recommendation 2**

Reduce time required for refueling -

a) by development of better fuel sipping techniques and equipment;

b) through more rapid refueling techniques, equipment and program management, training programs and application of integrated NED and I&SE skills;

c) by replacement fuel planning to be sure emergency fuel will be available for premature refueling on short schedule.

Core Management

F-3 4 **Findings**

a) Core management can reduce PCI failures by avoiding shuffling of partially exposed fuel into more active zones of the reactor.

b) A six-month refueling cycle would offer energy cost savings, but could reduce availability. Achievement of short refueling time makes six-month refueling more attractive.

R-3 **Recommendation 3**

Avoid power level increases in bundle swapping for operating reactors.

R-4 Recommendation 4

Evaluate six-month refueling as shorter refueling outages are achieved, and offer to utilities as it becomes attractive.

SPECIFIC FINDINGS AND RECOMMENDATIONS - MID-TERM

Thick Clad Corner Rods

F-5 Finding

Thick cladding on fuel rods in the corner of the bundle nearest the control rod may achieve substantial reduction in pellet clad interaction (PCI) failures.

R-5 Recommendation 5

Reevaluate and consider adopting a thick-clad corner rod configuration as the design standard; initiate parallel programs for licensing and manufacturing process development.

Modified Pre-Conditioning Maneuvers, Control Rods and Drives

F-6 Finding

It may be possible to modify present PCIOMR procedures in such a way as to reduce the unfavorable impact on capacity factor, particularly if such modified procedures could be implemented with modified control rods and drives. The modified control rods might, for example, be designed to have short notch spacing and the rods to have short grey tips.

R-6 Recommendation 6

Explore the feasibility of improved preconditioning power maneuvers that could be implemented more easily than present PCIOMR.

Cladding Concerns

F-7 Finding

a) The price of zirconium products has been increased slightly and probably will increase more in the near future.

b) Although Wilmington-produced tubing meets NED engineering specification, joint fuel companies indicate that its uniformity of dimension and finish is not as high as that of another supplier's tubing. There appears to be a market preference for the other supplier's.

c) Zircaloy shells from NED's supplier, while within specifications, are becoming lower in hardness and tensile strength as impurities are reduced.

d) External corrosion on fuel rods has not recently been a problem but might become troublesome as exposures increase.

R-7 Recommendation 7

a) Recognize potential for sharply rising zirconium prices. Qualify additional vendors.

b) Improve tube rocking process to alleviate imperfections caused by mandrel pickup and to meet closer dimensional tolerances.

c) Review hardness specifications placed on NED's supplier. Be sure the supplier does not add troublesome impurities to meet this specification.

Powder Production

F-8 Finding

Ammonium oxalate pore former and ammonium bicarbonate binder with GE Co. process powder produces pellet sintered density nearly independent of green density and with a very stable microstructure.

R-8 Recommendation 8

a) Explore potential of pore former and binder adders with powder made by the ammonium diuranate (ADU) process.

b) Explore theoretical density specifications for pellets.

SPECIFIC FINDINGS AND RECOMMENDATIONS - LONG TERM

Barrier Strategy

F-9/10 Finding

Basic approach of barrier strategy seems sound. It includes thin plate metal surfaces, coextruded inner surfaces and "buried" layers, and coatings deposited from suspension. The data at present are insufficient to draw conclusions as to probable success. Hollow pellets, vibratory compaction and inverse rifling represent related approaches that deserve evaluation along with barriers AB Atom Energie indicated at the joint ANS-CNA meeting in Toronto April 27-30 that they were exploring inverse rifling means to reduce PCI, apparently along the lines proposed by CRD*

R-9 **Recommendation 9**

Continue at high priority the basic study of pellet-clad interaction failure mechanism.

R-10 Recommendation 10

a. Continue at high priority the engineering evaluation of five or so most promising candidate barrier designs that appear to avoid failures by attaching the basic failure mechanism.

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* Corporate Research and Development

b) As soon as possible, but not later than the end of 1975, select two or three candidate barrier design configurations for further development and initiate parallel manufacturing program and pilot irradiation program.

c) Also evaluate hollow pellets, inverse rifling and vibration compacted fuel.

<u>Reduced Fuel Duty Design Strategy</u>

F-11 <u>Finding</u>

Reduction in individual fuel rod linear heat rate, e.g., by making rods smaller and putting more in a bundle as in 9 x 9, has high potential for reducing PCI fuel leakers.

This strategy would require analysis of NSS system design to assure ability to backfit into BWR/4, 5, 6, and assessment of its overall economic attractiveness.

Performance of reduced fuel duty strategy and the barrier strategies in parallel will permit selection of best approach in late 1977.

R-11 <u>Recommendation 11</u>

a) Determine the realistic reduction in fuel duty that could be obtained in BWR/6 by using fuel designs that have a lower heat generation rate, e.g., 9 x 9.

b) Evaluate system requirements and costs of accommodation reduced-duty fuel in BWR/4, 5 and 6.

<u>Fine Motion Control Backup Strategy</u>

F-12 <u>Finding</u>

a) A fine motion control rod drive should make it easier to observe PCIOMR for burn-up compensation and during startup.

In turn, this improvement in PCIOMR effectiveness should reduce the plant capacity factor limitations and operating limitations of present PCIOMR capability.

b) The basic fuel strategies may eliminate the need for PCIOMR, and related fine motion control development could be discontinued when this point is reached. On the other hand, if fine motion control remains technically and commercially sound, it can be introduced as an addition to the fuel strategy implemented.

c) Present indications are that an AEG screw jack drive can be redesigned to fit the space available on a GE-BWR, and that fast scram rod operating objectives can be met (proof Aug. '75) in a 50/50 mixed control concept, although the precision drive train will require considerable development to prove reliability. Addition of a short vernier drive to the present notching piston drive to impose a fine motion each notch step may be a feasible alternative.

R-12 Recommendation 12

a) Develop a fine motion control drive as a back-up to the barrier and reduced fuel duty strategies.

b) Coordinate development to allow introduction of fine motion drive at proper time if main strategies cannot eliminate this requirement.

SPECIFIC FINDINGS AND RECOMMENDATIONS - OTHER FUEL CONSIDERATIONS

End of Life Failure Modes

F-13 Finding

a) Few data are available on fuel exposed to more than 15-20,000 MWD/T*

b) As pellet clad interaction failures decrease and physics limitations are overcome, exposures will increase and other failure modes can be expected to show up. Some of these end-of- life failure modes can now be anticipated, but others probably cannot.

R-13 Recommendation 13

a) Analyze potential end-of-life failure modes for telltale, early-warning symptoms.

b) Maintain a surveillance program on nonleaker rods at intermediate and high exposures.

c) Run exploratory lab tests and special power reactor tests to explore fuel susceptibility to new end-of-life failure mechanisms.

Transient Loads On Fuel

F-14 Finding

a) Fuel specifications pertainint to PCI failures to not cover transients that may be imposed on regular operations or occur during emergency shutdowns. A fuel duty document being prepared will cover this area.

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* MWD/T - Megawatt-Days/Ton

b) Transients will extend beyond normal full load for periods which may vary from a few seconds to about half a minute. While such transients have not produced coincident failures, the fuel cannot be preconditioned for those power levels, and the tendency for PCI* failure may be increased by the longer transients.

R-14 Recommendation 14

a) As soon as possible, evaluate fuel duty for all transient conditions and establish the design basis.

b) Run tests to determine whether short duration transients increase PCI failure probability.

Incipient Cracks

F-15 Finding

Unfailed fuel of moderate exposure may contain multiple incipient cracks (not all the way through the clad) that cannot be detected by present nondestructive testing equipment. It is important to know whether or not this is true both for reasons of transient failure analyses, and to increase understanding of failure mechanisms.

R-15 Recommendation 15

a) Slit, clean and inspect inside surface of a significant, random number of unfailed fuel rods having medium to high exposure. Determine number of incipient cracks and correlate with past rod history.

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* Pellet Clad Interaction

b) Develop more sensitive methods of nondestructive testing, capable of detecting incipient failures in unfailed fuel rods.

I· national

F-16 Finding

NED's image with Joint Manufacturing Companies is not entirely satisfactory. More attention is needed to maintain NED's international position.

R-16 Recommendation 16

Examine NED fuel specifications, manufacturing and inspection processes for compatibility with international environments and modify or make special provisions as necessary.

Spent Fuel

F-17 Finding

Spent fuel bundles could be used instead of the natural uranium blanket planned for about one-eighth of the initial core in BWR/6 and might also be incorporated in BWR/5 initial cores. This could offer utility customers savings in fabrication and uranium supply, and relief in spent fuel storage requirements.

R-17 Recommendation 17

Use spent fuel instead of natural uranium whenever possible for the core blanket of BWR/5 and BWR/6 initial cores.

Fuel Test Facilities

F-18 Finding

a) GETR will require extensive modification for relicensing. The present license expires in October 1976; an extended outage will be required for modification.

b) The RML facility cannot accommodate full-length test rods and this imposes a serious limitation in fuel failure diagnostics, including incipient crack investigations and end-of-life failure mode analysis.

c) Cask handling facilities need improvement to handle spent fuel return casks.

R-18 Recommendation 18

a) Pursue timely relicensing of GETR as an essential NED objective.

b) Expand RML to accommodate full-length fuel rods and cask handling.

SPECIFIC FINDINGS AND RECOMMENDATIONS - FUEL RELATED COMPONENTS

Channels

F-19 Findings

a) From the viewpoint of creep, channels are not projected to have fifteen-year operational life desired by customers, but can probably be expected to last 8 - 10 years, which represents two complete refueling cycles.

b) Corrosion is a problem, but it can probably be controlled by proper heat treatment.

c) Difficulty is still being experienced in bending thick channel material at Wilmington.

R-19 Recommendation 19

a) Continue high-priority manufacturing development program for 100-120 mil channels and for channel heat treatment. Implement these methods as soon as possible.

b) Consider addition of flow restrictions in space between channels at top of core to reduce maximum pressure differential across channel wall.

c) Consider mechanical design means other than thickness (such as embossing) to stiffen channel walls.

Control Blades

F-20 Finding

Control blade manufacturing dimensional tolerance limits are not likely to be maintained up to the point of blade installation due to minor flexing of the blade. This is probably not significant since the blade is guided by fuel channels, but minor deviation of blade straightness or twist from factory specifications upon arrival at the reactor site has caused needless concern.

R-20 Recommendation 20

a) Check blade envelope tolerances needed in operations against those needed in manufacturing. Eliminate possible confusion at reactor site.

b) Check test data on control rod insertion with channel expansion and blade interference.

c) Resolve tolerance problems, stiffness problems and welding processes by joint efforts of the BWR Systems Department and Nuclear Fuel Department.

Spacers

F-21 Finding

Spacer design has limited seismic capability as presently developed for 8 x 8 fuel. Feasibility of extending current design approach to 9 x 9 has not been demonstrated.

R-21 **Recommendation 21**

a) Continue development of 8 x 8 spacer to fully satisfy present and future safe shutdown earthquake requirements, with demonstration of acceptable thermal-hydraulic performance.

b) Initiate development of 9 x 9 spacer; explore design alternatives as necessary.

ELECTRICAL, CONTROL AND INSTRUMENTATION

SPECIFIC FINDINGS AND RECOMMENDATIONS

Relief Valve Augmented Bypass (REVAB)

F-1 **Findings**

REVAB is an option offered on BWR/4, 5 and 6 to provide the capability of accepting sudden loss of electrical load without reactor scram, and to perform this function at an equipment cost significantly less than that of a more conventional full steam bypass system.

REVAB uses programmed opening and closing of Safety/Relief valves, along with selected rod motions and recirculation flow reduction, to supplement an approximately 35% bypass capacity in controlling reactor pressure until the power level can be brought within the control capability of the bypass system.

No REVAB systems have yet operated, but seventeen are committed, one for operation in late 1975. Recent NED transient studies resulting from other system considerations raise serious doubt as to the ability of the REVAB option to perform effectively its intended function (i.e., prevent scram or loss of electrical load), particularly near the end of a fuel cycle.

The fundamental nature of this technical problem, therefore, raises doubts that a truly effective REVAB system, meeting the desired BWR quality image, can be implemented.

There are alternate ways to provide the function of accepting loss of electrical load without reactor scram; for example, providing 100 steam bypass capability, but this alternative would add significantly to plant cost.

R-1 Recommendations

a) Review the ability of REVAB to meet its design objectives. Consider modifying the REVAB operational objectives in light of impact on plant operational availability.

b) Review alternative means for providing the capability to accept loss of electrical load without reactor scram. Compare with REVAB on technical and economic bases to form the basis for future approach.

Control Rod Drive System

F-2 Findings

A fast-scram drive with control rod insertion in significantly less time than presently designed for BWR/1 through 5, is planned for BWR/6. Tests indicate it can be provided by a design modification to the present control drive mechanism. A development/design program is currently underway, with the goal of providing fast scram drives for initial operation of the first BWR/6 (Cofrentes, 1978). Because of the urgency of this program, it is highly likely that full-scale production will begin before all of the desired field feedback data are available on the initial factory production run. Extensive performance and life testing during the development period is needed to assure a qualified design and minimize the technical risks involved. Adequate test facilities must be available for testing prototype drives with blades under operating temperature, pressure, and clearances, and appropriate water quality during the development program.

An attractive future design consideration, a fine motion control drive, is under development. A fine motion drive could be effective in alleviating some of the present fuel conditioning operational constraints. The advantages offered by a fine motion control drive in reducing fuel transient duty caused by plant maneuvering justify a development/design program. NED currently has a program underway, but it is limited to the evaluation of a mechanical ball-screw drive (with hydraulic scram) now used by AEG.* The AEG design is complex, expensive, foreign to NED, and would have to be significantly modified to be used with the NED-BWR configuration.

The current program should be broadened to evaluate other approaches, specifically the potential to add a "vernier" slow motion to the planned fast scram drive. This would have the advantages, if feasible, of 1) using the already developed NED hydraulic design and 2) maintaining the capability for fast scram and slow drive in every rod if desired

R-2 Recommendations

a) Continue the present BWRSD program for fast scram development. Give the program sufficient high level management review to assure that it maintains the required priority, program direction and resource level needed to make available well tested drives for initial operation of first BWR/6 (Cofrentes, 1978).

* Allgemeine Elektricitats-Gesellschaft - NED licensee

Assure that adequate developmental test facilities are available for testing of prototype drives with blades under simulated operating conditions.

b) Initiate a program in parallel with the present evaluation/redesign of the AEG control drive, to evaluate alternate approaches to the fine motion control drive. Specifically, evaluate the potential for a "vernier motion" added to the planned hydraulic fast scram drive. Schedule the program for a final choice of fine motion drive concept by year-end 1976 at the latest.

Focus the fine motion control drive development toward these objectives:

(1) Retention of as much as possible of the present, proven hydraulic drive design.

(2) Incorporation of fast scram, fine motion capability in a single drive, if feasible.

(3) Capability to incorporate the new design in all reactors in design or construction.

<u>Dynamic Control and Load Following Capability</u>

F-3 <u>Findings</u>

The BWR dynamic control systems appear to have adequate stability margins, in either manual or automatic range, over the power-core flow range in which each is designed to operate.

There is a question of the flexibility and range needed in the system to satisfy the possible future requirements of electric utility network for following load demand changes.

BWR systems are capable of meeting the load following commitments made in contracts and proposals to date. However, increased load following capability may be needed in the future to remain competitive and, if so, the technical feasibility and cost of achieving it must be known.

Automatic control, via control of recirculation flow, is constrained to the range of 75% to 100% rated power with normal control rod positioning, and proportionate variation when control rods are set for lower power levels. To get greater than a 25% power variation requires moving control rods, a very slow process.

To adjust to daily load changes, and maximize system economics, a utility may desire that an individual generating plant change its output by 50% or more of rated capacity in perhaps an hour. BWR systems are limited relative to fossil plants in achieving such wide swings, even at slow rates, because of rod motion restrictions. There is reason to believe that PWR plants suffer the same penalty to at least some degree. Up till now, with nuclear being a small part of generation, this restriction in power swing capability has not been a serious concern, but it may be in the future.

In the "intermediate" load following range, for load changes of up to 25% over a period of 3 to 10 minutes, such as might be required for tie line thermal backup, the BWR responds at least as well as fossil plants and the PWR, if the BWR is positioned within its flow control range. BWR/6 capability is 10% power change in 10 seconds, followed by a further change of up to 15% within the next minute, if this total change can be made by flow control.

The BWR response is quite limited in the fast load following mode required for network frequency control. Here only small amplitude changes (perhaps ± 3% of rated power) are required, but with almost instantaneous response. The BWR is also less effective in handling sudden, large load demand transients such as might occur after severe electrical network disturbances. The current BWR control system would not control frequency effectively under the situation where the BWR is isolated, supplying its own electrical network. NED offers an "isolated grid" option to meet this requirement, but no utility has elected it, and the ability of this option to function effectively is not proven. Both fossil plants and the PWR are more responsive than BWR in their ability to meet frequency control requirements and handle fast load demand transients.

The limitations of the BWR in following load demand changes are not limits imposed by the control system design, but are inherent limitations imposed by fundamental BWR system parameters; namely, 1) limited stored steam volume, 2) large effects of steam voids on reactivity and power, and 3) low rates of power rise which can be accommodated by the current fuel design.

Ned is evaluating new control schemes which may somewhat increase the capability of the BWR/6 in frequency control performance, but to make significant changes in BWR response would require alleviation of the fundamental constraints noted above.

R-3 <u>Recommendations</u>

a) Perform an overall systems evaluation of the technical feasibility of, and the economic justification for, modifying the BWR dynamic control system in the future to provide increased capability for 1) normal electrical grid frequency control duty and 2) coping with network disturbances (such as might lead to isolated grid operation).

b) Make the evaluation a joint NED/Electric Utility Systems Engineering team effort, with appropriate participation by Large Steam Turbine-Generator.

Set-Point Drift

F-4 Findings

Approximately 10% of the abnormal occurrences reported to the Nuclear Regulatory Commission by nuclear power plant operators since 1972 have involved unplanned changes in the set-points of instrumentation installed in protective circuits of BWR's such as to cause the set-point to be out of compliance with the plant's Technical Specification. In many cases, the Technical Specification has been found to be needlessly tight. Solutions have also been found for most of the remaining problems.

NED established a program in mid-1974 to correct the problem and the required Engineering Change Authorizations (ECA's) for all affected plants are scheduled for issue by August 1975.

Excessive occurrence of set-point drive problems affect the BWR quality image adversely, and are a source of extra maintenance work, and therefore inconvenience, to plant operators. Timely correction of the present situation is essential to meet operating availability/capability goals.

R-4 <u>Recommendations</u>

a) Continue to give the required priority to this problem and its corrective program to assure that at least the present schedule for issuance of Engineering Change Authorizations is met.

b) Take the initiative with the involved customers, and with the Nuclear Regulatory Commission, as appropriate, to assure that the required changes get implemented on a timely basis.

<u>Electrical and Instrumentation Subsystems</u>

F-5 <u>Findings</u>

Historically, the electrical, control, and instrumentation systems have not been a major contributor to BWR plant shutdown time.

The implementation of the electrical and electronic C&I hardware appears to be generally well done and in accordance with appropriate industrial standards.

Several development programs, now underway, should further reduce availability problem in C&I systems, namely:

a) Neutron Monitoring System

New sensor and seal developments for longer operating life; bottom-entry probes for easier maintenance and replacement.

b) Solid-State Safety System

Introduction of solid-state logic with ability to test while maintaining full protection; replacement of pressure switches with pressure transmitters to reduce drift problems.

c) Power Generation Control Complex

To give better control room wiring and equipment placement; fewer wiring errors.

d) Nuclenet Control Complex

To give the plant operator greater, more convenient, and more timely data display; will improve operator ability to operate the plant efficiently and avoid errors which could lead to down time.

The following recommendations are made to assure that the maximum benefit in reliability/availability gains is achieved with these new developments.

R-5 Recommendations

a) Solid-State Safety System

At an appropriate time in the detail design stage, hold a design review, involving experienced engineers from other Company operations, of the adequacy of design for electrical noise immunity.

b) Neutron Monitoring System

Review and modify the design of the Traveling In-core Probe system to solve the position read-out problem and the problems of bending and contamination of the guide tubes.

c) Nuclenet Control Console

Schedule design reviews of the Display Control System (third quarter 1975 and first quarter 1976) involving experienced personnel from outside NED. Arrange to get early data on the Honeywell 4400 (Nuclenet computer) relative to hardware/software reliability problems.

d) Plant Auxiliary Electrical Systems

Centralize the responsibility for specification and integration of plant electrical systems to provide the leadership needed to improve availability. Use this central responsibility to place increased emphasis on specification of non-safety-grade but high availability-related power systems.

e) Quality/Reliability/Maintainability Program

Assign to C&I Standards and Qualification Engineering the additional resources and the responsibility to review and approve the qualification programs for all systems and components for which C&I has responsibility.

d) A recent inspection of an early Hammel-Dahl design after 300 hours of functional testing (a very limited endurance test) reveals some substantial design problems for which corrective design modifications are urgently needed despite the vendor's apparent complacency.

e) The review team concludes that a qualification test of a production valve complete with actuator and controls in a recirculation pump test loop is necessary to provide reasonable assurance that these valves will operate satisfactorily from startup to first refueling. Even so, the extended life characteristics of these valves remain to be learned from actual operational experience.

f) The operational risks are high because present schedules call for roughly forty flow control valves, fourteen from Hammel-Dahl and the remainder from Fisher, to be on site before any go into operation in a NSSS. Furthermore, the first four Hammel-Dahl valves and the first eight Fisher valves will be installed in other overseas NSSS's. There is no back-up presently planned for these valves.

g) Only an outline/assembly drawing was available in the BWRS Department for review.

h) The Review Group concludes that present plans to deliver so many flow control valves to site without test verification of the design entails unwarranted risk and justifies extra-ordinary effort to mitigate the situation.

R-1 Recommendations

a) BWRS Department should immediately organize and convene a special test group to review the design details of both the

Hammel-Dahl and Fisher flow control valves and specify design modifications to be incorporated in the first production valves. This task group should utilize the most knowledgeable technical people available in addition to BWRSD design engineering and development engineering. The task group should also include NED purchasing because serious vendor relations problems are foreseen.

b) Accelerated, extra-severity endurance testing with pressure, temperature, flow, water chemistry, etc., should be performed on a production valve design from each vendor to reveal mechanical and life problems as early as possible.

c) Valve procurement should be delayed to the slowest rate and latest possible dates that can be tolerated by construction requirements accepting out-of-sequence installation of the FCV, if necessary.

d) The recirculation pump/flow control valve test loop at Bingham Pump should be engaged and activated as soon as possible to obtain several thousand hours of simulated reactor control operation at pressure and temperature with flow modulations representative of, but more severe than, expected power plant operating conditions. A production valve, complete with actuator and control systems, should be tested. The test loop should include a FCV bypass valve and provisions for adding a BWR/6 jet pump assembly test when it becomes available to verify the solution to the jet pump flow induced vibration problem. The test loop at Bingham Pump would be used for production acceptance testing of pumps (approximately

100 hours/pump); however, additional endurance hours of FCV testing might be scheduled by extending the test period of production pumps.

Flow Control Bypass Valve

Operates in parallel with main flow control valve during plant startup.

F-2 Findings

a) Basic design and application of the valve is sound.

b) Valve not as critical or technically difficult as main flow control valve.

c) GE does not review or have much knowledge of design details.

R-2 Recommendations

a) Accelerated, extra-severity endurance testing in the Flow Control Valve test loop is needed. This should be included in the Bingham test loop.

b) GE should obtain a better knowledge of vendor design and details and processes, including changes.

Crosby Safety Relief Valves (SRV)

F-3 Findings

a) BWR/5 and 6 will use Crosby direct spring loaded SRV's instead of the Target Rock or Dresser pilot operated valves.

b) Crosby valves are expected to be more reliable because they do not employ a pilot valve system, which is the source of most of the trouble with the Target Rock and Dresser valves. Therefore, they should significantly improve the reliability of the SRV system.

c) The Crosby valves will include a pneumatically powered operator to provide quick relief action as required to meet BWR/6 transient control requirements.

d) The net seating force in the Crosby valve in the normal operating condition will be much lower than the main element in the pilot operated valve, and may be somewhat more sensitive to leakage due to debris under the main seat than previous valves.

e. Large Steam Turbine Department valve experts believe that it is difficult to maintain leak tightness in frequent operation as may be required in BWR/6.

f) The likelihood of frequent SRV blow-down will inevitably lead to the need for regular maintenance (especially on the valves with the lowest pressure set points). If SRV maintenance is required between Maintenance-Refueling outages, it will contribute to unavailability.

R-3 Recommendation

The addition of two or three steam turbine type bypass valves installed in parallel with the SRV's to relieve the duty on the lowest set-point valves should be studied and evaluated.

Main Steam Isolation Valves (MSIV)

These valves have an important safety function of isolating the reactor vessel from the external environment and in the event of a primary system break, they serve to isolate the containment. They remain open during normal operation and close on signal in three to five seconds. NRC imposes a leak tightness requirement of 11.5 standard cubic feet per hour leakage under 30 psi differential air test.

F-4 Finding

a) These valves have had a long history (approximately 37 reactor years) of problems, primarily failure to meet leak tightness tests and actuator malfunction.

b) Failure rate has responded to improvement programs to the extent that on a Duane plot, incident rate has dropped from 6×10^{-4} incidents per valve-plant hour at 200,000 accumulated valve-plant hours to 3×10^{-4} at 2,000,000 plant hours.

c) While actuator problems should be amenable to aggressive improvement programs, it is the judgment of the Review Group that for a valve of this configuration, the present high level of maintenance, requiring frequent lapping to meet tightness requirements, will continue for some time.

d) The development of effective tools, fixtures, and procedures for valve lapping has greatly reduced maintenance time.

e) Vendors show little, if any, interest in doing fundamental development work to improve the leak tight worthiness of these valves.

f) NED has two options to improve valve performance.

(1) Implement a valve development program of its own to explore existing ideas for improve leak tightness and demonstrate their effectiveness.

(2) Change to a valve having the configuration and essential features of a steam turbine stop valve. Such a valve would be more costly and have higher pressure drop but, based on steam turbine experience, would require less maintenance to meet leak tightness requirements.

R-4 Recommendation

a) BWRSD should establish a MSIV development facility and implement an aggressive development program to find ways to improve the leak tightness of these valves. Successful improvements should then be called for in purchase specifications.

b) In the meantime, a joint program with I&SE should be developed to continue to facilitate MSIV maintenance.

Jet Pumps

F-5 Findings

a) Mechanical failures, due to flow induced vibration, have been encountered in the upper end of the jet pumps in operating reactors.

b) Strong excitation of this pump assembly occurs because much of the power of the recirculation pumps is dissipated in turbulence in these jet pumps.

c) NED Engineering has a major redesign planned for BWR/6 that provides substantial improvements in mechanical strength and stiffness, raises the natural frequencies, greatly reduces the possibility of misassembly, and by another design change, eliminates one of the potential cause of trouble in the earlier design.

d) The direction and plan for redesign, as described to the Review Group is comprehensive and sound.

R-5 Recommendations

a) Because of the inherently high excitation for vibration, the redesigned jet pump assembly should be tested under simulated operating conditions and with thorough instrumentation, before the new design is released finally for BWR/6. If it must be released before test, then it must be recognized that some rework may be required, based on the tests.

b) The jet pump assembly should be mounted in a test fixture that will reasonably simulate the hydraulic conditions and should simulate also the mechanical stiffnesses and masses so that resonant frequencies will be similar to the reactor situation. This test should be performed in conjunction with the flow control and flow control bypass valves/recirculation pump test loop which has been recommended for activation as soon as possible.

c) In the future, development and performance testing of jet pumps should include high frequency response instrumentation to gain greater knowledge of excitation phenomena.

Flow Induced Vibration Facility

F-6 Finding

a) Analytical methods alone are not sufficient to predict the vibration behavior of components where the interaction of a compliant structure with fluid flow is affected by structural vibratory modes, frequency, structural and fluid damping.

b) The ability to model such flow induced vibration is also quite limited.

c) Reliance on field vibration measurements, although helpful in verifying designs and solving field problems after they occur, will not assure vibration-free new designs.

R-6 Recommendation

The task group endorses the general concept of installing a large or full scale flow induced vibration test facility that will permit major components to be tested full scale to determine their vibratory stress levels and structural stability under realistic conditions of fluid flow excitation.

Dynamic and Static Loads on Structures Resulting from Loss of Coolant Accident (LOCA)

F-7 Findings

a) Suppression pool phenomena have been identified.

b) Analytical models are reasonably well developed and appear to produce calculated loads that are conservative relative to test results.

c) Dynamic and structural loads have been released to Architect/ Engineers.

d) Loads on structural elements will be verified by the 1/3 scale model test program now in progress.

e) Surprises that will require major changes in Architect/ Engineer's structural design appear unlikely.

R-7 Recommendations

a) The 1/3 scale model test program should be carried out as planned as rapidly as possible and expanded, if necessary, to resolve uncertainties.

b) The possibility of a direct pipe break jet impingement on the weir/pool and its asymmetrical effects should be examined. Preliminary judgment is that this is not serious.

Suppression Pool Behavior and Resulting Dynamic Loads on Containment and Structures Caused by Safety Relief Valve (SRV) Operation

F-8 Findings

a) Analytical models need further development.

b) Several issues to be resolved:

(1) Options being considered for reduction of air vending loads:

- Design of an appropriate discharge muffler (The Kraftwerk-Union*) type quencher is the only design with full scale test data and, therefore, it might be the most licensable design without additional full scale tests.)
- Elimination of air clearing bubbles by maintaining steam in the SRV discharge pipes.

(2) Determination of maximum suppression pool temperature permissible to avoid high temperature variation.

(3) Determination of the magnitude of second SRV pop.

(4) Determination of the containment loads resulting from multiple/consecutive SRV activity.

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* German Licensee (KWU)

(5) Verification of the combined affects of simultaneous multiple SRV action.

(6) Dynamic behavior of containment shell, particularly dynamic buckling criterion.

c) Heavy dependents of 1/8 scale model tests being prepared and KWU test data.

R-8 Recommendations

a) Resolution of these issues should be given the highest priority not only because of the effect on licensability of Mark III, but also because of the need to address these issues on Mark I and Mark II.

b) Available experienced, knowledgeable personnel elsewhere in NED should be marshalled for the direction and execution of work necessary to resolve this problem.

Combined Effect of Phenomena Identified Above on Mark III Containment Design and Licensing Requirements

F-9 Findings

a) There is a growing conviction among NED engineers that the Nuclear Regulatory Commission (NRC) will require the Mark III containment to be designed for some combination of the loads referenced in Findings 5 and 6 above.

b) It is not unreasonable to postulate that SRV operation can occur concurrently with a LOCA event.

c) This may increase previously estimated containment loads and may result in structural design changes in the suppression pool portion of the Mark III containment.

R-9 Recommendations

a) SRV model verification test program should be carried out as planned as rapidly as possible and expanded, if necessary, to resolve uncertainties.

b) Resolution of this issue should receive high priority, not only because of the effect on licensability of Mark III, but also because of the need to address these issues on Mark I and Mark II.

c) Because of the importance and urgency of resolving this issue along with those listed in 7 above, and the present already heavy engineering workload in the Boiling Water Systems Department (BWRSD), an infusion of capable manpower is recommended as stated in 8 above.

d) BWRSD should resort to a reasonable overkill in this situation and release promptly to Architect/Engineers a set of containment design loads due to the combination of SRV operation and a LOCA event which is certain to be acceptable to the NRC; the purpose being to minimize containment redesign work by Architect/Engineers, even field modifications in some cases, that may occur while waiting for completion of model test verification.

Radiation Concerns on Mark III

F-10 Findings

a) Because of access requirements for maintenance during operation in Mark III, the probability of radiation exposure of personnel is increased over that in previous design.

b) Operator re-entry following an isolation event could be highly restricted for a time of the order of two days.

c) An undesirable degree of uncertainty exists in calculated predictions of radiation exposure of plant personnel.

R-10 Recommendations

a) This situation should be given much further study.

b) Re-entry restrictions following normal SRV action need further analysis.

c) Effects of otherwise tolerable SRV leakage need to be studied.

d) A thorough, objective reassessment of the use of the Mark III containment concept as a long-term future standard BWR product should be made.

C. F. Braun's Views on Structural Design Feasibility of Mark III Steel Shell Containment for STRIDE Plant

F-11 Findings

a) Design of Mark III containment requires a high level of sophistication in structural analysis and design.

b) C. F. Braun's capability and work is impressive.

c) C. F. Braun is confident it can specify an adequate Mark III steel containment provided that static and dynamic loads previously discussed for all Mark III containments are not increased substantially. (Exception -- dynamic buckling criteria and dynamic response is still an open question, but believed to be manageable)

R-11 Recommendation

If dynamic buckling behavior of containment cannot be satisfactorily modeled analytically, a 1/10 size physical test model should be built and appropriately tested. This work could be farmed out to an external organization with requisite expertise.

Possible Required Retrofits of Mark I and II

F-12 Findings

a) The NRC may require that some Mark I and II pressure suppression systems be modified eventually.

b) NRC is aware of the situation and can be expected to require a resolution of the problem during the next several months.

c) The results of model verification tests now underway in connection with Mark III ar necessary to further evaluation of Mark I and Mark II.

d) This situation will add considerably to an already heavy engineering workload.

e) Although any necessary modifications to Mark II will necessarily have to be made before these plants become operational, any modifications required for Mark I plants now in operation could require either a shutdown for retrofitting, or possibly an extension of a maintenance and refueling outage. In either case, it could contribute to reduced availability of operating Mark I plants during the next few years.

R-12 Recommendations

To minimize the negative effect of any necessary retrofitting on future plant availability, the necessary field work should be carefully planned, tools and procedures developed, using the resources of I&SE, so that the General Electric Company will be in a position to offer to owners the service that can reduce the down time required.

Nuclear Plant Availability Improvement Program

F-13 Findings

a) The potential for decreased BWR nuclear plant availability is real, recognizing the new problems that are surfacing, such as torus vibration, pipe cracking, and the probability of increasing NRC inspection requirements.

b) Availability improvement programs being implemented including Advanced Maintenance Planning Service II, Refueling-Maintenance Outage Service, and the increasing involvement of I&SE are all positive steps, but need support with a greater sense of urgency.

c) The most meaningful and effective action that can be taken now to improve nuclear plant availability is to implement a revolutionary step in the area of BWR service.

R-13 Recommendations

NED should establish a BWR nuclear service business operation with the following priority of objectives:

a) Establish and maintain a system of reporting in detail all component failures and causes of nuclear plant shutdowns, funding the system initially as a cost of doing business.

b) Provide BWR owners a broad range of services that will enhance their utilization of nuclear plants and improve overall plant availability; be supportive of and function cooperatively with I&SE.

c) Develop the service operation into a profitable business.

Control of Design of Purchased Components

F-14 Findings

a) The emphasis in present procurement practices on reliability and availability of purchased equipment is unlikely to result in attaining availability improvement goals.

b) NED relies almost entirely on vendor design expertise to produce components and equipment to performance and functional purchase specifications. In fact, NED acts like an Architect/ Engineer with regard to purchased equipment; however, unlike the Architect/Engineer, NED warrants the system and is expected by the customer to be responsible for it.

c) Because of this reliance on vendor expertise, NED has not developed sufficient in-house purchased component design expertise and has exercised little control of design details of critical purchased equipment.

d) The LWR industry practice of committing NSSS's before designing them, coupled with long procurement cycles, generally precludes adequate qualification testing of new purchased components.

e) The feedback of field operating experience and its translation into vendor design improvement actions is a slow and difficult process because of vendor inertia.

R-14 Recommendations

a) That NED recognize the unlikelihood of its availability improvement goals being attainable with continuance of the present engineering-purchasing-vendor relationships.

b) BWRSD should take steps, with management support, to acquire and develop design expertise pertaining to selected purchased hardware, particularly valves, including MSIV, SRV, Flow Control Valves and large shut-off valves.

c) NED management should implement a procurement policy calling for the development of vendor relations that provide NED engineering review and approval of design details and materials of critical purchased components, even though this may increase the cost of these items.

MATERIALS, PROCESSES AND CHEMISTRY

GENERAL CONCLUSIONS

Since the design, more than twenty years ago, of the first BWR's, there has been a continuing improvement in the understanding of, and knowledge about, the behavior and application of materials in the BWR environment. Despite such progress, a number of materials problems are of current concern. Two of the most serious problems are: a) the stress corrosion cracking of stainless steel piping; b) the solution, activation, and deposition of cobalt as radioactive cobalt-60. The Division goals for improved availability and reliability will require solutions to these problems and further improvements in materials performance.

The persistence of these materials problems in the face of dedicated effort by competent people indicates that the materials effort must be given increased stature and strength to achieve the essential goals and to assure the incorporation of the latest and best materials information in the nuclear steam supply design.

The general conclusions, therefore, are:

1. The NED BWR materials and chemistry effort should be increased.
2. The existing organizational structures should be modified to increase the influence of materials expertise on management decisions by:
    a) creating a unified Materials and Processes Component at San Jose,
    b) strengthening the new Manufacturing Laboratory at Wilmington,
    c) imposing an added discipline on materials and processes control in the design process.

SPECIFIC FINDINGS AND RECOMMENDATIONS

Stress Corrosion Cracking

F-1 Findings

a) Under special circumstances, type 304 stainless steel can be susceptible to stress corrosion cracking (SCC) in BWR environments.

(1) SCC has occurred in only about half of the 45 BWR's which have operated.

(2) Cracks occur almost exclusively in heat affected zones near welds.

(3) Cracks have occurred near 76 out of 18,000 welds.

(4) Cracking in pipes has occurred only in seamless pipe which is used in sizes up to and including 10-inch diameter.

(5) Cracking is more likely in restricted-flow location.

b) A smaller number of stress corrosion cracks have also occurred in nitrided stainless steel parts, in furnace sensitized reactor components, and in heavily cold-worked bolts.

c) It has been demonstrated in the laboratory that the BWR environment can accelerate fatigue crack growth in low alloy and carbon steels as well as in stainless steels. No in-service failures of this type have been observed in carbon steels.

d) Because of the design of the plants and the "leak before break" characteristic of the cracks, the stress corrosion cracking problem has not been a direct threat to safety, but inspection, repairs, and replacements have been costly in terms of money and down time.

R-1 **Recommendations**

The solution to stress-corrosion cracking problems must continue to have the highest Division priority with concurrent efforts on cause and possible required changes in environment, materials, and design. Specifically, NED should:

a) Establish a high level program to develop and qualify replacement materials for 304 stainless steel in BWR primary coolant piping.

b) Replace some 4" diameter recirculation pump discharge valve bypass lines with a stabilized stainless steel on a selective and monitored basis, to obtain field experience.

c) Expand studies on stress-corrosion cracking to obtain more data on Inconel and nitrided materials, and to search for alternate choices for 304 stainless.

d) Expand studies on the origin and magnitude of localized stresses in pipes and other stainless steel components for correlation with cracking susceptibility.

e) Increase effort on studies of environmental effects on the growth of fatigue cracks to provide guidelines for water chemistry control.

f) Determine the relationship between changes in reactor operating practices and the occurrence of stress-corrosion cracking.

g) Recommend to customers that no changes in BWR water environment be made without careful and extensive review by BWR Systems Department.

Reactor Pressure Vessel

F-2 **Findings**

a) The probability of a sudden disruptive failure of the reactor pressure vessel (RPV) is judged to be less than 1×10^{-6} per reactor-year. This estimate applies to all presently designed BWR plants.

b) A detailed analysis of RPV integrity in BWR's under loss of coolant accident (LOCA) condition was last made in 1968; it showed that RPV integrity would be maintained. Much more recent reviews by NRC and ACRS have reached similar conclusions.

c) Calculations of peak pressures under postulated anticipated transient without scram (ATWS) conditions have been made within the past year for various BWR's. Peak nressures in the 1600 to 1650 psig range have been calculated for certain BWR/3 plants and considerably lower values for other BWR's. These pressures are well within the capacity of the vessel.

d) NED's studies provide strong support that fatigue crack growth in vessel steel under BWR environment conditions does not have an adverse impact on RPV integrity. Other NED work indicates that stress corrosion cracking would not occur in RPV steels in BWR water within specifications.

e) Cracks have been observed in the cladding around feedwater nozzles at Millstone and Dresden-2, but were small enough to be readily removed. Ultrasonic indications of possible cracks at Pilgrim are being monitored on a continuing basis. In BWR/6's the cladding has been eliminated around all nozzles so this type crack should no longer be observed.

f) The BWR/6 has been designed to accommodate currently specified and reasonably anticipated future RPV inspection requirements. However, inspection of RPV's in older plants, if required, can only be performed to a limited extent with currently available equipment and methods.

g) The oldest BWR plants (e.g., Dresden-1, Humboldt Bay and Big Rock Point) did not have jet pumps and have the pressure vessel closer to the core than is the case with later reactors. This has resulted in higher radiation levels and the potential for a higher degree of radiation embrittlement than will be encountered in subsequent reactors. No operating problems are foreseen, but thermal annealing of the RPV may be desirable at a later date to assure meeting cold hydrostatic test requirements.

R-2 Recommendations

A key NED objective must be production of the highest quality reactor pressure vessels. Specifically, NED should:

a) Increase effort on participation in government-industry test programs to broaden understanding and minimize cost.

b) Update LOCA (loss of coolant accident) integrity analysis for BWR/6.

c) Document the evaluation of vessel integrity under anticipated transient without scram (ATWS).

d) Improve in-plant and in-service inspection technology, equipment and techniques for crack detection, and crack propagation rate measurement.

e) Prepare plans for vessel field inspection, repair, and annealing in the event that such action should be required.

f) Continue study efforts in fatigue crack growth and stress-corrosion behavior in pressure vessel steels in normal and abnormal reactor environments to obtain additional assurance of long pressure vessel life.

Radioactive Contamination

F-3 Findings

a) Radiation levels of operating plants increase with time and, as a result, total radiation exposure received by workers at all types of light water nuclear plants has been increasing over the past five years. Average total exposure per plant per year (as reported in WASH-1311) has increased from about 180 to 320 man-rem between 1969 and 1973 for BWR's, and from 220 to 770 man-rem for PWR's over the same period. Maximum permissible exposure per individual is now closely regulated. Utilization of increasing numbers of workers for maintenance and operation, particularly the former, is now a pattern in the industry.

b) If allowed to continue unchecked, this trend, plus possible stricter union and government limitations on permissible individual radiation exposure levels, will generate increasing manpower requirements for maintenance. With present designs and operating procedures, it appears that major decontamination methods will have to be developed.

c) The exact cause for the radioactivity increase and its dependence on water chemistry, composition of components and NSS operation are not known.

R-3 Recommendations

A key NED objective should be to design reactors not requiring decontamination during their design life. Specifically, NED should:

a) Modify design to facilitate maintenance, accessibility for repairs, and in-service inspection, with particular attention to minimizing crud traps.

b) Identify and specify low cobalt grades of stainless steels and Inconels for critical piping and other components in contact with reactor water and feedwater.

c) Develop and specify low-cobalt hard facing alloys to replace high-cobalt Stellites wherever possible.

d) Increase development efforts to understand better the origin, transport, and deposition of radioactive contaminants throughout the system.

e) Improve control of feedwater chemistry and in particular, re-evaluate design changes that have introduced forward pumping of drains.

f) Broaden program to monitor radiation levels in operating plants to establish correlations between water chemistry, materials of construction, and mode of plant operation.

g) Immediately initiate a program to evaluate different approaches to decontamination.

h) Provide guidance to utilities in operation of radwaste systems.

Level of Materials Effort

F-4 Findings

The findings in previous sections have pointed out specific needs for extra effort on stress corrosion cracking and radioactive contamination by Co^{60}. Other materials areas exist where continuing, although less severe, problems should receive more attention. Components involved include reactor pressure vessels, control rods and control rod drives, reactor core internals, steam separators and dryers, pumps, isolation and safety relief valves, condensers, heat exchangers, electrical insulation, and protective coatings and paintings. While active work is in process in most of these areas and no significant deficiencies have been identified, the Review Group believes that additional effort is necessary to meet the high availability/capability goals on which Division strategy is based.

R-4 Recommendations

A significant increase in materials engineering development and design is recommended to meet NED commitments and to establish a basis for enhanced quality and reliability. NED should increase effort on:

a) Control rod materials and failure mechanisms with particular reference to factors controlling life-time of control blades.

b) Crevice-corrosion studies and studies of corrosion of carbon steel to assess its potential for broader reactor applicability.

c) Dimensional stability of stainless steel for improved performance in reactors and improved manufacturing yields.

d) Development of improved gasket, seal, and packing materials

e) Zircaloy channel materials and processing, with particular emphasis on extension of life.

f) Surveillance of materials in vendor-supplied components, such as pumps, valves, condensers and heat exchangers.

g) Radiation damage studies, with particular reference to in-core sample irradiation monitoring and reliable experimental verification of neutron spectrum and fluence at critical in-core locations.

Materials Information System and Control

F-5 Findings

a) Although materials problems are among the commonest facing BWR Systems Department, the department's material functions are diffused and the role of materials engineering needs strengthening. Existing laboratory facilities at Vallecitos and San Jose are excellent, but not large enough or accessible enough to the materials engineers.

b) NED relies largely on the ASME design codes for design procedures, materials specifications and allowable stress limits. More use should be made of materials engineers. More systematic effort should be undertaken to analyze and get additional data, to establish policies on data use, or to develop Company specifications superior to those basic to the materials industry.

c) The Wilmington Plant is initiating a quality control laboratory, and a manufacturing process development and control laboratory, but additional manpower is needed.

d) Manufacturing managers at Wilmington indicate the desirability of minimizing development changes and of fixing as early as possible the design of a new component. Manufacturing would like to have the design drawings and specifications be firm before going into production.

e) Reliability, maintainability, and plant life impose greater requirements on BWR materials than was originally perceived. For example, the range of properties of stainless steels processed to ASME and NED specifications may affect consistent behavior in the system environment, and tighter specifications may be desirable.

R-5 Recommendations

Strengthen the effectiveness of materials use in boiling water reactors by organizational changes and improved materials information and control procedures. Specifically, NED should:

a) Establish a Materials and Processes component at San Jose to include Materials Engineering and a Materials and Process Development Laboratory for structures and fuels and for water chemistry.

b) Within the Materials and Processes component, establish a Materials Data function responsible for continuing data analysis, control, and publication; for development of data standards and policies; and for representation of NED on materials code committees.

c) Establish a reactor equipment manufacturing and fuels Quality Control and Manufacturing Process Laboratory at Wilmington.

d) Establish a policy that prior to production of a new component, it must pass a qualification test, and upon completion of the test, the design and manufacturing processes are thereby fixed and any further process changes require formal change approval.

e) Tighten specification control of critical materials, establish approved sources of materials, and maintain closer contact with critical materials suppliers.

PRODUCTION, PROCUREMENT AND CONSTRUCTION

SPECIFIC FINDINGS AND RECOMMENDATIONS

Proliferation of Designs

F-1 Finding

Proliferation of designs of fuel and reactor components adds cost, and control problems confronting manufacturing at all locations, and hence hinders NED in reaching its goals in availability/capability.

R-1 Recommendation

a) Implement plans to have only one size fuel rod and one size fuel pellet in manufacturing at Wilmington at any point in time. This means phasing out of present 8 x 8 fuel design before phasing in BWR/6 fuel design. It is the Review Group's understanding that this is the present NED plan.

b) Make one standard design of fuel rod at Wilmington and make up variations through arrangement of rods (by enrichment) in bundle make up.

c) Review the decision to reduce the BWR/6 fuel pellet diameter by .006" and reduce the fuel rod wall thickness by .002". The cost of tooling charges is high and the technical problem of fuel rod leaks must be re-evaluated.

Quick Response to Fuel Failures

F-2 Finding

The sharp increase in number of fuel bundles reaching mid-life radiation exposure time will result in a corresponding increase in the number of fuel bundles requiring replacement and may result in some plant refueling shutdowns earlier than expected with consequent need for early fuel replacement. Future production plans should provide for adequate replacement fuel bundles.

R-2 Recommendation

An immediate review should be made to determine total production of replacement fuel bundle needs.

A preliminary study shows the need for more bundles per year in the next four years for replacement of failed fuel than are included in present NED plans.

If the above study shows a requirement for additional inventory, Wilmington should be authorized to build that inventory, starting immediately while excess capacity is available at Wilmington and also while fuel is available from ERDA.

Channel Creep

F-3 Finding

a) Present zircaloy channel walls deform under the imposed hydraulic load and oxidize under ambient conditions.

b) Difficulties have been encountered in the manufacture of .120" wall channels to present drawing tolerances.

c) Binding of control rods is a possibility from channel creep. Routine periodic measurement of scram time provides early warning of any such tendency, however.

R-3 Recommendation

a) Change manufacturing tolerances on channel bending requirements to enable manufacture and use of more creep resistant thick wall channel (.120").

b. Institut> long range alloy development program aimed at improved creep and corrosion resistant zircaloy.

Zircaloy Tubing Quality

F-4 Finding

Although Wilmington-produced zircaloy tubing meets current NED engineering specifications, there are indications that its quality does not equal that of another tubing supplier. There appears to be a market preference for tubing from the other supplier.

R-4 Recommendations

NED should implement a program to improve quality of Wilmington-produced zircaloy tubing from the standpoint of:

a) Ability to maintain consistently round tubing and internal diameter tolerances.

b) Ability to eliminate small surface flaws.

c) Improve measuring and gauging equipment used in inspecting tubing.

Fuel Manufacturing - Cost Effectiveness

F-5 Finding

Cost of fuel manufacturing is constantly increased due to design changes, UO_2 cost and steep increase in overhead costs. High volume output is required to make the fuel manufacturing a viable business.

Offshore reload orders are increasingly being supplied by offshore fabricators.

R-5 Recommendations

a) Develop a strategy to retain domestic and international reload fuel orders in order to assure economic loading of the Wilmington Plant.

b) Carefully control future capacity oriented investment programs.

c) Support quality improvement programs.

d) Reduce engineering and manufacturing costs by early design standardization and reduction of changes.

In-House Manufactured Components

F-6 Finding

Quality of products manufactured at NED Plants in San Jose, Wilmington and CBIN* are not causing significant plant availability/capability problems. (The exception is fuel. The problem here is basically engineering, not manufacturing, and safety is not an issue.)

R-6 Recommendation

Concentrate on availability/capability projects versus shop capacity oriented investment projects.

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* CBI Nuclear Company - GE joint venture with Chicago Bridge & Iron Company which manufactures reactor pressure vessels.

Vendor Supplied Components

F-7 Finding

If NED is to attain its availability/capability goals, it must improve the following:

a) Design standardization

b) Design and supplier/qualification formulation of qualification plans and commitment of facilities.

c) Engineering resources in Nuclear Energy Division to control vendor selection and design.

d) Manufacturing capacity to meet present schedule requirements.

e) Spare parts program.

f) Management commitment to establishing an integrated reliability program for critical procured components.

R-7 Recommendations

a) NED should establish one standard design for each critical component and own all detail drawings for this design.

 (1) Subsequent production for this component should be on a build-to-print basis, supplemented with detail specification requirements.

 (2) Suppliers should be formally qualified to produce each standard design component.

b) (1) Develop a standard valve design specification which includes requirements for:

 - new supplier qualification
 - design analysis for code-controlled and non-code controlled parts
 - design qualification testing

- subsequent production valve tests
- approval of all detail parts drawings
- interface requirements

(2) Prepare specific ordering data for each generic valve type and application, to supplement the standard valve specification.

(3) Implement the design and supplier qualification requirements above on all new designs presently under development.

(4) Life test such critical valves as the main steam isolation valve, flow control valve, bypass valve and the safety/relief valve.

(5) The requirements for, and scope and depth of design reviews should be issued in an engineering section instruction.

c) (1) Increase GE-applied engineering effort on new component designs. New component designs should be handled by the most experienced senior engineers.

(2) Separate responsibility for new component designs from engineering support of components in the field.

(3) Eliminate suppliers with insufficient and/or inadequate engineering support.

(4) Combine under engineering leadership all personnel involved in specifying and procuring critical components on new reactor designs. After full qualification of a new critical component, the procurement activity could then be assigned to the normal NED purchasing organization.

d) (1) Standardize on one design for each application to aid casting and capacity availability.

(2) Perform detail resource loading studies prior to placing new valve orders.

(3) Perform an in-depth study of the nuclear quality casting industry, including GE needs versus potential suppliers.

e) Procure, as part of the original component order, selected critical spare parts which experience has shown to be needed during plant startup (valve packing, pump seals, valve actuator, etc.). These spare parts should be delivered concurrently with the original equipment.

f) Greater emphasis must be placed on making reliability analysis an integral and required part of the design process.

Vendor Capability

F-8 Finding

a) Vendor components are a significant cause of availability problems.

b) Vendor manufacturing capacity is marginally adequate.

c) GE contributed value in the Nuclear Steam Supply System is too low to control quality of critical components of the Nuclear Steam Supply System.

R-8 Recommendations

a) Initiate a program aimed at "owning the design" of critical GE-responsible valves and pumps.

b) Consider manufacturing in-house those critical components where reliable outside sources are not available. Use best-suited utility group facility for this purpose.

c) Initiate in-house manufacture of components where major cost reductions and reasonable return opportunities exist -- such as installation of the proposed solvent extraction system for reclaiming UO_2 scrap at the Wilmington Plant.

Flow Control Valve

F-9 **Finding**

There is a high probability that a qualified recirculation system Flow Control Valve will not be available to meet the 1977 startup of the initial BWR/5 power plants.

R-9 **Recommendations**

a) Accelerate plans to qualify flow control valves (Hammell-Dahl and Fisher).

b) Order two sets of FCV and hydraulic servo spares (actuators and others to be available at Tokai 2 at startup.)

c) Initiate a system study to re-evaluate the long range system decision. Include -

- Technical performance (load following and effect on fuel performance)
- Plant capability/availability
- Cost and schedule including design and vendor qualification.

d) Systems studied to include:

- Constant speed pump/flow control valve
- Variable speed pump supplied by:
  - BWR 4 type M-G Set
  - M-G Set excited by cycloconverter
  - Cycloconverter

- Cycloconverter exciting the field of a wound rotor pump motor
- Other

Vendor Capability

F-10 Finding

There are numerous changes in the electrical and control system due to many on-site changes, and to shipment of Control and Instrumentation components prior to completion of design. The time required for documentation of field changes is too long. A new procedure for staging and testing complete Power Generation Control Complex/Nuclenet System at Control and Instrumentation manufacturing plant should greatly reduce field changes and delays on BWR/6 plants.

R-10 Recommendation

A review of the field change system should be made with the objective of reducing the time required to document field changes in NED drawings.

Control and Instrumentation Manufacturing Space

F-11 Finding

Proposed C&I move to Raleigh/Durham would have adverse impact on plant availability/capability delivery and cost due to:

a) Move while increasing output

b) Move while introducing new product

c) Training of new exempt/hourly (50% of engineering and manufacturing exempt will not move)

d) Tight schedule for staging successive units

e) Multi-million dollars additional cost to move

f) 3000 mile communication

g) GE history has shown that a move such as this one (which involves a new product, a new plant, a new work force and a new location, coupled with a fast build-up in production schedule) is the sure formula for failure.

R-11 Recommendation

Expand in San Jose by:

a) Leasing space for offices and laboratories

b) Using space made available in buildings A, B, C, J, from above for staging, testing and other C&I manufacturing operations.

Communication

F-12 Finding

Geography (3000 miles separation) of Engineering/Manufacturing personnel presents many problems affecting communications and efficiency of day-to-day operations at Wilmington and CBIN.

R-12 Recommendation

Initiate a study of the "Frequency of Communications and Interfaces" to determine the optimum split between San Jose and remote site engineering and manufacturing personnel. For example, could all systems and test engineering personnel be located at San Jose, and all component design personnel be located with the process development and manufacturing personnel at Wilmington?

This page is part of the
1975 "Nuclear Reactor Study".
See the 1987 "Update Report"
for current status.

Productivity at Reactor Site

F-13 Finding

There are numerous opportunities at reactor construction sites to improve construction efficiency, schedules, cost and availability/capability. The scope and responsibilities of:

a) Contractor

b) Architect/Engineer

c) Nuclear Energy Division

d) Installation & Service Engineering

e) Utility

are not as clearly defined and documented as necessary to achieve these ends. The rapid increase in new plant construction and existing plant refueling cycles (30/yr by 1980) requires a clarification of plant site responsibility for greater operating efficiency.

R-11 Recommendations

a) GE should reconsider assignment of responsibility for GE plant site installation and service, to a single organization, instead of dividing it between two as it is at present. Installation and Service Engineering has fifty years of experience in the installation and service of power plant equipment and performs this work on nuclear plants for turbine generator and electrical equipment.

b) Therefore consideration should be given to accelerating the transfer of responsibility from NED to Installation and Service Engineering for the management and technical direction of installation and service of the Nuclear Steam Supply System thus reducing the number of involved organizations performing work at the plant site. In this case, project relationship responsibility would remain with NED, as it does with Large Steam Turbine Generator for their equipment. Plant startup responsibility would also remain with NED.

c) To achieve early improvement in supporting service capability, and specific improvement in refueling time, more resources should be made available for:

(1) Development of procedures, and recruiting and training of people to provide the service.

(2) Applying NED engineering support to operating plant problems as a first order of priority.

(3) Redesign of special equipment and tools in the NED scope, that have caused significant extension of refueling outages in operating plants.

(4) Design, manufacture and construction of additions to special Installation and Service Engineering training equipment and facilities for the refueling process, to include both existing and future BWR designs.

Plant Availability - Short Range Improvement

F-14 Finding

Outage time during a refueling cycle (averaging 60-70 days) is a major loss to power plant availability.

R-14 Recommendation

A task force of experienced manufacturing engineers, and field service personnel (comprised of NED, Installation & Service Engineering and Manufacturing Engineering Corporate Services personnel) should study and document step-by-step procedure and tools required, to reduce the necessary time to perform the tasks during a refueling period. Objectives of this study should be to cut the outage time by at least 50%.

QUALITY CONTROL SYSTEMS OVERVIEW

FINDINGS

F-1 **Nature of Major Quality Problems**

The greatest challenge to fulfillment of availability objectives centers on those responsible for engineering design.

F-2 **Quality Control System Overview**

The staffing and organizational status of design assurance effort in the Boiling Water Reactor Systems Department* has not been such as to optimize its effectiveness across all development/design activities of the Department.

F-3 **Field Information Feedback**

a) In the past year the Boiling Water Reactor Operation has taken many actions to strengthen and broaden the portion of its product quality control system concerned with acquiring and responding to field information feedback. Progress in this area has special significance in that the knowledge and corrective actions that result apply directly to improving the availability of existing plants, and indirectly to all future plants.

b) There is no engineering component responsible for across-the-board reliability/maintainability management.

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* As used in this Quality Systems discussion, "Boiling Water Systems Department" refers to the department as it existed prior to the reorganization of April 1975.

F-4 Governmental Requirements

Governmental requirements applying to nuclear plant suppliers are severe and expanding, along with penalties for violation. It is essential that disciplines requisite to compliance be established and maintained.

F-5 Self Audits

a) Recent P&QAO audits have revealed instances of non-conformance to Boiling Water Reactor Systems Department engineering practices and procedures, some involving issues that are basic to achievement of design integrity. The present allocation of engineering resources does not provide for optimum balance between engineering design "operating work" and engineering design assurance work.

b) Acrimony in Product and Quality Assurance Operation/ Boiling Water REactor Systems Department relationships and attitudes is a problem which itself needs to be overcome through mutual effort in both organization.

Recommendations

R-1 In each Department of the Division responsible for design of any portion of the Division's product offering, establish a design assurance component, principal responsibilities of which shall be to formulate, document, administer and appraise conformance to practices and procedures essential to or effective in assuring design excellence. The component shall be so placed in the Department's organizational structure as to clearly affirm that its design assurance responsibilities apply to the Department's total scope of product design responsibility.

Three activities recommended for immediate emphasis by such components are:

a) Review of Department Engineering Practices and Procedures, aimed at simplification, clarity and mutual consistency.

b) Training Courses or Seminars aimed at assuring that designers are familiar with and understand established practices and procedures.

c) Overview of the design verification process.

General Managers responsible for product design and the Manager P&QAO should mutually explore means for improving cooperation in the process of P&QAO's audits of design activities. Increased tact on the part of representatives of P&QAO, coupled with affirmative support of the process as a necessary way of life in the nuclear business by design managers appear to be appropriate first steps.

R-2 Establish a "reliability" component within each department responsible for any portion of the Division's product offering responsible for systems availability planning ... i.e., for:

a) Recommending availability-relevant field information needs.

b) Analyzing failure and repair data for precise causes and circumstances.

c) Reliability/maintainability modeling by system, subsystem, equipment, component, etc.

d) Establishing/maintaining applicable analytic methods and computer codes.

e) Identifying major availability upgrading needs and opportunities.

f) Recommending upgrading design program, to be undertaken by any of the Engineering Sections, along with priorities based on potential for availability improvement.

g) Monitoring program progress and results to assure fulfillment of program objective.

h) Integrating with the Operating Plant Services Section of Boiling Water REactor Projects Department, in order to establish common perspective on service as well as design approaches to availability upgrading, and to assure that the availability model properly provides for and reflects progress in both.

MANAGEMENT/INFORMATION SYSTEMS

GENERAL FINDINGS

1. The NED management systems and procedures are not a significant contributor to the Division's business and product problems, with the exception of the need for Systems Engineering and Systems Design Reviews.

2. In general, there seems to be reasonable understanding and compliance with the extensive procedures and management systems implemented to date within BWRO.

3. A BWRO Change Control Board has been organized recently to monitor and authorize changes, with an appraisal of the overall effect of changes on all projects. It is too early to assess the effectiveness of this Board.

4. Field Construction Management adequately performs the limited scope of GE's responsibility as varyingly specified in present contracts.

5. Project Managers are properly operating within the limited scope of their authority and responsibilities, as established by the BWR Projects Department.

6. The range and complexity of the BWR business have outgrown the capabilities of existing management information systems to provide sufficient "real time" control data. Extensive plans to streamline and update the management information system are currently under way.

7. Adequate procedures, management systems and experienced trained personnel within Operating Plants Services are available to conduct pre-operational and startup testing as scheduled.

8. Operating Plants Systems adequately monitor plant performance, report problems, and plan corrective action.

9. Offshore projects tend to be much more "Turnkey Project"-oriented because of customer demands. Project Managers offshore have more involvement and greater self-sufficiency overall than domestic Project Managers.

SPECIFIC FINDINGS AND RECOMMENDATIONS

Systems Engineering Organization

F-1 Finding

There is no visible Systems Engineering Organization and the procedures for overall BWR Systems Design Reviews need improvement.

R-1 Recommendation

Establish and implement a Systems Engineering Organization and a formal Systems Design Review process.

Design Standardization

F-2 Finding

There appears to be no clear-cut program or assigned responsibility for achieving a Standard or Reference BWR/6 design that will provide increased performance margins.

R-2 <u>Recommendation</u>

Appoint a separate design team with strong leader to develop a BWR/6 Standard or Reference BWR/6 design of significantly increased "CAPABILITY."

<u>Development and Training of Design Engineers</u>

F-3 <u>Finding</u>

Design engineers are isolated from current Field Construction and Operating Plant Problems.

R-3 <u>Recommendation</u>

Establish a rotational training program to increase the "real world know-how" of hardware design engineers.

<u>Verification of Calculational Models</u>

F-4 <u>Finding</u>

There appears to be no consistent program for verification of calculation models.

R-4 <u>Recommendation</u>

Investigate ways to get additional experimental data to check calculational models. In addition, calculational models should be more thoroughly reviewed for consistency of predictions with other models in use, prior to release to designers.

<u>Completion of Lead BWR/6 Plant Design (Cofrentes)</u>

F-5 <u>Finding</u>

The incomplete design for 218"-BWR/6 is pacing construction of the lead BWR/6 plant. The situation is further complicated by its offshore location (Spain).

R-5 Recommendation

Review the construction schedule to determine which phases of construction can be carried out with minimum risk of rip-out or changes.

High Percentage of Control and Instrumentation Field Changes

F-6 Finding

A high percentage of Field Changes are related to Control and Instrumentation. Quality Control manpower becomes more involved in field installation work than in performing independent Quality Assurance.

R-6 Recommendation

Investigate causes and establish a corrective action program such as:

a) Advance the schedule for establishing and releasing BWR Preliminary Systems Specifications and Requirements, so that C&I will be provided with earlier start.

b) Review the man-loading for C&I design and manufacturing to determine whether or not adequate manpower and required skills are in place.

c) Select reasonable ranges early for required design parameters to minimize the impact of "later's" and "to-be-established" on plant construction.

d) Consider the transfer of additional C&I-experienced personnel to Quality Assurance to provide independent review capability.

GE Scope of Technical Direction

F-7 Finding

There appear to be too many ambiguities in contract scope of "Technical Direction of Installation" to be provided by GE.

R-7 Recommendation

Evaluate the feasibil' of including a clearly-defined normal or extended scope "Technical Direction of Installation" package in the contract as a condition of sale.

Availability of Architect/Engineers' Drawings and Schedules

F-8 Finding

Project Management's access to Architect/Engineer drawings and schedules appears to be a function of the A/E experience, expertise and degree of cooperation.

R-8 Recommendation

Specify contractually the A/E drawings and schedules to be provided to NED.

Project Management

F-9 Finding

The present Project Management Information Systems do not provide sufficient "real-time" cost and schedule measurements and control.

R-9 Recommendation

Continue to develop a streamlined and updated Project Management Information System to provide better "real-time" cost and schedule measurements and control, and expedite implementation.

Division Quality/Reliability Goals

F-10 Finding

There is a lack of a positive Division-wide high-visibility Reliability Improvement Program to achieve increased plant availability/capacity factors.

R-10 Recommendation

Set realistic reliability improvement goals and establish a Division-wide program to be achieved on a measurable basis and expedited schedule.

REGULATORY CONSIDERATIONS

SPECIFIC FINDINGS AND RECOMMENDATIONS

Response to Nuclear Regulatory Commission Regulation

F-1 **Finding**

NED's policy toward the Nuclear Regulatory Commission regulations is, of course, one of compliance, but it is not unusual for only partially responsive, or inadequately supported information, to be submitted initially to Nuclear Regulatory Commission. This has sometimes led to the need to significantly revise or supplement information already submitted, and has resulted in embarrassing and difficult customer situations, when a design had to be modified in order to get it licensed. Changes in concrete or steel already in place or on order have occurred also. It should not be inferred from this that NED has a policy of postponing design confirmatory work until forced by Nuclear Regulatory Commission to do it, but such postponement is not uncommon. Examples are the various problems which have come up on the Mark III containment and the BWR/6 reactor. Additional problems of this type in connection with these products are to be expected. In some cases it has been more a matter of NED not recognizing the need for design substantiation, than of NED intentionally postponing needed work of this type. The result however is much the same. Design changes and sometimes backfitting follow.

A greater effort should be devoted to the avoidance of unsubstantiated commitments or, at the very least, highlighting such commitments and displaying the program and schedule for substantiating them. The alternatives to the unsubstantiated design (should its substantiation later be found to be impractical) should be clearly established in internal documentation. Admittedly, the present practice has the characteristic that the cost of doing work to substantiate license submittals is deferred as long as possible. It is doubtful however that this saves any money in the long run.

R-1 Recommendation

NED should adopt a policy of avoidance of unsubstantiated design representations in licensing documents, or where such avoidance is not completely possible, such representations should be highlighted to the Nuclear Regualtory Commission and the program and schedule for their substantiation should be displayed. The alternatives to the unsubstantiated design (should its substantiation later prove impractical) should be clearly established in internal documentation. Systematic, documented, and detailed study of each design proposed to be submitted for licensing should precede such submittal, and management should at that time commit support to the programs agreed to be necessary to substantiate the design.

Need to Anticipate Regulatory Action

F-2 **Finding**

New regulatory requirements are seldom a surprise to the NED Licensing component. However, other parts of the organization seldom take specific action on a given prospective requirement until the requirement becomes firm, even though its eventual advent is believed to be inevitable. The result is that the design and development components are not infrequently in the position of being unprepared to implement a new requirement. If adequate preparations are made in advance of their becoming requirements, NED may avoid some expensive design changes or backfitting due to these issues. The appropriate preparatory action varies from case to case. In some cases design studies should be made; in others work should be started with vendors of key equipment; in others experimental work is needed.

R-2 **Recommendation**

NED should adopt the practice of formally and systematically identifying probable future regulatory requirements, and establishing programs to implement, in a timely and economical fashion, those future requirements which are judged to be reasonable or inevitable.

The following recommendations deal with the more significant of the many likely future regulations and are intended to be illustrative rather than exhaustive.

Limit on Annual Man Rem Per Plant

F-3 Finding

It is likely that the Nuclear Regulatory Commission will eventually issue "guidelines" which will tend to limit man rem/plant-year exposure. Such "guidelines" usually have the force of regulations. The problem will become more serious as plants get older due to buildup of long-lived radioactivity. To a large degree the solution to the problem is in the hands of the plant owner who must devise working procedures and control his personnel, but some contribution to a solution can be made by NED in its role as a designer and supplier.

R-3 Recommendation

NED should adopt the practice of establishing a radiation exposure budget for each system, area, etc, and should not consider a piece of design work complete until it can be shown that the design is such that, by using practical working procedures, the plant owner can expect to keep annual man rem exposure within the budget throughout the useful life of the plant. This recommendation has particular applicability to STRIDE*, but should apply to all plants in which GE furnishes the reactor.

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*Standard Reactor Island Design

Period of Safety of Unattended Reactor

F-4 Finding

Present NED product safety standards require that, in the event of a reactor upset or accident, all actions necessary in the next 10 minutes to assure that the reactor is in a safe condition (i.e., that the core cooling is maintained) take place automatically, so that the operator need take no action himself until 10 minutes have elapsed, if then. A proposed NED revision to this requirement (based partly on observed operator response to unexpected events) is that the "hands-in-pockets" period should be lengthened to 30 minutes.

R-4 Recommendation

NED should undertake to establish a logical basis for the minimum length of time that a reactor should be capable of safely remaining without human intervention, and if that time is longer than 30 minutes, should establish a program for implementing the requirement thus implied.

Plutonium Handling, Storage and Transport

F-5 Finding

The handling, storage and transfer of separated plutonium is not of itself a matter of direct concern to the suppliers or operators of reactors. However, the economy of the fuel cycle depends critically on plutonium. Thus it is of great importance to NED that the Nuclear Regulatory Commission, Environmental Protection Agency and other bodies are in the process of creating regulations which conceivably could make

MO_2 fuel so uneconomic as to preclude its large scale use in the power reactors.

NED is by no means unaware of the problems and implications of MO_2 fuel. However this awareness has not yet advanced to the inclusion of physical security features for the storage of new MO_2 fuel at reactors.

R-5 Recommendation

New fuel storage provisions at reactors should be adaptable to the segregation, behind physical security barriers, of new MO_2 fuel. This recommendation is particularly relevant to STRIDE.

Failed Fuel Identification

F-6 Finding

The possibility exists also that Nuclear Regulatory Commission might require some "improved" technique of detecting failed fuel. No more sensitive means of doing this is known than the "sipping" technique now used. This technique does require opening up the reactor, but so of course does the removal of any failed fuel which might be found by any other means. Equipment for locating failed fuel has been installed and has been routinely used for about 11 years on one large reactor (the Hanford N Reactor) but is expensive and is justified only because, in that case, an extremely prompt shutdown must be made if the cladding of the metallic uranium fuel is breached even slightly.

R-6 **Recommendation**

More sensitive means of detecting fuel failure may be required, but a requirement also to locate failed fuel without opening up the reactor is unlikely. NED should take the initiative and develop an improved failed-fuel sensor.

Specification for Power and Self-Operated Valves

F-7 **Finding**

Valves of these types have been a source of operating problems in both PWR's and BWR's. In the BWR system the main problems have been the safety relief valves and the main steam isolation valves, with considerable trouble from other types of valves.

Little is known about the regulatory requirement which Nuclear REgulatory Commission is considering in this area, but it seems likely to be one which will put specific functional requirements on the valves and require a demonstration of their conformance. Such requirements will probably raise the cost of valves and may require more in-plant surveillance testing of them.

R-7 **Recommendation**

Nuclear Regulatory Commission functional specifications for power and self-operated valves should be expected. NED should establish such specifications for itself, and initiate or intensify programs with valve manufacturers to assure that the specifications are met.

Protection Against Sabotage

F-8 Finding

European thinking in this area extends to the point of trying to thwart a saboteur who has already gained entry to the plant, who is carrying explosives, and whose objective is to cause a nuclear accident (an uncooled core).

U.S. requirements have not progressed this far, and may not for some time, but it seems clear that some thinking of this type will become regulatory requirement in the next few years.

R-8 Recommendation

NED should review the equipment and wiring layouts in STRIDE with a view to making them adaptable to compliance with anti-sabotage requirements of the type currently being adopted in German and Swiss projects.

Reduction of Occupational Exposure Limits

F-9 Finding

The philosophy acronymed ALAP states that radiation exposures should be kept "as low as practicable," and from time to time Nuclear Regulatory Commission has tentatively advanced the notion of hardening up ALAP by stating numerical "guidelines," which would be lower than the regulatory limits.

R-9 Recommendation

NED should re-examine its shielding criteria, source term estimates and shielding calculation methods, and should make any changes in these items necessary to ensure compliance with future occupational ALAP numerical guidelines practical. This recommendation applies particularly to STRIDE.

N-2 Safety Logic

F-10 Finding

Current Nuclear Regulatory Commission requirements provide, in effect, that for any single accident (except massive vessel rupture) which might result in an uncooled core, two emergency cooling systems must be available, either of which could by itself cool the (shutdown) core, and both of which have considerable internal redundance. Thus, if N = number of emergency cooling systems available, it may be said that the safety logic is N-1, since that is the number of safety system failures which can be tolerated should an event occur requiring that safety systems respond. German and Swiss authorities require, however, that in effect there be three backup systems, or that the safety logic be N-2. The argument runs that one backup system could at any time be out of action, because of repair work or surveillance testing, that a second could fail to work because of an unknown defect, and that the third would then be available if needed.

R-10 Recommendation

NED should make its own study of the possible need for an N-2 safety logic, using actual plant experience as a basic input to the study.

Removable Reactor Internals

F-11 Finding

While no specific Nuclear Regulatory Commission intention to issue a requirement on this subject can be cited, there have been problems with jet pump installations in the Quad Cities plant, and many of the parts in question are made of 304 SS and will be subject in BWR/6 to fast neutron fluences over a 30-40 year period which may significantly degrade their structural properties. Anticipating Nuclear Regulatory Commission requirements for removability seems a prudent engineering precaution.

R-11 Recommendation

NED should develop and exhaustively test a reactor design in which the internals are removable, with the objective of using this design in later members of the BWR/6 series, if possible, but in the BWR/6 successor in any event.

Core Catcher

F-12 Finding

The judgment of NED and of the Fast Breeder Reactor Department is that the probability of a core melt event is negligibly small, that the consequences of such an accident to the public have not been realistically analyzed, but rather have been sensationalized without engineering basis, and that a core catcher which would be completely effective under the extreme conditions postulated would be very difficult to develop, expensive to provide and infeasible to demonstrate. The validity of these judgments will, it is felt, eventually be recognized by the regulatory authorities and by ERDA, but probably not without some effort on the part of NED.

R-12 Recommendation

While the probability that Nuclear Regulatory Commission will establish a core catcher requirement for LWRs is small, NED should carry on a small study aimed at enabling NED to respond soundly to the Nuclear Regulatory Commission should the latter bring up the subject.

SCOPE AND STANDARDIZATION

SPECIFIC FINDINGS AND RECOMMENDATIONS

Scope

F-1 **Finding**

a) Since 1971 General Electric has been adding back significant scope in the form of PGCC/Nuclenet and STRIDE in such a manner as to improve standardization.

b) The Reactor Scope as offered is the highest source of unavailability Contributions by the balance of plant areas are small.

c) Reactor Equipment Limitations represent most causes of derating and will continue to in future years, unless significant equipment modifications are made.

d) Operator error is a minor cause of unavailability. Good utility management of operations is much more important in avoiding unavailability.

e) As a first approximation, for a nominal 1000 MW plant:
at percentage point in capability factor is equal to
3¢/MBTI in fuel costs, which is equal to
approximately $18 million in capital cost. (1980 dollars)

R-1 **Recommendations**

a) Concentrate Technical Resources on improving the plants both in operation and in the backlog.

b) In considering the design of a new product, optimization must be based on total system economics rather than NSSS or nuclear fuel cycle costs alone. This might involve a significant change in Market Strategy.

c) A study should be made of the Scope of Supply considering interface with others (such as A/E's) as well as options now supplied. The intent should be the determination of the most reasonable scope boundaries and options provided by NED based on Availability/Reliability, Sources of Engineering Knowledge, Difficulty in A/E Quality, etc.

Standardization

F-2 Finding

a) There is a strong correlation of manpower with number of customers.

b) BWR/6 fuel uses 25 different rod designs for six different initial core standard bundles and three different standard reload bundles.

c) Although BWRSD claims that the systems for accomplishing standardization are in place and working, a Quality Assurance audit indicated that:

- The existing BWR Standardization Program has not adequately penetrated to the level of detailed hardware designs.
- Overall coordination and direction of the BWR Standardization Program is lacking.
- The character of the BWR business requires a multiplicity of requirements that restrict the degree of standardization.
- A more systematic basis to assure that the BWR/6 designs conform to applicable licensing requirements is needed.

d) Industry is ready for major standardization efforts.

R-2 **Recommendations**

a) A "building block" concept of BWR standardization be implemented on a priority basis.

b) Initiate and execute an Aggressive BWR/6 Hardware Standardization program encompassing GE and Vendor Supplied Hardware.

c) The Standardization effort must be coordinated within the entire Division, including the International Operation. The differences of opinions between the Product Quality Operation & BWRSD must be resolved.

d) Proceed with a Product Offering Study to decide on the optimum size standardized plant and scope of offering and to recommend the plan for implementation.

Statistical Significance of Availability Goals

F-3 **Finding**

a) The immediate future regarding availability and capacity will be worse than historically seen because of pipe cracks, upgrading of some equipment, etc.

b) PWR's on average are generally even with BWR's in availability and W plants are ahead of BWR's in capacity factor.

c) BWR's are clearly ahead of the CE & B&W's as regards availability and capacity factor.

d) Most of the large PWR's (< 800 MW's) have been derated by AEC. This reduces their duty and improves their availability.

R-3 Recommendations

a) NED needs the ability to backfit equipment of existing plants and those coming on line between now and 1980.

(1) To give this proper emphasis, some change in organization should be considered such that the design engineers become more familiar with operating problems.

(2) This problem should be studied in conjunction with Service Activities described further on.

b) Reexamine statements of goals and their feasibility.

(1) NED Availability Goal.

(2) 2-3% Availability Advantage to overcome any Balance of Plant and Fuel Evaluation deficits.

Services

F-4 Finding

a) 3-4 availability improvements were made by better services during 1974.

d) NED service activities are presently organized on a lower efforts basis than other Power Generation Group Service Organizations.

c) Utility management of operating plants may badly need improvement regarding operation and maintenance because:

(1) Nuclear is more complex than traditional fossil plants

(2) More planning is needed.

(3) There are not enough qualified people.

R-4 Recommendations

a) It is strongly felt that the most meaningful action that can be taken now is in the BWR Services area. In recognition of potentially decreasing availability of existing operating plants, an improvement in the NED service programs approach should be made.

b) Programs being put in place or now in place, such as Advanced Maintenance Planning Service (AMPS II) and Refueling-Maintenance Outage Service, should be accentuated.

c) New programs should be put into place to increase availability. Programs that appear to have a large potential for availability improvement include:

(1) Extended Scope Start-up Service

(2) A Full-time GE Representative on Site -- an "availability engineer", so to speak.

(3) Expanded Outage Services

(4) Engineering Field Change Services on all Backfits (manpower estimates, plant conditions, procedures, tools, along with the equipment being supplied).

(5) Outage Control Center

(6) Better tools

(7) Maintenance Training Mockups

(8) Special Programs for

- Feedwater Sparger Replacement
- Relief Valve Repair

- MSIV Modifications
- Piping Change Out

(9) Systems to track component failure and redesign accordingly.

d) Strengthening the service engineering effort with dedicated engineers should help increase the customer's sense of participation in the engineering effort.

e) It is recognized that such efforts will require increased people, engineering, field and spare parts support. Such support should be provided with a higher sense of urgency and level of funding than is presently taking place. Special emphasis should be made in strengthening the organization (BWR Services) and its interfaces within and without NED to carry out this mission, including the addition of dedicated plant engineering people to BWR Services as well as the Spare Parts Operation.

Services offered overseas may open up unique opportunities and help preserve the BWR.

One of the most important concepts should be the singular responsibility for BWR Services focused within the entire Division ... Domestic as well as International ... one organization should be responsible for Service and the duties should not be split.

In the long run, Organizational Plans should be drawn up that elevate BWR Services to a Department-level function within NED.

APPENDIX A

DESCRIPTION AND EVOLUTION OF THE BOILING WATER REACTOR

A. BWR PRODUCT LINE EVOLUTION

B. BWR QUALITY STRATEGY/PRODUCT QUALITY

C. BWR DESCRIPTION AND TECHNICAL EVOLUTION

D. BWR AVAILABILITY AND CAPABILITY

E. BWR-PWR COMPARISONS

LIST OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ADU	Ammonium Diuranate
A/E	Architect Engineer
AEG	Allgemeine Elektricitats-Gesellschaft
ALAP	As Low As Practicable
ATLAS	Advanced Test Loop and Simulator
ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CBIN	CBI Nuclear - Chicago Bridge and Iron Nuclear
C&I	Control and Instrumentation
ECA	Engineering Change Authorizations
ECCS	Emergency Core Cooling System
EOC	End of Cycle
ERDA	Energy Research and Development Administration
FABLE	Computer Code Name
GEBLA	Computer Code Name
GEBS	Computer Code Name (Three-Dimensional BWR Steady State Core Analysis)
GETAB	General Electric Thermal Analysis Basis
HPCI	High Pressure Core Injection
I&SE	Installation and Service Engineering
IOMR	Interim Operating Management Recommendations
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MAPLHGR	Maximum Average Planar Linear Heat Generator Rate
MCPR	Minimum Critical Power Ratio
MLHGR	Maximum Linear Heat Generator Rate
MO_2	Mixed Plutonium - Uranium Oxide
MSIV	Main Steam Isolation Valve
MWD/T	Megawatt Days/Ton
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
PCI	Pellet-Clad Interaction
PCIOMR	Pre-Conditioning Interim Operating Management Recommendation
P&QAO	Product and Quality Assurance Operation
PRT	Prompt Relief Trip
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
REDY	Computer Code Name (Zero-Dimensional Core Model for Systems Transient Analysis)
REVAB	Relief Valve Augmented Bypass
RML	Radioactive Materials Laboratory
RPV	Reactor Pressure Vessel
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
STRIDE	Standard Reactor Island Design
TIP	Traversing In-Core Probe

DESCRIPTION AND EVOLUTION OF THE BOILING WATER REACTOR

A. BWR PRODUCT LINE EVOLUTION

The major steps in evolution of the Boiling Water Reactor product line are summarized in Table 1.

BWR/1

The first General Electric commercial BWR was Dresden-1 which started operation in 1960. Commitments included warranties on fuel and operating costs, as well as the traditional warranties on plant electrical output and heat rate. The Dresden project was undertaken in 1955, the plant went into commercial operation in 1960, and was uprated to its current power level of 210 MW in 1961. The operational warranties were demonstrated and accepted by the customer (Commonwealth Edison) by 1964.

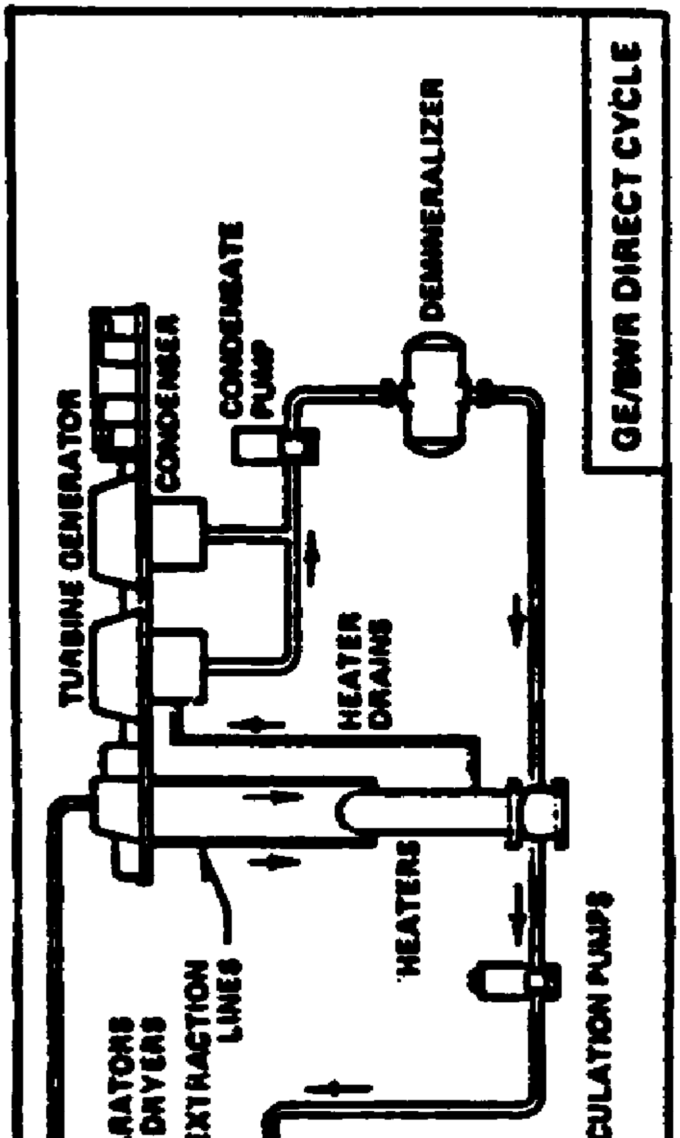
Following the Dresden-1 reactor, other plants of similar design were constructed and successfully operated during the 1960's in Italy (Garigliano 1964 - 160 MW), Germany (KRB 1967 - 250 MW), and India (Tarapur 1969 - 2 reactors each 210 MW). Also during this time two smaller prototype reactors with advanced design characteristics were constructed: Big Rock Point (1963 - 75 MW) and Humboldt Bay (1963 - 70 MW). These units included prototypical features incorporated into later designs of commercial size.

All of the foregoing units have been classified as BWR/1 units, although they were not all of the same design. The smaller reactors were single cycle which used only steam generated in the reactor core (see Figure 1a). The larger units of this period used dual cycle boiling water reactor systems (Figure 1b) in which steam for the turbine was generated from two sources: a) directly in the reactor core and b) in a heat exchanger heated by water from the reactor core.

TABLE 1

BWR PRODUCT LINE EVOLUTION

Reactor	MW Output	Year of Order	Year of Operation	Features
BWR/1				
Dresden-1	210	1955	1960	Dual cycle
KRB	250			Internal steam separation
BWR/2				
Oyster Creek	640	1963	1969	Commercial size Single cycle - 5 loop Pump flow control
BWR/3				
Dresden-2	809	1965	1972	Jet pump - 2 loop Improved ECCS: Core spray and flood
BWR/4				
Browns Ferry 2	1118	1966	1973	Increased power density 20%
BWR/5				
Zimmer	807	1969	1978 Est.	Improved ECCS - faster flooding
Other plants up to	1135			Valve flow control
BWR/6				
Grand Gulf	1290	1971	1980 Est.	Increased power density Reduced fuel thermal duty Improved ECCS



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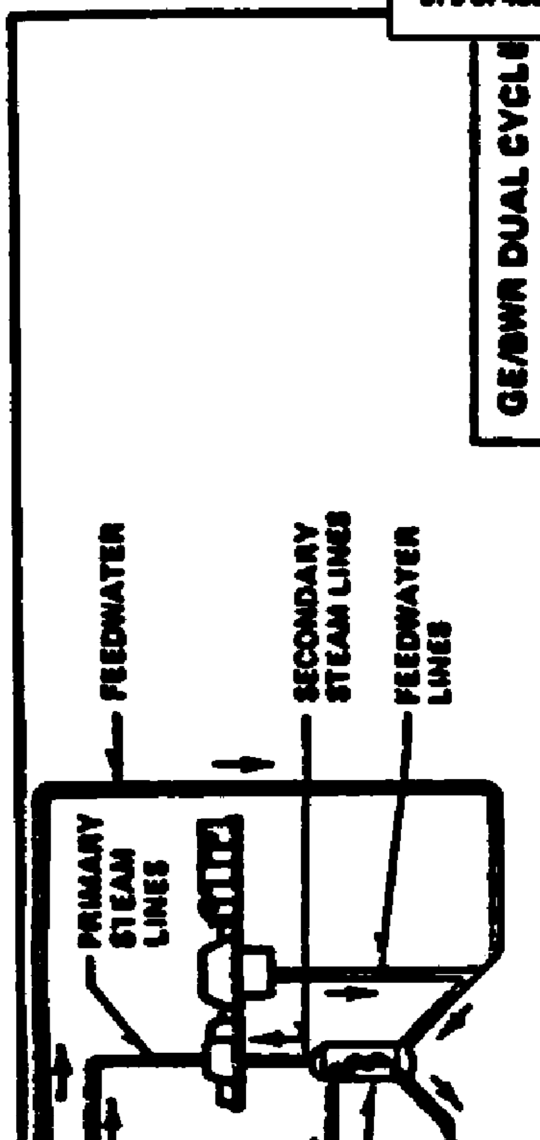


1b

FIGURE 1

BWR/2

Commercial units in the 500-600 MW size range were offered in 1962-1963. NED received orders for the Oyster Creek and Nine Mile Point reactors. These BWR/2 reactors were based on improved single cycle designs aimed at reduction in plant cost.

BWR/3

In 1965-1966 NED received orders for nine turnkey plants, six domestic and three overseas. Size had been increased to about 800 MW. The design for these orders was the BWR/3 class which introduced jet pumps for further reduction in plant cost.

BWR/4

NED introduced the BWR/4 class of reactors in late 1966. In this reactor class sizes were increased to about 1100 MW. The core power density was increased 20% which resulted in significant decrease in reactor plant cost per MW produced.

BWR/5

The BWR/5 class introduced in 1969 incorporated new features for emergency core cooling in response to a developing trend of increasing regulatory requirements. Valve operated flow control is another new feature of this class. Reactor rating remained about the same.

BWR/6

NED introduced the BWR/6 and Mark III containment in late 1971 - early 1972. The maximum rating was increased to 1290 MW. The BWR/6 has an increased power density, lower thermal fuel duty and more conservative Emergency Core Cooling System (ECCS) performance.

The growth in size and number of plants and the year of commercial operation are shown graphically in Figure 2.

YEAR OF COMMERCIAL OPERATION VS. NUMBER OF PLANTS IN OPERATION

AND GROSS MW SIZE

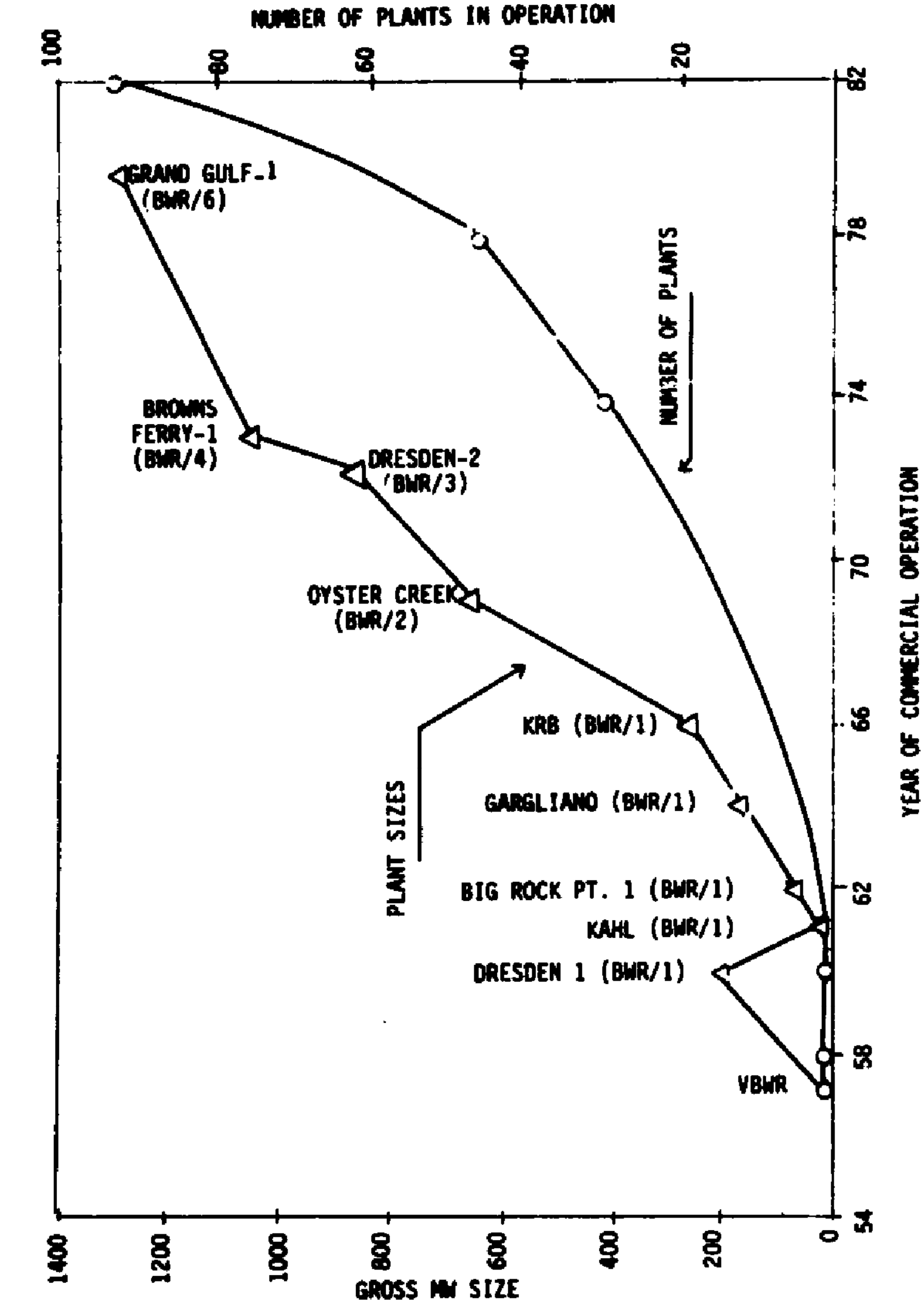


Figure 2

B. QUALITY STRATEGY/PRODUCT QUALITY

The Nuclear Energy Division introduced its Quality Strategy in 1974. Product Quality under this strategy is measured by the following criteria for the total BWR plant as perceived by the customer, recognizing that the NED-BWR supplied equipment is contributory to total plant performance.

- The plant must be available to furnish electrical power to the grid as required by the customer. Customer expectation of availability for mature plants is 75 to 85% or greater.
- The plant must be capable of producing full output during its available operating period. Customer expectation of the capacity factor is 70 to 80% or greater.
- The plant must be capable of responding to change in load required by the electrical system within accepted industry practices and automatically withstand anticipated transients within specified limits.
- The plant must be readily operable in accordance with operating procedures and regulations without excessively complicated procedures and limitations, and with minimum radiation exposure of operating personnel.
- The plant must be readily maintainable within stated time periods with minimum radiation exposure of maintenance personnel.
- The lifetime of NED supplied components must be consistent with stated lifetimes.
- The plant must be capable of meeting government regulations and accepted industry codes and standards for design, construction and operation.

- The BWR core and fuel must be capable of delivering the stated energy output without unplanned reduction of core thermal output or unplanned shutdowns for fuel replacement.

C. BWR DESCRIPTION AND TECHNICAL EVOLUTION

BWR Configuration

Functionally, the Boiling Water Reactor directly replaces the combustion heated steam generating boiler in the conventional electric generating plant. Plant configurations are shown in Figures 3 and 4. As shown in Figure 5, steam generated in the reactor is used directly in the turbine. The turbine thermodynamic cycle is basically the same as a conventional plant, but is modified for the lower steam pressure and temperature capability of the nuclear reactor core. The BWR steam conditions are 985 lbs./sq. in. and 543 F^{o} compared to conventional boiler steam conditions of 2400 psi and 1000 F^{o}. In the Pressurized Water Reactor plant, also shown in Figure 5, water heated in the nuclear reactor and kept liquified under high pressure is used to produce steam in a heat exchanger known as a steam generator. This steam is then used in a turbine thermodynamic cycle very similar to that of the BWR.

In passing through thc reactor, the water and steam become radioactive, but their radioactivity decays quickly after reactor shutdown. In the BWR, radioactive steam is piped directly to the turbine, which requires shielding of the turbine. Because most of the steam radioactivity dies out quickly after shutdown, it is not a significant factor in turbine maintenance. In the PWR, the steam generated in the heat exchanger is not radioactive, however, in case of leakage of radioactive reactor water through the heat exchanger tubes,

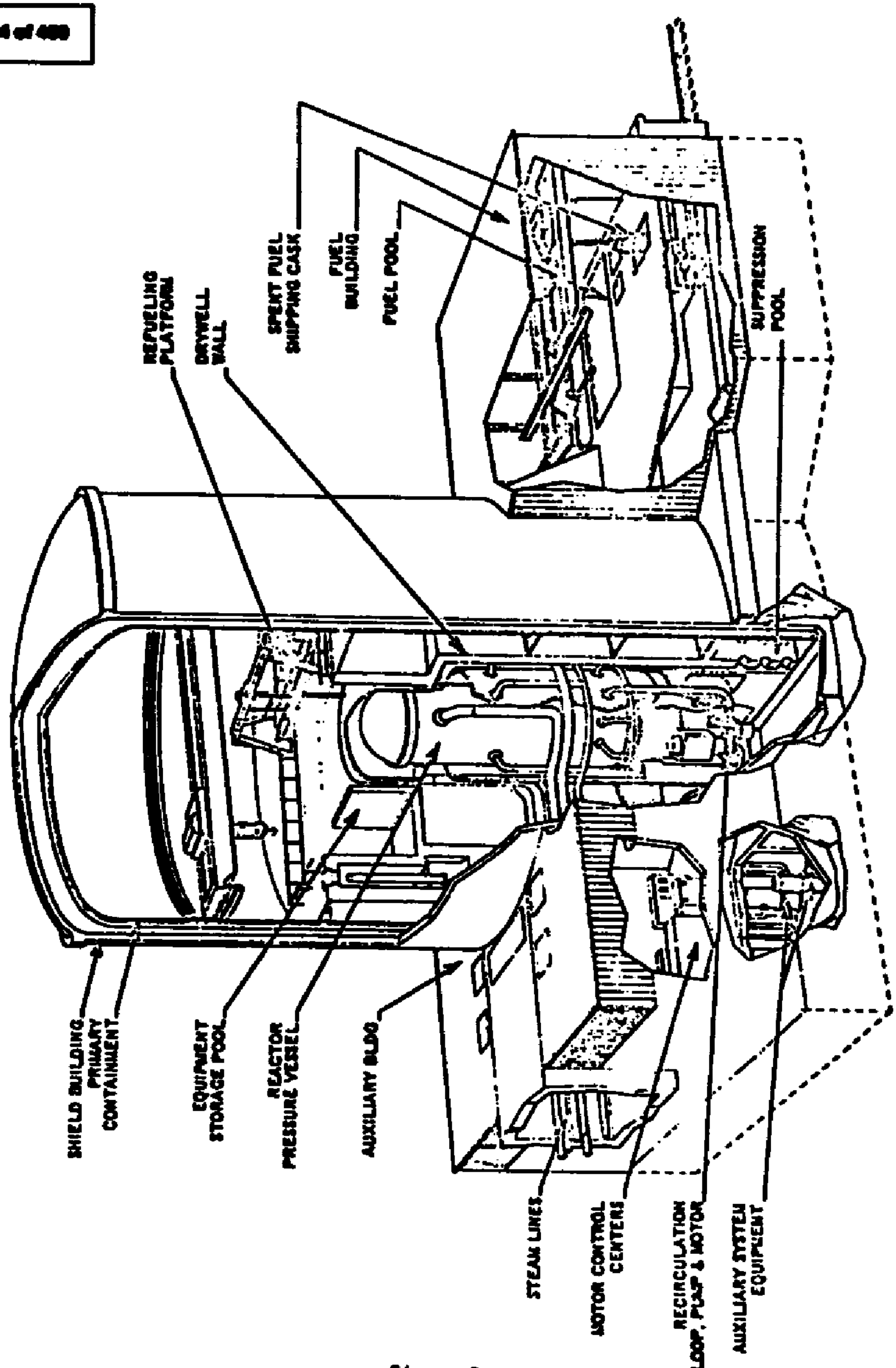
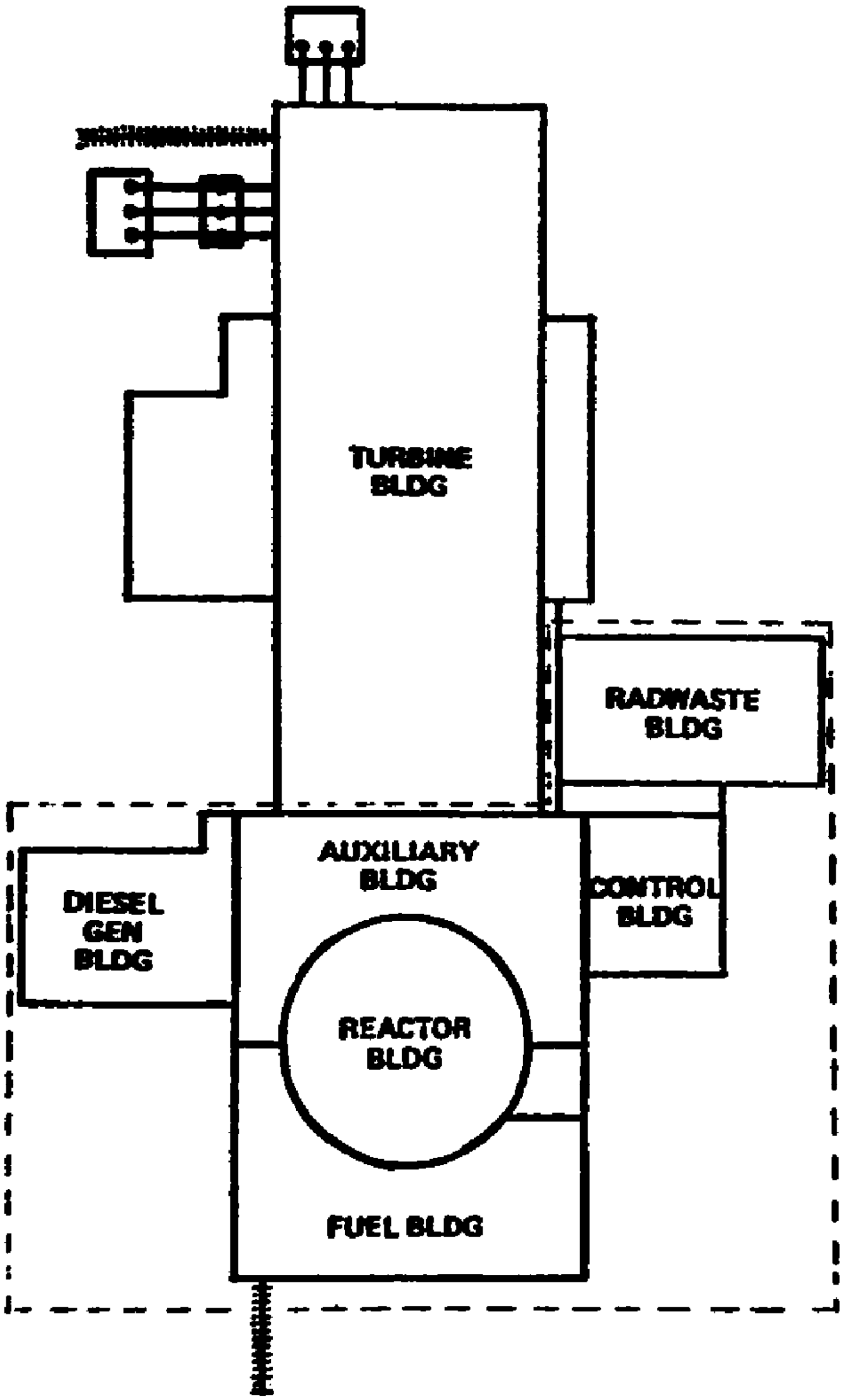


Figure 3

Mark III Configuration
Reactor, Fuel and Auxiliary Building

BWR PLANT LAYOUT



REACTOR ISLAND - - -

Reactor Building

Auxiliary Building

Fuel Building

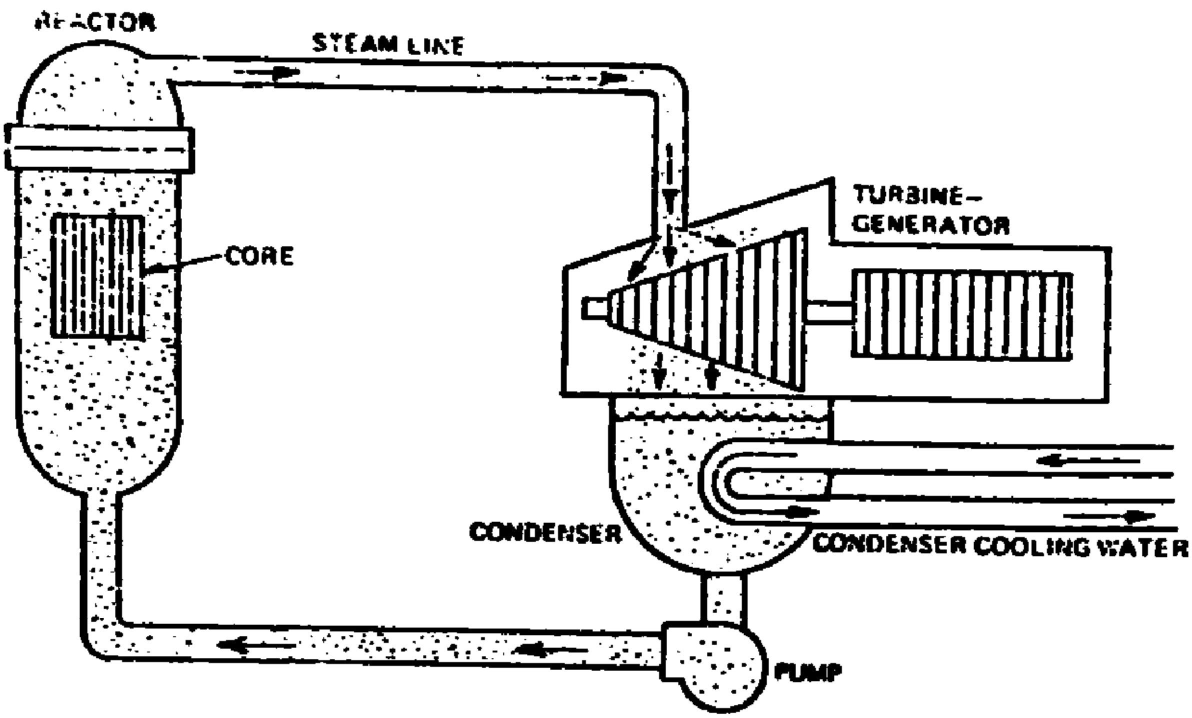
Control Building

Radwaste Building

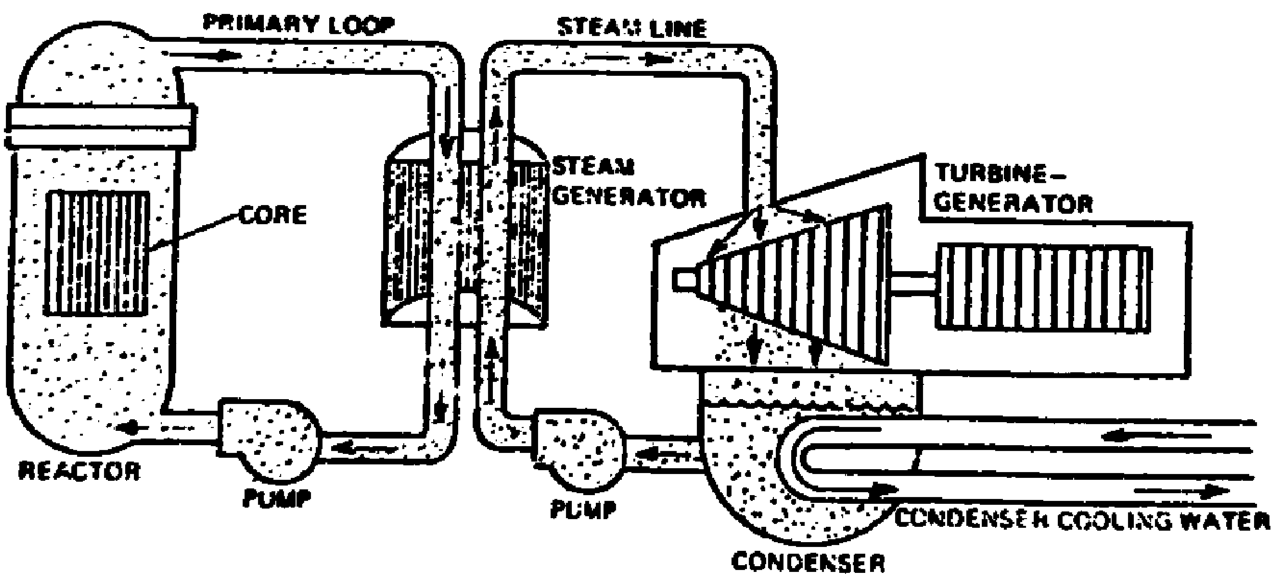
Diesel Generator Building

Figure 4

BWR-PWR COMPARISON



Direct Cycle BWR



Indirect Cycle PWR

Figure 5

operational problems occur, since the steam systems are not now designed to accommodate radioactive fluids.

A modern boiling water reactor is shown schematically in Figure 6. Steam is generated within the reactor core (1) and leaves the core in a mixture of steam and water. The steam is first separated from the water in steam separators (2), then passed through dryers (3) which remove the remaining moisture from the steam. Dry steam (less than 0.1% moisture) is piped to the turbine (4). The core recirculating water which is discharged from the steam separators is mixed with feedwater (5) (condensed steam from the turbine), and pumped through the reactor core by jet pumps (6), located in the annular region surrounding the core. Jet pumps have no moving parts and work on the principle of momentum exchange, i.e., high velocity driving flow from the external pumps accelerates the core recirculation flow within the jet pump.

This compact arrangement with the functions of core recirculation, steam separation and steam drying all performed within the reactor vessel, has evolved in steps from the earliest BWR's in which all of these functions were performed external to the vessel. This is shown in Figure 1b which is a schematic of the dual cycle boiling water reactor concept of Dresden-1. In this concept, steam generated in the reactor is separated in an external primary steam drum and piped directly to the high pressure (~ 1000 psi) inlet of the steam turbine. The reactor recirculating water is pumped through a secondary steam generator which delivers steam to the turbine at an intermediate pressure (~500 psi).

Although technical successful, the cost of this plant configuration was high and work was undertaken to reduce the cost through simplification and compaction of the reactor plant. At KRB, internal steam separation was introduced, eliminating the external steam drum which had included this function. The next

BWR SCHEMATIC

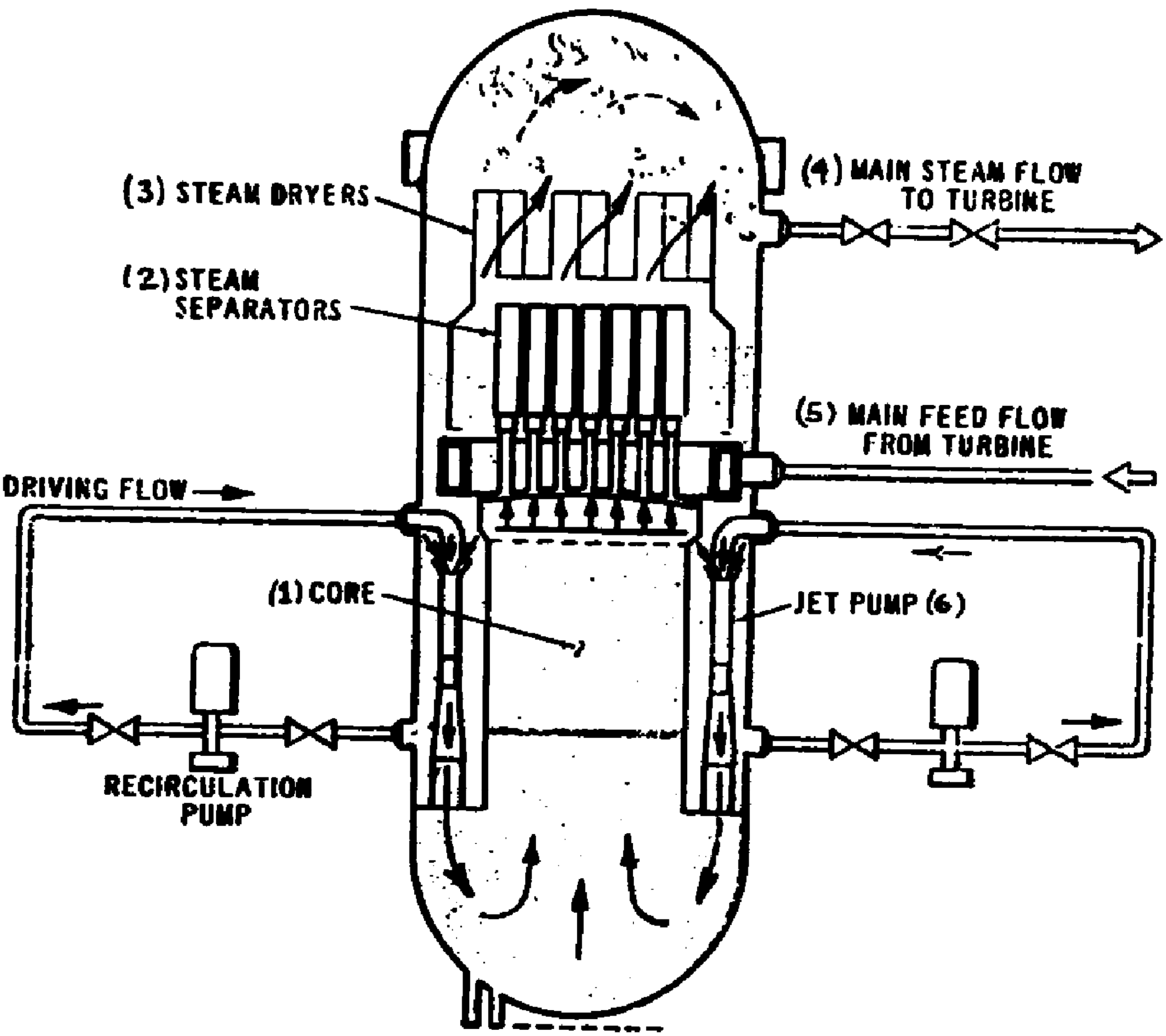


Figure 6

major step (BWR/2) was the adoption of the single cycle or direct cycle concept in which all of the steam is generated in the reactor and delivered to the turbine (Figure 1a). This eliminated the cost and complexity associated with the steam generator and the secondary turbine inlet and control valves. Further compaction was achieved in the BWR/3 class by incorporating jet pumps within the reactor vessel. This resulted in reduction in size of the external pumps and piping since the driving flow rate for the jet pumps is about 1/3 to 1/2 of the total core recirculation flow. The jet pump configuration also permitted incorporation of emergency core flooding provisions for the reactor core.

The basic configuration of the reactor assembly and its components has remained essentially the same since BWR/3 (see Table 1). The principal change in BWR/4 was to increase core power density about 20% to attain greater steam output with little change in equipment size. In the BWR/5 class, improved emergency core cooling systems were incorporated in response to increasing stringency of AEC regulatory requirements.

The changes in BWR/6 represent another significant step. An additional 20% increase in steam output was achieved in the same vessel by increasing core size and core power density. At the same time, the fuel design was changed to reduce the thermal duty on individual fuel rods. This change in fuel design is described in detail in later discussion. The reduced fuel thermal duty provided significant margin for emergency cooling requirements and is also expected to result in reduced fuel failures during normal operation.

The incentive for compaction has been, and continues to be, the reduction of capital cost to the customer, which is sensitive to the volume of the reactor and its recirculation and auxiliary systems. Another major factor in capital cost reduction has been the rapid increase in introduction of plants with higher ratings and lower $/KW costs due to size effects. This is summarized in Table 1 which shows a doubling in plant size in eight years - from 640 MW in 1963 to 1290 MW in 1971 (also see Figure 26).

BWR Containment

The compaction of the reactor and its systems has been accompanied by simplification of structures and compaction of the reactor containment systems. The reactor and directly connected systems of the BWR and PWR are enclosed within a containment system which is intended to contain all of the hot water and steam from those systems even in the event of a major pipe rupture.

A feature of containment unique to the BWR is associated with the direct cycle in which steam from the reactor leaves the containment in going to the turbine. Quick closing isolation valves, included in the steam lines, form part of the containment of the reactor systems. These isolation valves are closed automatically in event of steam line rupture outside of the containment. Extreme leak tightness is a regulatory requirement. The small amount of escaped steam will not result in significant exposure to radioactivity of personnel off the plant site.

The containments for most of the BWR/1 reactors were simply steel shells of sufficient volume and strength to contain the released fluids. For the Humboldt reactor, Pacific Gas and Electric and GE cooperated in the development of the pressure suppression containment concept in which the escaping hot fluids are directed into a pool of water to be condensed, thus reducing the pressure and volume of the containment. Starting with the BWR/2 class, all BWR's have been housed in pressure suppression containment. The various arrangements of this concept which have evolved are shown in Figure 7. In each case the reactor and its connected systems are housed in a primary containment or "dry well"[1]

PRESSURE SUPPRESSION CONTAINMENT

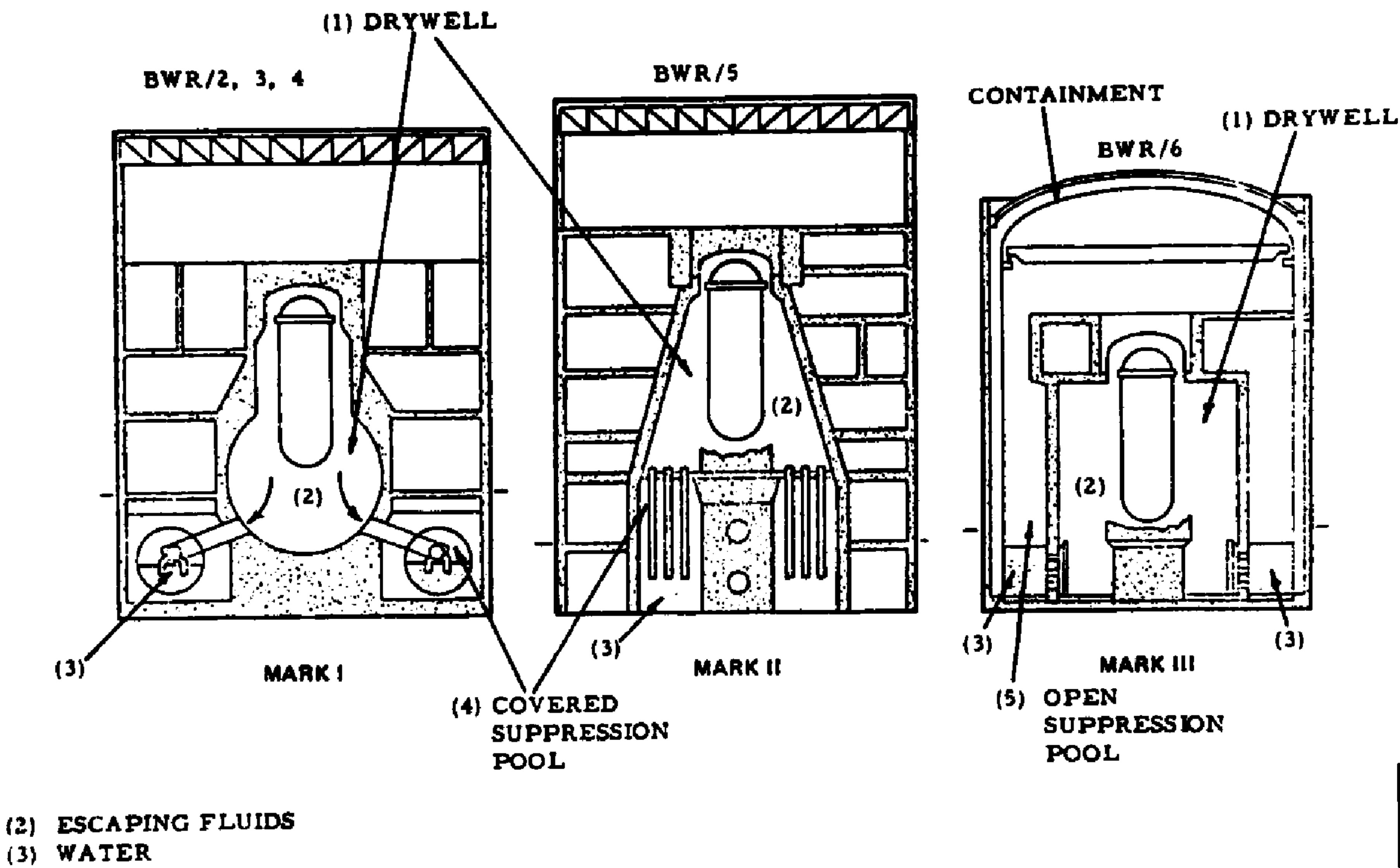


Figure 7
A-15

which is designed for the maximum transient pressure which may be encountered after a major pipe rupture. The escaping fluids[2] are ducted into a pool of standing water[3] where they are condensed. The noncondensable air and gases pass through the water. In the Mark I and II containment these gases are trapped above the water in the covered suppression chambers[4]. In Mark III these gases are allowed to escape into the containment atmosphere[5].

The objective of these evolving designs has been to reduce overall size and construction complexity of containment, which results in reduced construction cost to the utility. This is illustrated schematically in Figure 7. The construction complexity has been reduced in each succeeding generation. In Mark II the entire configuration of the dry well, vent pipes and suppression pool were simplified to provide shapes more readily adaptable to concrete construction. In Mark III further simplification was achieved and vent pipes eliminated by providing an annular suppression pool surrounding the bottom of the dry well.

The Mark I concept was introduced with BWR/2 and was used in all BWR/3 and 20" of the BWR/4 plants. The Mark II has been used in both BWR/4 and BWR/5 plants. The Mark III was introduced with BWR/6 and is used exclusively with that class.

The BWR compaction trend in reactor primary systems and containment is illustrated in Figure 8, which shows the variation in pounds of steel in the plant and containment for successive BWR plants. The change from Dresden-1 to KRB was due to elimination of the external steam drum through the use of internal steam separation and use of smaller secondary steam generators. The reduction in BWR/2 plants was achieved by eliminating the steam generators in going to the single cycle. The improvement in BWR/4 was due to increased core power density and use of jet pumps. The changes in containment were described above.

BWR COMPACTION HISTORY

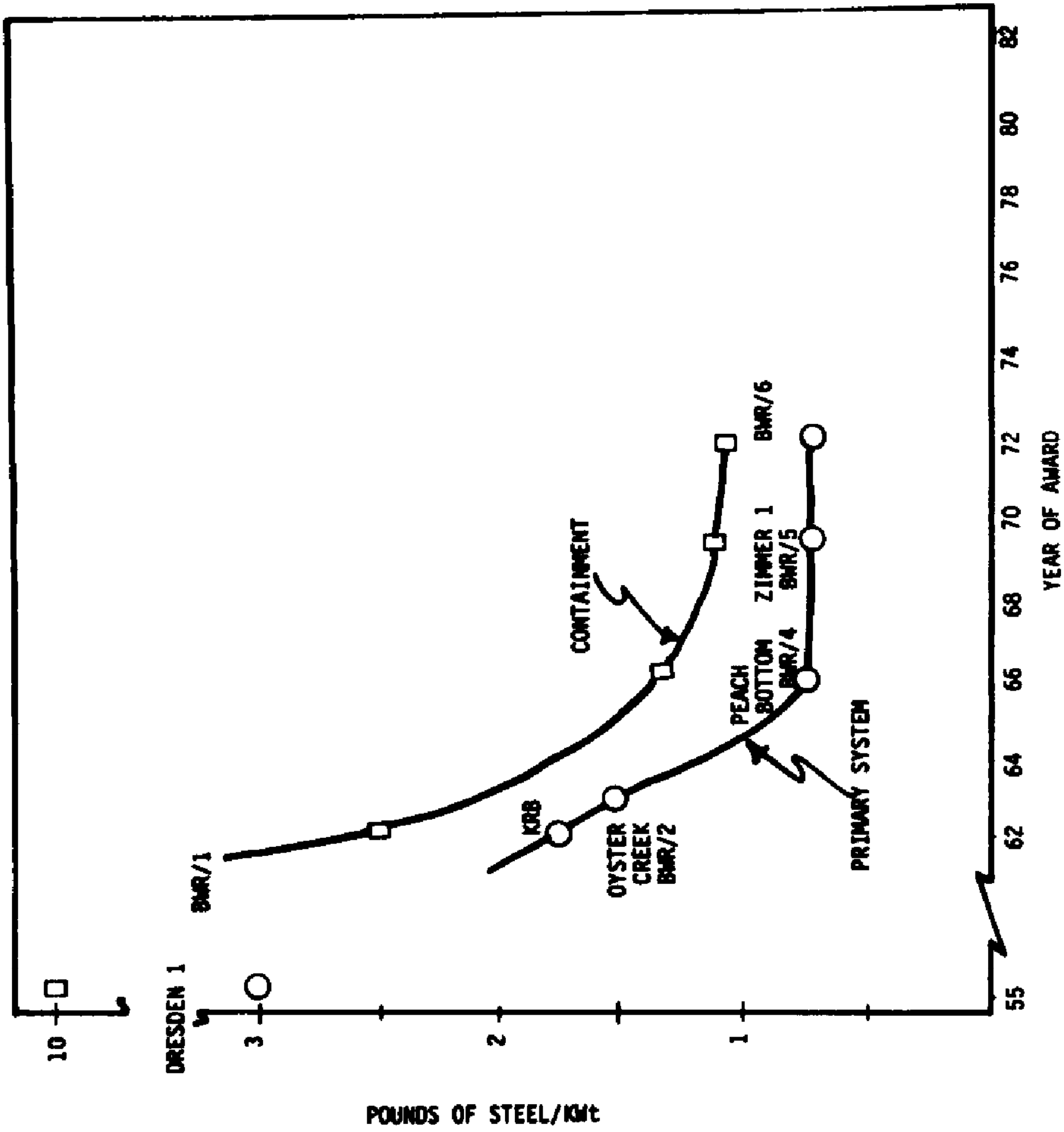


Figure 8

Plant Cost Trends

The need for plant simplification and compaction is highlighted by the trend of increasing costs in other factors such as plant construction, escalation, interest during construction and additional requirements to meet safety and environmental regulations. This trend is illustrated in Figure 9, which shows that these costs for a 1000 MW plant were estimated to increase by more than a factor of 5 over the period of 1967 to 1974. In the same period, time-related costs, escalation and interest during construction, increased from about 20% to nearly 50 of the total. These time-related costs, which can be viewed as multipliers on basic plant costs, emphasize the importance of reducing the basic costs to help offset some of the increase in total cost.

Unit Size Standardization

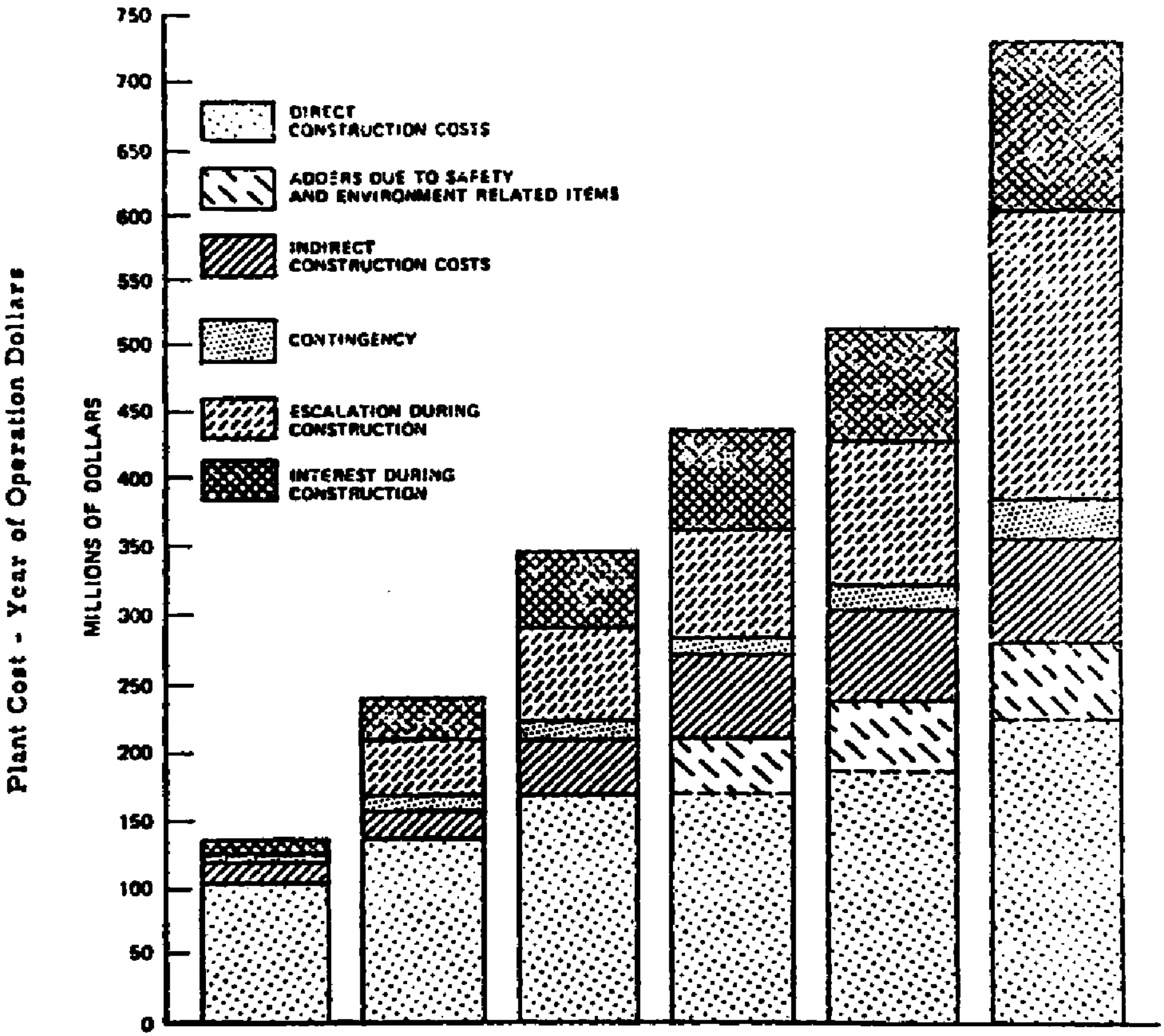
A summary of unit sizes for the various classes of BWR is shown in Figure 10. It should be noted that starting with BWR/4 the sizes have been standardized with most of the units in vessels of 218, 238 and 251 inches diameter. The number of units of each size is indicated by the bar. This chart also graphically illustrates the steps of increased thermal output from the same vessel size. In BWR/3 the 251 vessel is used to generate 2500 MWt; in BWR/4 and 5, 3300 MWt, and in BWR/6, 3800 MWt*

Reactor Assembly

An overall view of the BWR/6 reactor assembly is shown in Figure 11. The reactor vessel is 23 ft. in diameter and 73.5 ft. high. The reactor core consists of a multiplicity of fuel assemblies, each about 14 feet long[15]. Four fuel assemblies surrounding a cruciform shaped control blade[16] comprise a standard module, which is repeated to form the total core. Each control blade

* NRC limit - Capability is 4146 MWt

COMPARISON OF NUCLEAR PLANT COST ESTIMATES
[Total investment cost for 1000-MW(e(units]



Start of Project:	**March 1967**	**June 1969**	**Jan. 1971**	**Jan. 1971**	**Jan. 1973**	**Mid-1974**
Commercial Operation :	**Late 1972**	**Mid-1975**	**Jan. 1978**	**Jan. 1978**	**Jan. 1981**	**Jan. 1983**

Figure 9

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BASIC STANDARDIZATION HISTORY

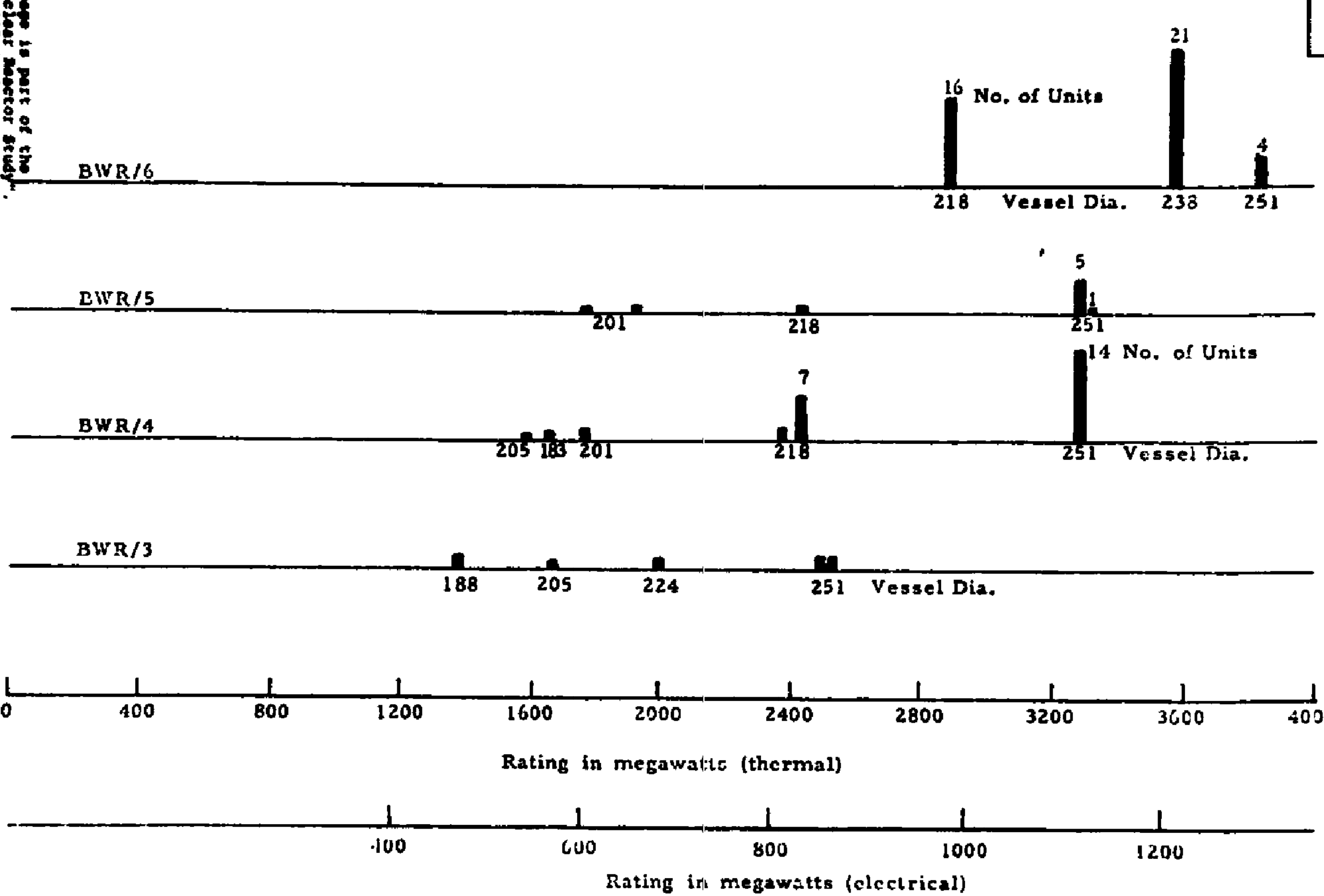

Figure 10
A-20

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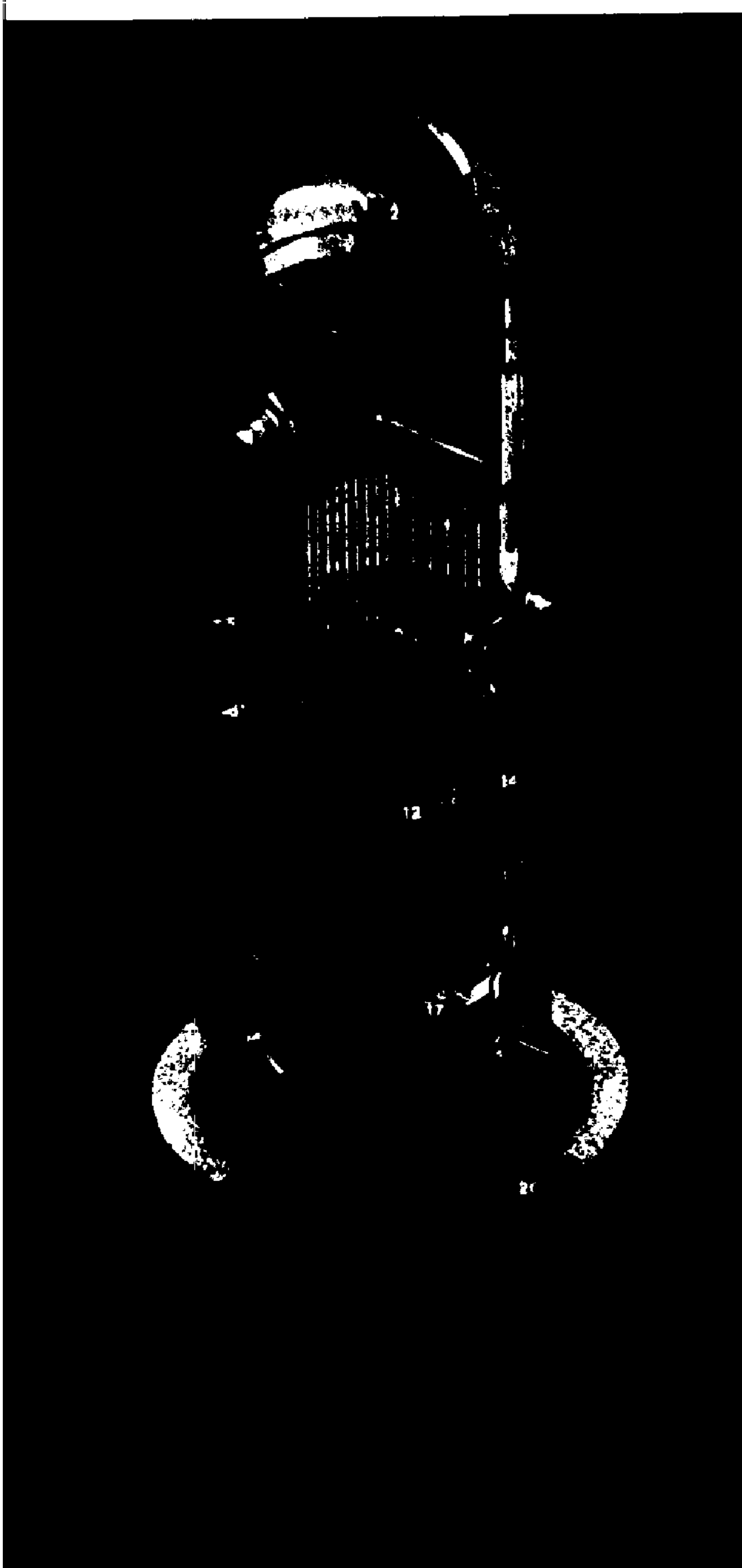


BWR/6

REACTOR ASSEMBLY

1. VENT AND HEAD SPRAY
2. STEAM DRYER LIFTING LUG
3. STEAM DRYER ASSEMBLY
4. STEAM OUTLET
5. CORE SPRAY INLET
6. STEAM SEPARATOR ASSEMBLY
7. FEEDWATER INLET
8. FEEDWATER SPARGER
9. LOW PRESSURE COOLANT INJECTION INLET
10. CORE SPRAY LINE
11. CORE SPRAY SPARGER
12. TOP GUIDE
13. JET PUMP ASSEMBLY
14. CORE SHROUD
15. FUEL ASSEMBLIES
16. CONTROL BLADE
17. CORE PLATE
18. JET PUMP / RECIRCULATION WATER INLET
19. RECIRCULATION WATER OUTLET
20. VESSEL SUPPORT SKIRT
21. SHIELD WALL
22. CONTROL ROD DRIVES
23. CONTROL ROD DRIVE HYDRAULIC LINES
24. IN-CORE FLUX MONITOR

GENERAL ELECTRIC

This page is part of the 1975 "Nuclear Reactor Study". See the 1987 "Update Report" for current status.

Figure 11

is positioned by a hydraulic mechanism[22] mounted in a tube protruding from the bottom of the reactor vessel. Each module of four assemblies is supported from the inner bottom of the vessel by a tube which also serves as the guide for the control blade when it is extracted from the core. All of these pieces are of a standard design within a group or class of reactors.

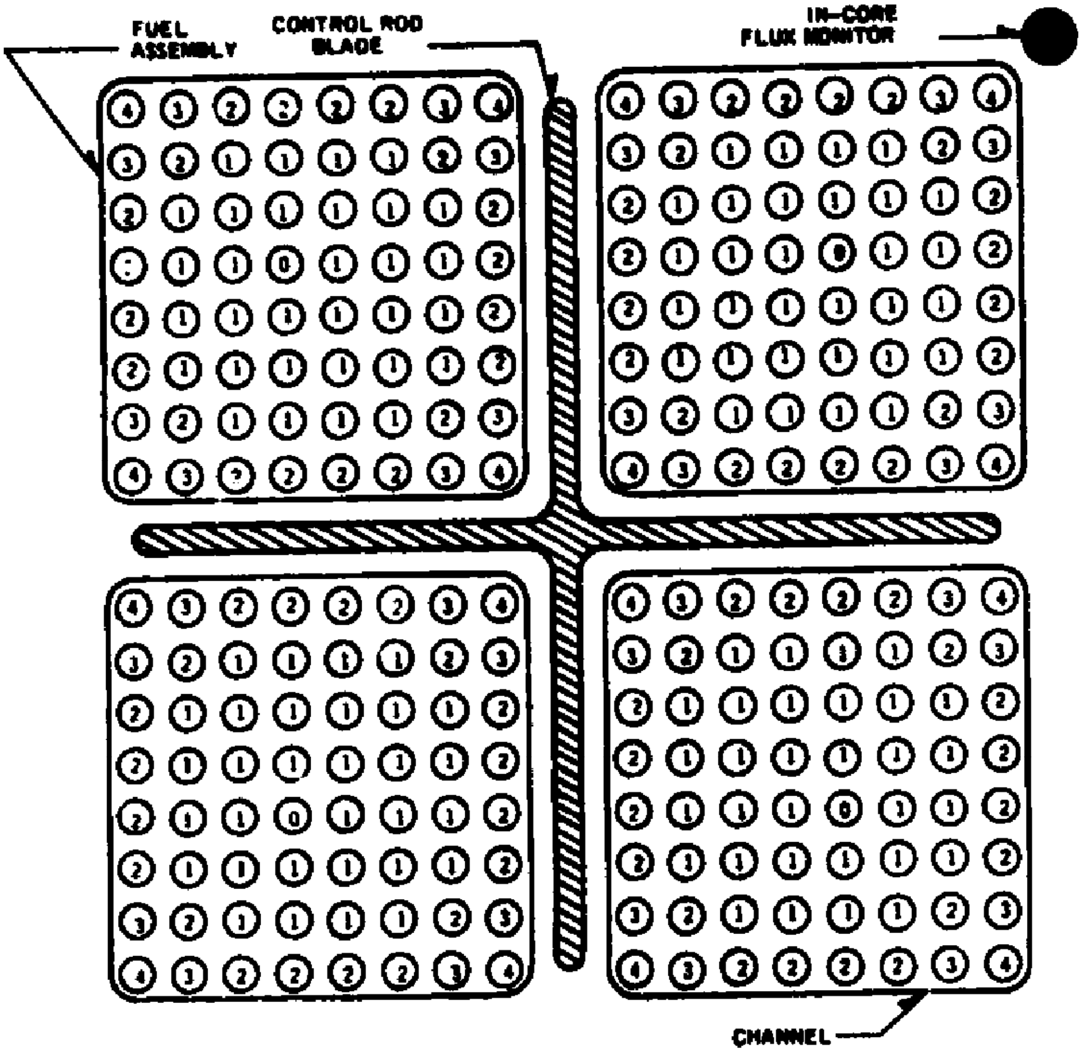
A multiplicity of steam separators[6] is shown above the core. In these units the steam is separated from core recirculating water by centrifugal action. The water is discharged downward to be mixed with incoming feedwater[7] and recirculated through the core by jet pumps[13] located in the annulus surrounding the core. The steam is passed through "dryers"[3] which remove moisture particles by impingement on the surface of the dryer. The jet pumps and high performance steam separators have been developed by NED for the BWR.

Reactor Core and Fuel Design

Figure 12 is a cross sectional view of a typical core module consisting of four fuel assemblies surrounding a control blade. A fuel assembly[1] consists of an array of fuel rods enclosed in a channel[2] which performs the dual function of channeling flow around the fuel rods and providing a guiding surface for the control blade. Each fuel rod consists of a tube of Zircaloy 2 filled with short cylindrical pellets of uranium oxide (UO_2) and sealed. The channel is made of Zircaloy 4. Zirconium, which has the unique characteristic of low neutron absorption, is the main constituent of the zircaloys which have been developed to provide corrosion resistance in hot water and steam. Further detail of the fuel assembly is given in Figure 13.

The overall nuclear characteristics of the core are determined by the relative amounts of uranium and water in the core. Since water is a strong neutron moderator, the core control characteristics and its nuclear efficiency vary

CORE LATTICE



NOTE 1, 2, 3, & 4 INDICATE LOCATION OF FOUR DIFFERENT ROD ENRICHMENTS IN FUEL ASSEMBLY 0 INDICATES A WATER ROD.

Figure 12

BWR/6 FUEL ASSEMBLIES & CONTROL ROD MODULE

1.TOP FUEL GUIDE
2.CHANNEL FASTENER
3.UPPER TIE PLATE
4.EXPANSION SPRING
5.LOCKING TAB
6.CHANNEL
7.CONTROL ROD
8.FUEL ROD
9.SPACER
10.CORE PLATE ASSEMBLY
11.LOWER TIE PLATE
12.FUEL SUPPORT PIECE
13.FUEL PELLETS
14.END PLUG
15.CHANNEL SPACER
16.PLENUM SPRING

GENERAL ELECTRIC

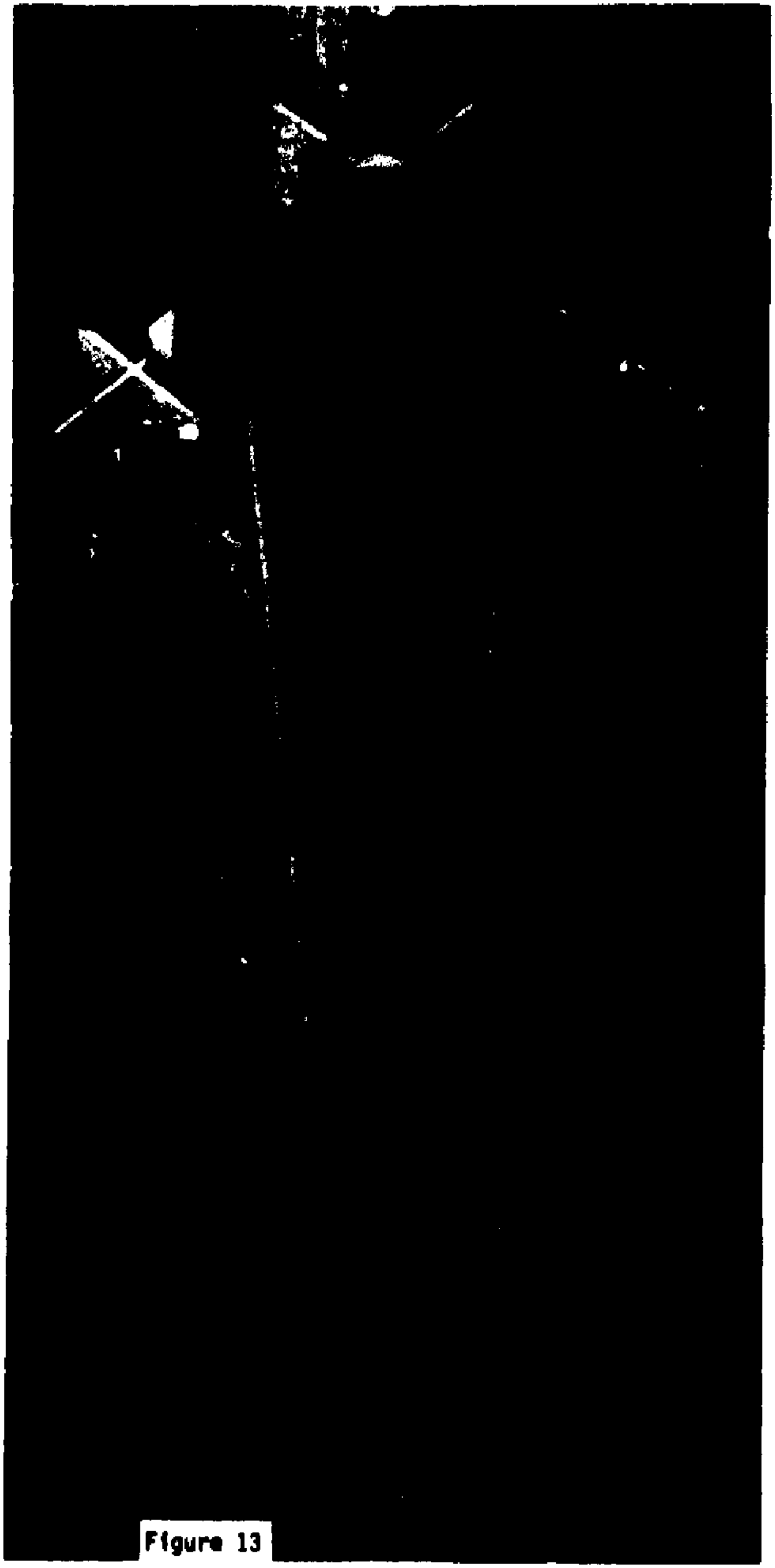


Figure 13

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for current status.

with the water to fuel ratio. This makes the basic water to fuel ratio a key parameter to core design of any water reactor. In the boiling water reactor, the local displacement of water moderator by steam, which acts like a nuclear void, creates the unique reactor physics characteristics of the BWR. The presence of steam voids throughout the BWR core, and their variation with core power, account for the main difference between the BWR core and the PWR core in which the water within the core is maintained in liquid form.

Fuel used in the BWR is a mixture of uranium isotopes in the approximate ratio of 3% U-235, 97% U-238. The total amount of the fissionable isotope U-235 in the fuel determines the amount of energy which can be extracted from the fuel. The degree of enrichment of the uranium (percent U-235) is selected to provide the required energy for a planned period of operation.

Basic nuclear control of the reactor is achieved by positioning of the cruciform control blades (Figure 12, item 3). The control material within the blade is boron, which is a strong neutron absorber. The boron in the form of boron carbide powder is compacted into individual stainless steel tubes which are grouped into the wings of the cruciform. There are 177 control blades in the reactor of a 1220 MW plant.

The control blades perform a number of control functions. They are used for normal reactor startup and shutdown and also for fast reactor shutdown (SCRA,) when it is needed. During operation they are positioned throughout the core to minimize the inherent difference in heat generation between the central and peripheral regions of the core. The blades are gradually withdrawn to compensate for fuel depletion which occurs slowly over the approximately annual period between refueling, when 1/4 of the fuel is replaced.

A supplementary method of nuclear control is also provided to compensate for fuel burnup, in order to reduce the multiplicity of requirements on the blade control system. This supplementary control makes use of a unique property of the element gadolinium which is normally a strong neutron absorber, but which is transformed to a weak absorber after absorbing a neutron. This element in oxide form, is mixed with the uranium oxide in several rods in each bundle. As fuel is consumed by the nuclear reaction, the gadolinium nuclei are transformed from strong to weak absorbers, which gradually reduces the amount of strong absorber approximately in proportion to the amount of depleted fuel.

Since the rates of fuel consumption and gadolinia transformation do not match exactly, the control blades are used to maintain the final detailed control for depletion compensation. However, the use of gadolinia supplementary control reduces the total demand on the control blades, permitting greater flexibility for their function of control of overall power distribution throughout the core.

The local power distribution within the core is influenced by nonuniformity within the core. The ideal of complete uniformity of fuel rods and water is interrupted by the control blade between channels or the water gap left when the blade is withdrawn. To prevent excessive heat generation in the peripheral fuel rods adjacent to the water gaps, (fuel rods numbered (2), (3) in Figure 12) the U-235 enrichment in these rods is reduced relative to the central rods of the fuel assembly. Also the enrichment is adjusted a different amount in the corner rods of fuel assembly to avoid power peaking in these rods (numbered (4)). To further equalize heat generation among peripheral and central rods (numbered (1)), the latest fuel design provides more water in the center of the fuel bundle to increase heat generation among the central fuel rods. This is accomplished by removing fuel from a central fuel rod and allowing water to fill the rod

(rod numbered (0)). The additional water moderator results in increased heat generation in the central rods.

As explained above, considerable care and effort is expended to achieve uniformity of heat generation among fuel rods. The peak heat generation rate anywhere in the core must be controlled within limiting values during normal power operation and reactor maneuvering transients or reactor malfunctions. The design objective is to avoid overheating during accidents and to maintain fuel temperatures during normal plant operations which will avoid excessive deformation and/or cracking of the fuel cladding and leakage of radioactive fission products into the reactor system.

Fuel Design

The design variables which affect fuel temperatures are the fuel rod linear heat generation rate, the fuel rod heat transfer surface, and the rate of core coolant water flow. As in any design, there are conflicting effects which must be considered in determining final design. The fuel rod linear heat generation rate can be reduced and fuel heat transfer area can be increased by subdividing the fuel into a larger number of rods of smaller diameter within a bundle. This effect is illustrated in Table 2 which summarizes the comparative performance of two fuel designs for the BWR/2 through BWR/5 reactors:
(1) the 8 x 8 array with fuel rods of 0.493 in diameter, and (2) 7 x 7 array with fuel rods of 0.563 in. diameter. Both fuel bundles contain about the same total weight of UO_2, but the 8 x 8 array with the smaller diameter rod results in significantly lower peak fuel heat generation rate and lower UO_2 and fuel clad temperatures under all conditions. The principal disadvantage of the 8 x 8 array is the increased cost of fabrication of the larger number of rods per bundle.

TABLE 2

COMPARISON OF FUEL BUNDLE CHARACTERISTICS

Fuel Array	7 x 7	8 x 8	Ratio 8x8/7x7
Fuel Rod Diameter - inches	0.563	0.493	0.875
Water Rod	0	1	-
Heat Transfer Surface Sq. Ft./Bundle	88	99	1.12
Active Fuel/Linear Ft/Bundle	596	767	1.29
Linear Heat Generation			
Peak KW/ft. of rod	18.5	13.4	0.73
Peak UO_2 Temperature F	4500	3300	0.73

A summary of the evolution of fuel thermal duty in the BWR is shown in Table 3. Dresden-1 was designed for peak thermal duty of 15 KW/ft. The BWR/2 through 5 reactors were originally designed using the 7 x 7 fuel array. The BWR/2 and 3 reactors were designed for 16.5 KW/ft, and later uprated to 17.5/KW/ft. The BWR/4 and 5 reactors were designed for 18.5 KW/ft. These increases were accompanied by an increase in the colre volumetric power density (KW/liter) also shown in Table 3. The incentive for these increases was to achieve reactor compaction while retaining the 7 x 7 fuel array primarily for its fabrication cost advantage.

In 1971 it was decided to reduce the peak fuel thermal duty by changing the fuel in BWR/2 through 5 reactors from the 7 x 7 to the 8 x 8 fuel array. Several factors account for this reversal in trend. Reactor operating experience indicated that the incidence of fuel failures was related to peak fuel thermal duty. Also, adoption of a more conservative method for calculating fuel clad temperatures in the theoretical loss of coolant accident indicated need for reduction in peak fuel thermal duty. Changing the fuel design from 7 x 7 to 8 x 8 provides significant improvement in both areas.

The 8 x 8 fuel is being introduced into operating BWR/2, 3 and 4 reactors as they require refueling. In new BWR/4 and 5 reactors, 8 x 8 fuel is provided in the initial core. The 8 x 8 fuel now in production for these reactors is described in Table 2, and is referred to as interim 8 x 8 fuel in Table 3.

Concurrent with the decision to change the fuel array in BWR/2 through 5 reactors, the BWR/6 was designed using the 8 x 8 fuel array. As detail design work has proceeded on the BWR/6 reactor, some changes have been made to the earlier 8 x 8 fuel design. These are summarized in Table 4. The bundle size has been reduced to provide more space for nonboiling water between the fuel channels. This reduces the voide coefficient and improves reactor transient performance. The new design

TABLE 3

EVOLUTION OF FUEL THERMAL DUTY

	Design Linear Heat Generation Rate KW/ft. of Fuel Rod		Average Core Power Density KW/Liter of Core Volume
	Peak	Average	
Dresden-1	15	4.5	35
BWR/2-3			39
Current 7 x 7	17.5	5.6	
Interim 8 x 8	13.4	4.4	
BWR/4-5			51
Current 7 x 7	18.5	7.0	
Interim 8 x 8	13.4	5.5	
BWR/6			52-54
8 x 8	13.4(1)	5.4	

(1) Design and licensing value including operating margin; expected peak value in normal operation is 11.8 initial core, 12.1 reload.

TABLE 4

COMPARISON OF INTERIM AND BWR/6 FUEL DESIGN

	Interim	BWR/6	Difference
Fuel Rod Array	8 x 8	8 x 8	---
Active Fuel Length - In.	146	150	+4
Fuel Rod Diameter - In.	0.493	0.483	-0.010
Clad Thickness - In.	0.034	0.032	-0.002
Number of Water Rods	1	2	+1
Water Rod Diameter - In.	0.493	~ 0.600	+0.107
Bundle Size - In.	5.278	5.215	-0.063

incorporates two water rods, each of larger diameter, to improve fuel cycle performance. The fuel rod diameter is reduced to provide equivalent flow area within the smaller bundle. The active fuel length is increased by four inches to offset the reduction in heat transfer surface and linear feet of active fuel resulting from reduction in rod diameter and addition of the second water rod.

In addition to being used in all BWR/6 reactors, when it is placed into production this latest 8 x 8 fuel design will replace the interim 8 x 8 fuel in BWR/2 through 5 reactors.

Reactor and Plant Control

In the core design discussion it was stated that the generation of steam within the core displaces water with a resultant effect on the nuclear reactivity of the core. With reactor pressure held constant, an increase in steam generation and steam void acts to reduce nuclear reactivity and vice versa. This negative steam-reactivity characteristic provides a unique method of automatic control of the BWR, permitting changes in reactor power and steam generation without movement of control blades, provided pressure is held constant. In dual cycle reactors, this control feature was achieved through variation of the secondary steam flow, which resulted in temperature variation of reactor inlet water and reactor steam generation.

In Dresden-1 this system would permit change in load from 45% to 100%. In the single cycle reactor, this type of load control is achieved by variation of the reactor coolant recirculation flow. The BWR/2 and 3 reactors have a flow control range of 50-100% power. As reactor designs evolved in the direction of higher power density and greater compactness, the range of flow control has been reduced. The BWR/4 and 5 range is 65% to 100%, and in BWR/6 this range is 75% to 100% power.

Another benefit of the strong negative steam-reactivity characteristic is its effect on reactor stability. PWR reactors, which do not possess this inherent stabilizing effect because their temperature coefficient is only slightly negative, can encounter internal nuclear instability when power distribution changes occur within the reactor. Large, high performance PWR's (800-1000 MW and above) are subject to internal oscillations of core power which can affect their ability to respond to load changes. The strong negative steam-reactivity characteristic of the BWR provides inherent stability.

The achievement of satisfactory reactor control depends upon ability to maintain constant reactor pressure during normal operation. Reactor pressure regulation is one element of the total turbine-reactor control complex, which is shown schematically in Figure 14. The reactor pressure is controlled by a pressure regulator(1) on the turbine which adjusts the turbine control valves(2) to control steam flow and pressure. In the automatic load control mode, a demand for increased load is indicated by reduced turbine speed in the speed/load control(3) which is signalled simultaneously to the pressure regulator(1) and the recirculation flow controller(4). The pressure regulator opens the turbine control valves(2) by an amount and for a period which are functions of the load/speed error signal. The opening of the turbine valve lowers the reactor pressure, causing a temporary increase in steam flow by flashing of water to steam. Concurrently, the recirculation flow controller(4) opens the recirculation flow valve(5) increasing core recirculation flow and sweeping more steam bubbles out of the core, which increases core average water density and increases the steam generation rate. The higher steam generation rate increases reactor pressure which is a signal for the pressure regulator(1) to open the turbine control valve(2) to the position which controls pressure at the increased steam flow corresponding to the increased load.

Although this control system permits the reactor to respond to load changes

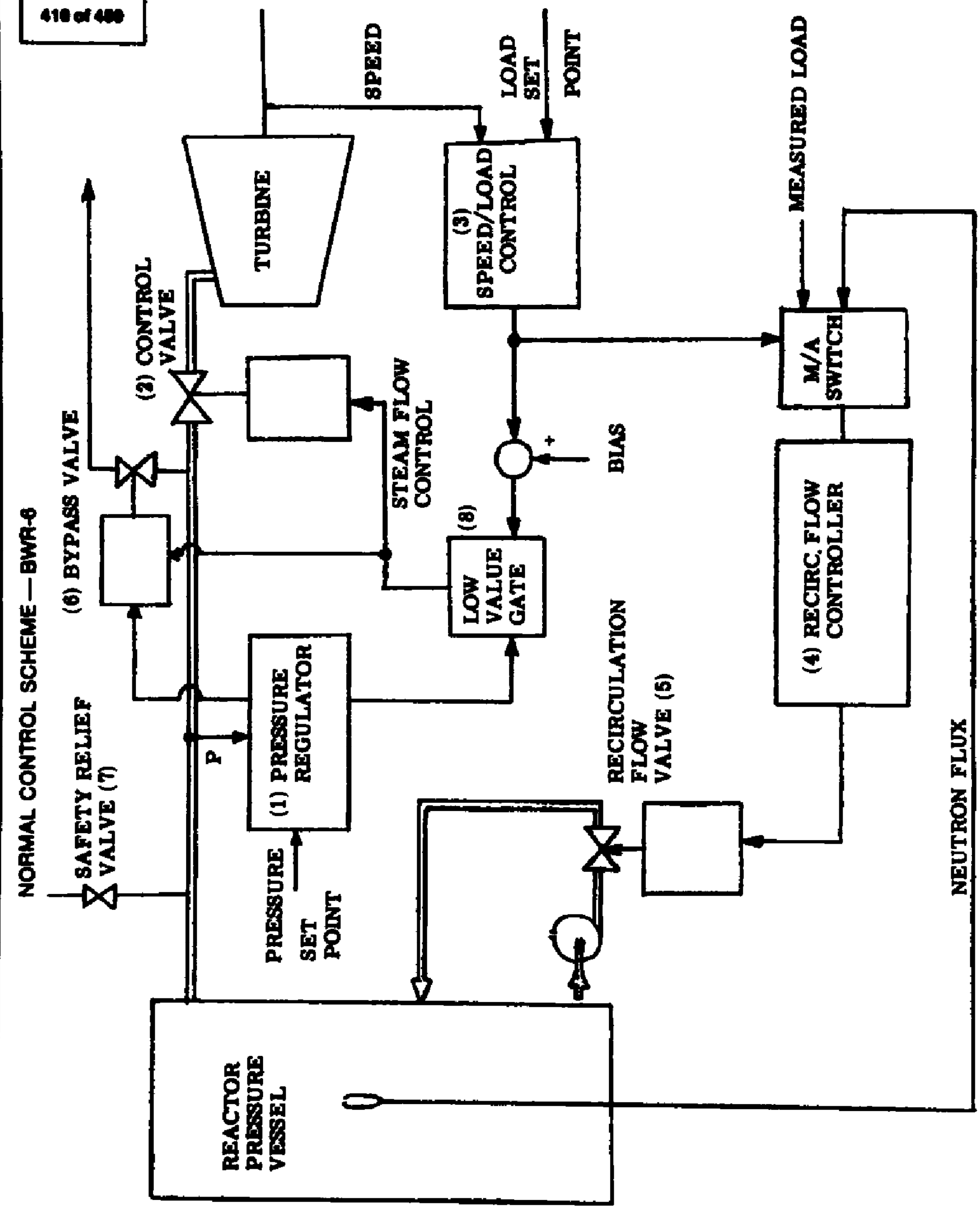

Figure 14

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required by the turbine, the dominant control mode is the regulation of reactor pressure. The low value gate[8] which receives signals from both the pressure regulator and the speed/load control, discriminates between these two signals and positions the turbine control valves to assure that the requirements of reactor pressure regulation are met. The turbine is slave to the reactor, that is, it can deliver increased electrical output only as fast as the reactor can generate increased steam under pressure regulated conditions.

Major reactor pressure increase transients are controlled by turbine bypass valves[6] which are integrated with the turbine control valves and bypass steam to the condenser when the turbine valves[2] are called upon to close quickly. Pressure increase transients beyond the capability of the bypass system are controlled by safety-relief valves[7] which discharge steam into the pressure suppression water pool. As reactor power density has increased, the capacity of the safety-relief system has also been increased. The change in power density from BWR/3 to BWR/4 required about 30% greater relief capacity and the change from BWR/5 to BWR/6 about 20%. These steps correspond to the major changes in power density between succeeding reactor classes. The increasing complexity of reactor pressure control is one of the limitations to further compaction of the BWR.

In some plants, at the option of the customer, the safety-relief valves are integrated with the turbine bypass system to provide relief valve augmented bypass (REVAB) which permits up to 100% load rejection without reactor scram, while the reactor power is automatically reduced to supply auxiliary load only.

In operating reactors, it has become necessary to supplement the original capacity of the safety-relief system to limit peak transient pressure to acceptable values. The need became apparent after several years of reactor operation when it was observed that the time to accomplish fast automatic reactor shutdown (SCRAM) had increased because of undepleted fuel at the top of the core. This delay in reactor shutdown time increased steam output to be discharged by the safety-relief system in event of a turbine trip. To offset this problem, additional safety-relief valves are being added to some plants, and all safety-relief valves in each plant will be actuated earlier by using the turbine trip signal rather than waiting for the pressure rise to initiate valve opening. The earlier trip is referred to as the prompt relief trip (PRT).

Both the added relief capacity and earlier trip act to reduce the peak transient pressure.

For BWR/6 plants, this problem will be resolved by providing control drives and systems capable of faster reactor shutdown, which avoids the need for the prompt relief trip. It is not practical to retrofit the BWR/6 drives and systems to earlier reactors.

Emergency Reactor Control Systems

In the unlikely event that the control blade drive system malfunctions and cannot shut down the reactor, a separate backup reactor shutdown system is provided. This is the soluble liquid control (SLC) system which in the BWR is designed for emergency reactor shutdown only. Its use has never been required to date. This system provides a reserve supply of a concentrated boron solution which can be pumped into the reactor recirculating water and will shut down the nuclear reaction in about two hours when the boron concentration in the recirculating water reaches an appropriate value. This system is designed to shut down the reactor from normal operation, assuming the only malfunction to be that the control blades cannot be inserted. The NRC has stated that it will require accommodation of a major anticipated transient without scram (ATWS).

Alternative systems to meet this requirement are being evaluated. One alternate includes two steps: first, the reactor recirculation pumps would be tripped, which results in increasing core voids and reduces core power to 30%; second, soluble poison would be pumped into the core at a much faster than current rate to achieve shutdown within minutes. Another alternate is the incorporation of entirely different and independent control drives on approximately one-half of the control blades. This would provide added assurance of capability to shut down the reactors. The entire issue of ATWS is currently under discussion with the NRC and is expected to be resolved this year.

Emergency Core Cooling System (ECCS)

The nuclear reactor core continues to generate heat at a low rate after nuclear reaction is stopped. This residual heat generation is caused by radioactive decay of fission products which persists after reactor shutdown. The rate of decay heat generation is shown in Table 5. This characteristic requires provision for assured cooling of the reactor under any and all circumstances to prevent fuel clad overheating and damage which can result in release of the radioactive materials. As stated earlier, the containment is the final barrier to prevent dispersal of radioactivity in event of rupture of the piping of the reactor system. The function of the emergency cooling systems is to prevent overheating and damange of the nuclear fuel in event the normal heat removal systems fail. The fuel cladding represents the initial barrier to dispersal of radioactivity.

The emergency cooling systems are designed to respond to a wide range of possible failures, covering the instantaneous rupture of a major pipe as well as intermediate and small pipe failures. A high degree of reliability is required in all systems to assure availability of cooling water and power, and the capability to deliver the water to the core. The wide range of requirements to be met, combined with high reliability required, results in systems of considerable complexity and cost.

TABLE 5

DECAY HEAT GENERATION

TIME AFTER REACTOR SHUTDOWN - IN SECONDS	DECAY HEAT[1] - FRACTION OF FULL POWER
1	0.346
10	0.066
100	0.043
200	0.036
400	0.031
800	0.026
1000	0.025

(1) ANS Standard + 20%

Three independent methods of admitting cooling water to the core are provided: (1) high pressure core spraying; (2) low pressure core spraying; and (3) core flooding. The redundancy of systems is provided to assure that core cooling is available from two independent systems for any size line perforation or break up to the full severance of a steam line or recirculating water line. Automatic relief is also provided through the safety-relief valves to reduce reactor pressure quickly to permit the use of core flooding and low pressure core spray, both of which are effective only at low reactor pressure. These systems have evolved as the stringency of emergency cooling requirements increased.

One of the major design changes introduced recently is the reduction in fuel linear heat generation rate achieved by changing the fuel array from 7 x 7 to 8 x 8. In BWR/6 this change, plus an increase in core reflooding rate, results in a large reduction in fuel temperatures encountered in event of a major loss of coolant accident (LOCA). These changes are summarized in Table 6.

Reactor Refueling

Under normal circumstances, the reactor is refueled at planned, approximately annual, intervals by replacement of approximately 20 to 25% of the fuel assemblies. The number of new fuel bundles inserted and their average fuel enrichment is determined from the predicted energy needed during the planned operating interval. Some of the fuel assemblies are also relocated in the reactor core as part of the refueling operation. The strategy of fuel replacement and relocation affects the overall efficiency of fuel usage and fuel cycle cost.

The refueling operation requires the reactor to be shut down and the reactor vessel opened for access to the fuel. The principal steps involved in the total refueling operation are shown in Table 7. Spent fuel assemblies are transported and stored in deep canals of water which provides shielding of radioactivity.

TABLE 6

COMPARISON OF EMERGENCY CORE COOLING SYSTEM (ECCS)

IN

LOSS OF COOLANT ACCIDENT (LOCA)

Characteristic	BWR/4		BWR/6
Fuel Array	7 x 7	8 x 8	8 x 8
Peak Heat Generation Rate KW/ft.	18.5	13.4	13.4
Core Reflood Time - Sec.	200	200	170
Calculated Peak Clad Temperature- F	2300-2500*	2200	1940
NRC Clad Temperature Limit	2200	2200	2200

* Requires restriction on maximum bundle power to 85-90% of rated peak bundle power to meet clad temperature limit of 2200 under some operating conditions.

TABLE 7

SUMMARY OF REFUELING OPERATIONS

1. Drywell Head Removal
2. Reactor Vessel Head Removal
3. Steam Dryer Removal
4. Steam Separator Removal
5. Fuel Leak Testing
6. Fuel Removal to Storage
7. Fuel Relocation
8. New Fuel Installation
9. Core Nuclear Testing
10. Reclosure of Reactor and Drywell

Since these refueling operations result in the loss of power generation, there is great incentive to minimize the time required for the entire operation. Other reactor and plant maintenance is usually scheduled during the same outage to take full advantage of the time. Experience to date is that the duration of many of the combined refueling maintenance outages are determined by maintenance rather than refueling requirements (see Figure 22). Nevertheless, refueling requirements contribute to the critical path of many outages.

Refueling equipment and procedures are under continuing review for improvement to reduce the time required to perform the refueling operation. The principal savings are expected to be realized through better planning of the refueling operation and additional training and experience of the operators. A second source of time saving is through improvement in reliability of the refueling equipment, since equipment breakdown has contributed to lost time. The design of refueling systems and equipment is also being improved to reduce the time required to perform individual operations. Most of these improvements are in detail. For example, previously, the tools and fixtures used for unbolting and transporting the reactor vessel head were brought to the job separately in series operations. For BWR/6 they will all be incorporated in a carousel-like structure with provision to execute more of the operations in parallel. Each individual operation in refueling is being reviewed in like manner for possible time saving.

Another significant change in the BWR/6, Mark III refueling system is the removal of the fuel storage pool from within the containment building to a special fuel storage building. The change contributes to the simplification and size reduction of the Mark III containment discussed earlier. It requires a special mechanism for the transfer of spent fuel to the fuel building.

Reactor Water Chemistry

Impurities in the BWR reactor recirculating water and feedwater are maintained at extremely low levels through continuous purification. Condensate water is treated, prior to entering the feedwater heater system, by demineralizer or filter/demineralizer combination to reduce impurities to the parts per billion level. Reactor recirculating water is cleaned up in a bypass filter/demineralizer system which is sized at about 1% of the full feedwater flow and can process the water in the reactor and recirculating system in about 4-1/2 hours. No additives are used in the BWR for control of water quality. Materials used in the reactor recirculating and feedwater systems are selected for corrosion resistance. The resulting water is high purity (conductivity 1 micro mho/cm) and essentially neutral pH (range 5.6 to 8.5). Radiolytic decomposition of water into hydrogen and oxygen occurs in the reactor and the separated gases flow out of the reactor along with the steam. These gases are generated at a rate of about 0.1 lb/hr/MW thermal.

The liquid phase of the reactor water contains about 0.2 ppm of dissolved oxygen and a stoichiometric* quantity of dissolved hydrogen. Some technical people theorize that the oxygen level in the reactor recirculating water during normal operation is a major contributing environmental factor in the stress corrosion cracking of auxiliary piping of the BWR recirculating water system; however, recent detailed investigations do not substantiate this view.

Radioactivity and Plant Radioactive Releases

Most atomic nuclei exposed to neutron radiation in the reactor become radioactive, i.e., the original nucleus is transformed into an unstable nucleus which emits radiation as it spontaneously transforms into another nucleus or element.

---

* Ratio of molecular weights of H to O in H_2O (1/8)

The rate of radioactivity decay* and the energy of the emitted radiations are inherent atomic characteristics which vary widely. In the water reactor, the radioactive materials of concern are the water, corrosion products and fission products. In the reactor neutron field, the oxygen atom in water or steam is transformed into nitrogen 16 which is unstable and transforms back to oxygen with emissions of a high energy gamma ray. The half-life of nitrogen 16 is seven seconds, hence, for practical purposes, water radioactivity drops to a negligible value in minutes after reactor shutdown.

Fission products are an accumulation of many atomic nuclei which in composite are highly radioactive and remain so for many years. Fortunately, most of the fission products remain within the sealed cladding of the fuel rod. Of the small fraction of fission products which may leak out of cracks in the fuel cladding into the recirculation water and steam systems, most are transported through these systems and eventually wind up in the liquid and gaseous waste streams of the plant.

Corrosion products are also a composite of a number of atomic nuclei; however, the dominant source of long term radioactivity in water reactor system corrosion products is cobalt 60 which has a half-life of five years. Cobalt is an impurity in stainless steel and is a constituent of hard face alloys used in plant equipment. Because of this long half-life, radioactive decay has little practical effect in reducing the strength of this source of radiation.

Some of the fission and corrosion products in the reactor recirculating water are carried into the liquid and gaseous waste streams of the plant. These streams are treated to reduce the contained radioactivity to acceptably low levels prior to discharge into the environment. The cleanup of liquid waste streams from both the BWR and PWR utilizes essentially the same chemical technology.

* Rate of radioactive decay is measured in terms of half-life, which is the time required for the rate of radiation emission to be reduced by a factor of two.

In the BWR, the gaseous fission products which may leak out of the fuel in case of cladding failures, along with the hydrogen and oxygen produced by radiolytic decomposition of water in the reactor, are carried along with the stream, through the turbine, into the main condenser. These noncondensable gases and air inleakage are normally pumped out of the condenser to maintain its vacuum. With a low incidence of failed fuel rods (less than 0.1%), the radioactive fission gases can be safely discharged into the atmosphere (within regulatory standards) after a 30 minute holdup to allow for decay of their radioactivity.

In event of a high incidence of fuel rod failures, the greater volume of fission gases must be held up for a longer period of radioactive decay time prior to their discharge into the atmosphere. To provide a practical gas hold up, the radioactive gases are absorbed on activated charcoal where they reside for several months. In order to reduce the amount of gases to be processed, the hydrogen and oxygen are removed before the charcoal bed by recombination into water and condensation out of the process stream. The gas entering the charcoal bed is mostly air (approximately 30 standard cubic feet per minute) containing a few cubic centimeters of radioactive fission gases. The radioactive gases (Xenon and Krypton) are preferentially absorbed in relation to their higher molecular weights. Such "off gas" treatment systems are now included in all new plants and are being added to earlier BWR plants. The addition of off gas systems will reduce BWR gaseous discharges to extremely low levels below the values required by regulations. For example, a simple 30 minute hold up system discharging 100,000 micro curies per second of gaseous activity from about 100 leaking fuel rods, would result in an annual dose rate at the site boundary of 160 milli-rem/year (mr/year). This dose rate would be reduced to about 5-10 mr/year in an ambient temperature charcoal absorber bed system and to 0.05 to 0.10 mr/year in a refrigerated bed.

Some fraction of the radioactive corrosion products are deposited throughout the recirculating water system piping and equipment. The continuing deposit of additional cobalt during operation results in a build-up of radioactivity to the extent that maintenance of the systems and equipment is impaired. In older water reactor plants, some maintenance has to be performed in a carefully staged manner by teams of maintenance personnel whose time in the high radiation zone is limited. Although maintenance in both the BWR and PWR is affected by this radioactivity, the BWR has a superior historical record of radiation exposure to plant personnel, averaging about 1/2 of the PWR exposure (218 vs. 414 man-rem/yr/plant). The higher exposure in the PWR is due in large part to steam generator inspection and repair.

The continuing build-up of radioactivity in systems external to the reactor may eventually (10 to 30 years) reach intolerable levels and require reduction to permit maintenance work. Such reduction has been achieved in some early plants by chemical decontamination. A chemical decontamination of Dresden-1 is now being planned for implementation in 1976. It is a complex and costly operation and requires several months of plant outage time.

The reactor suppliers and utilities are engaged in cooperative efforts to study the sources, transport, deposition and clean up of radioactive materials in systems external to the reactor. The objective of these studies is to reduce the build-up of radioactivity in the external systems to avoid or minimize the need for a major decontamination.

Principal steps in the technical evolution of the BWR are summarized in Table 8.

TABLE 8

	1955 BWR/1	1963 BWR/2	1965 BWR/3	1966 BWR/4	1969 BWR/5	1972 BWR/6
Example	Dresden 1	Oyster Creek	Dresden 2	Browns Ferry 2	Zimmer	Grand Gulf
No. Units Operation	10	3	9	9	0	0
Requisition	-	-	-	17	10	41
Total	10	3	9	26	10	41
Core: MW/Bundle	1.5 to 2.4	3.5	3.5	4.3	4.3	4.9
Fuel Array	6x6, 7x7, 9x9	7x7	7x7	7x7	7x7	8x8
Kw/l	35	37	39	51	51	54
Kw/ft	10.3 to 15.8	17.5	17.5	18.5	18.5	13.4
Average Void Fraction			.34	.415	.415	.436
Recirculation	Multi-loop External	Multi-loop External	Jet Pump Two Loop	Jet Pump Two Loop	Jet Pump (5 hole) Two Loop	Jet Pump (Compact) Two Loop
Steam Separation	External	Internal	Internal	Internal	Internal	Internal
Load Following	Secondary Steam Pressure	Core Flow with M-G Set	Core Flow with M-G Set	Core Flow with M-G Set	Core Flow, Flow Valve	Core Flow Flow Valve
ECCS	None, or 1 LPCS	2 LPCS Feedwater ADS	Feedwater HPCI 2 LPCS LPCS ADS	Feedwater HPCI 2 LPCS LPCI ADS	Feedwater HPCS LPCS LPCI ADS	Feedwater HPCS 2 LPCS LPCI ADS
Containment	Dry	Pressure Suppression (Mark I)	Pressure Suppression (Mark I)	Pressure Suppression (Mark I)	Pressure Suppression (Mark I and II)	Pressure Suppression (Mark III)

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D. PLANT AVAILABILITY AND CAPABILITY

Definitions

The terms of most interest in measuring nuclear plant quality are Plant Availability, Capability and Capacity Factor. These terms may be defined as follows, and are shown graphically in Figure 15.

$$\text{Plant Availability} = \frac{\text{Operable Hours}}{\text{Period Hours}}$$

$$\text{Nonavailability} = 1 - \text{Availability}$$

$$= \frac{\text{Scheduled Downtime} + \text{Forced Downtime}}{\text{Period Hours}}$$

$$\text{Capability Factor} = \frac{\text{Achievable KW Hours}}{\text{Rated KW} \times \text{Period Hours}}$$

$$\text{Capacity Factor} = \frac{\text{Actual KW Hours}}{\text{Rated KW} \times \text{Period Hours}}$$

Availability is simply the fraction of time within a given period that a plant is available for operation, though not necessarily at its rated capacity. The Capability Factor is the fractional amount of energy (kilowatt hours) the plant is capable of producing over a period of time as compared to the energy that would be produced at full capacity and availability. Capacity Factor is the fractional amount of energy (kilowatt hours) the plant is actually called upon to produce over a given time period as compared to the energy that would be produced at full capacity and availability. Capacity factor can also be viewed as the plant's average or effective capacity during a time period relative to its rated kilowatt capacity.

Plant capability is determined not only by plant availability, but also by plant deratings or restrictions in the rate at which power level can be changed. If there are no derates or restrictions, the capability factor would be numerically equal to the plant availability. The plant capacity factor will be lower than the capability factor if the plant is used to load follow or in other ways which do not take full advantage of the plant's capability. Since nuclear plants are used for base load, they are generally operated at capacity factors close to their capability. Typical levels of availability, capability and capacity factor are shown in Figure 15.

Nonavailability is the complementary fraction of time that a plant is not available for operation. Nonavailability is composed of two parts: (a) the Forced Outage Rate which is the fraction of time the plant must be shut down for unscheduled repair or maintenance; (b) the Scheduled Outage Rate which is the fraction of time the plant is shut down for refueling and scheduled maintenance. The actual "scheduled" outage time may exceed that which was originally scheduled.

Other terms of interest are Durability, Reliability and Time to Repair. Durability is a measure of a hardware component's resistance to wear and abuse. Numerically it may be equated to the component's service life or replacement life. Durability failure is only one of the factors contributing to reliability.

The term Reliability is sometimes used as a measure of Forced Outage Rate.

Plant Reliatility = 1 - Forced Outage Rate

To say, in this sense, that a plant has "high reliability" is to say that it has a low forced outage rate.

In a more restricted sense, the term Reliability refers only to the probability of failure or shutdown as measured by the number of failures in a specified time period. The duration of the shutdown is handled as a separate parameter. Reliability is often used in this sense in referring to specific hardware components.

REPRESENTATIVE CAPACITY DISTRIBUTION

1974 Average - All Domestic BWR Plants

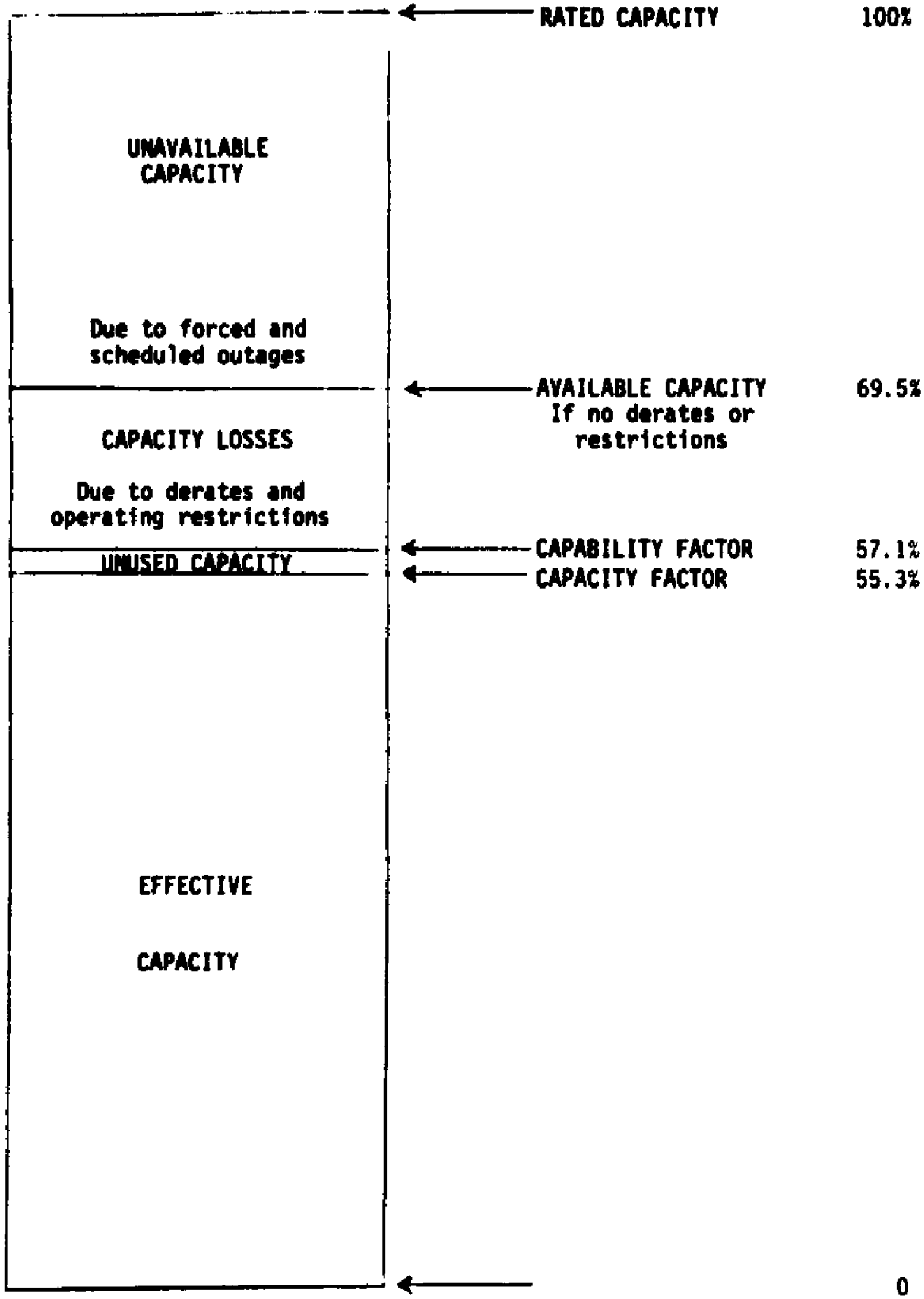


Figure 15

$$\text{Component Reliability} = \frac{\text{Number of Failures}}{\text{Period Hours}}$$

$$= \frac{1}{\text{MTBF}}$$

Where MTBF = Mean Time Between Failures. The availability of a specific hardware component may be defined as:

$$\text{Component Availability} = \frac{\text{MTBF}}{\text{MTBF + MTTR}}$$

Where MTTR = Mean Time to Repair or Replace the component. The latter term includes not only the actual time to perform the physical repair or replacement, but also that required to obtain parts and to mobilize the repair or replacement effort.

It is obvious that the achievable level of availability is determined not only by component reliability, but also by the duration of the shutdown required to repair or replace the component. Preventative maintenance, performed before component failure, improves reliability. On the other hand, reduction in the time to perform preventative and corrective maintenance is just as important as improved component reliability in achieving a high level of availability.

Availability Comparisons and Trends

Plant availability for all domestic nuclear plants as compared to that of domestic fossil fired steam plants is shown in Figure 16. This shows that large fossil plants (over 600 MW) have, on the average, a total plant availability which is not significantly different from that of nuclear plants which are in the same size range.

The comparative availability of operating BWR and PWR plants from U.S. suppliers is as follows: The data are averages through 1974, except foreign PWR data are through 1973.

CUMULATIVE PLANT AVAILABILITY

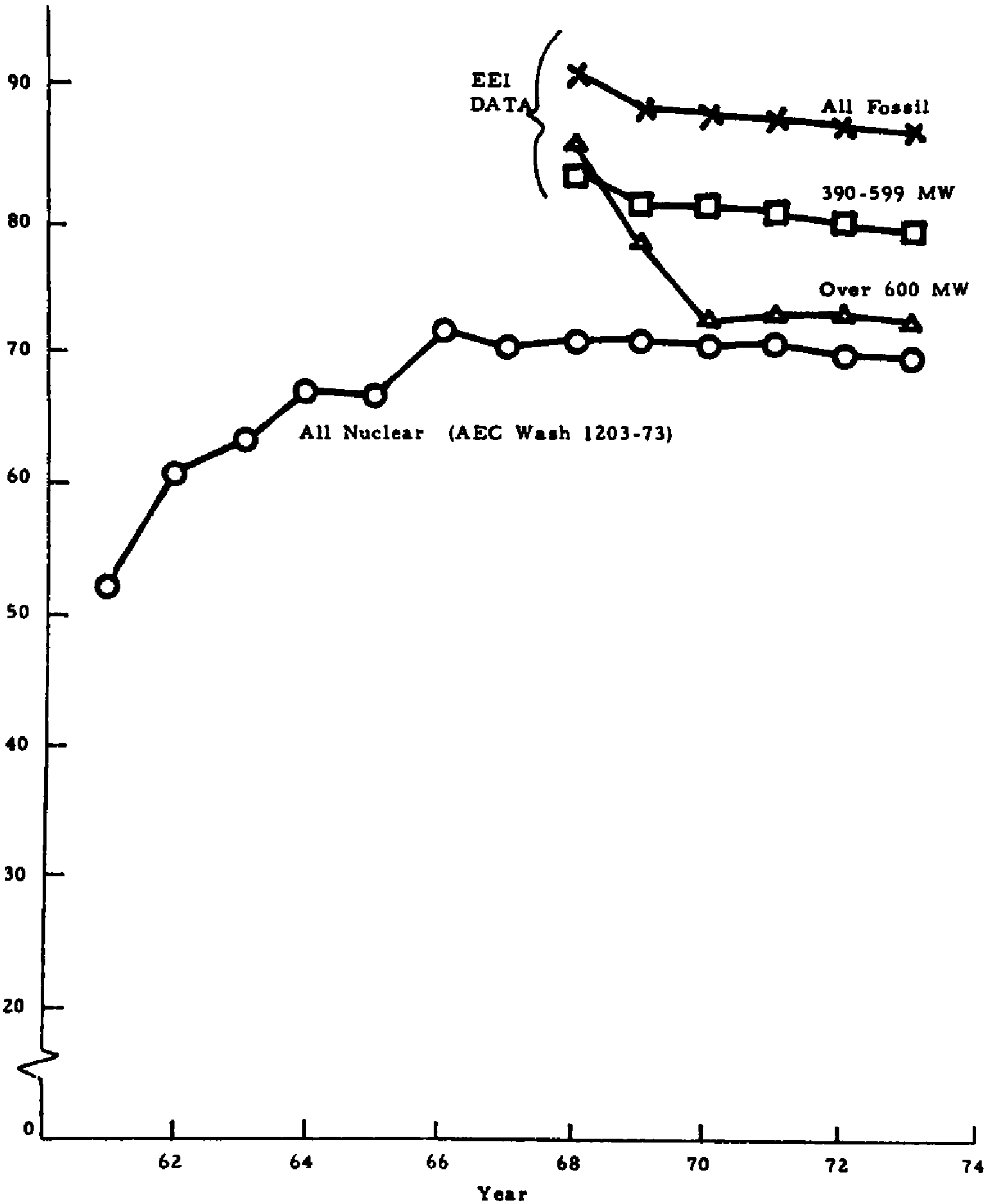


Figure 16

CUMULATIVE AVERAGE AVAILABILITY - %

	Domestic	Foreign	All Plants
GE - BWR	69.9	70.0	70.0
W - PWR	68.7	59.3	65.5
All PWR	65.0	59.3	63.5

It can be seen that Westinghouse domestic PWR's (68.7%) are almost identical in availability to GE domestic BWR's (69.9%). On the other hand, when all plants are considered, the BWR's (70.0%) appear to have been somewhat superior in operating availability to all PWR's (63.5%). The comparative trends in availability for all BWR's and PWR's are shown in Figure 17.

Figure 18 compares the availability and capacity factor of BWR nuclear plants. It can be seen that there is a significant difference between capacity factor and plant availability. This is due, in large part, to restrictions which have been imposed on operating power and rate of change of power. The effect of such restrictions on plant capacity is illustrated in Figure 19.

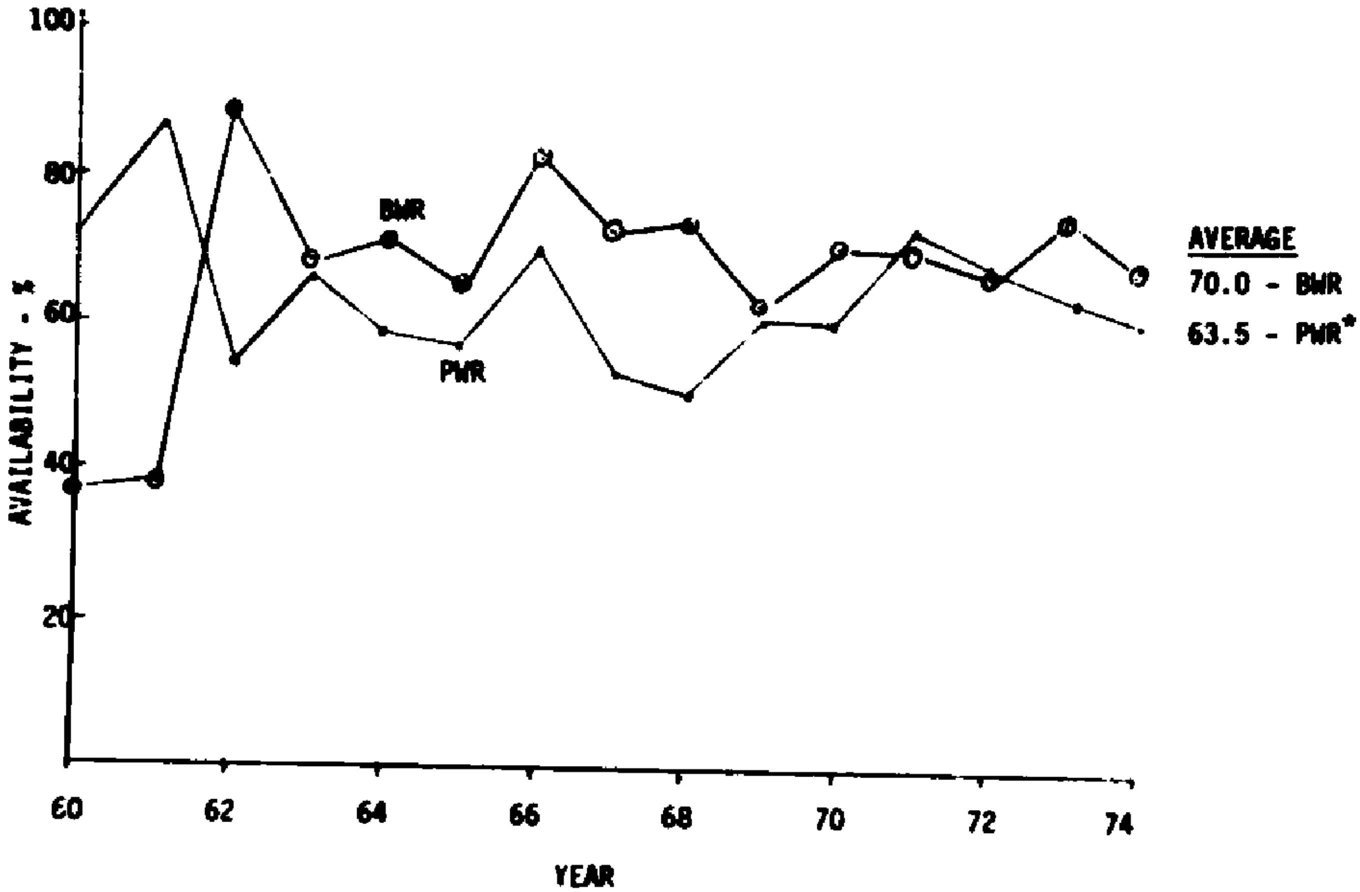
It can be seen from Figures 16, 17 and 18 that there is no significant trend in nuclear plant availability or capacity factor. Availability has persisted at a level of about 70% for several years; capacity factor at about 55%. This is also shown in the cumulative average nuclear plant availability plotted against cumulative years of plant operating experience (Figure 20).

BWR Availability/Capability History

Plant availability is important to the utilities since nuclear units are intended for base load. Any lost capacity, whether due to forced or scheduled outage, must be made up by plants having higher fuel costs. The second measure of importance is the amount of time the unit is operated under conditions of forced derate, i.e., at reduced capability or capacity factor. For example, operation at less than full capacity due to excessive offgas activity.

AVERAGE PLANT AVAILABILITY BY CALENDAR YEAR

ALL BWR AND PWR PLANTS



*Westinghouse PWR Average = 65.5 (all plants)

Figure 17

AVERAGE PLANT AVAILABILITY - ALL BWR PLANTS

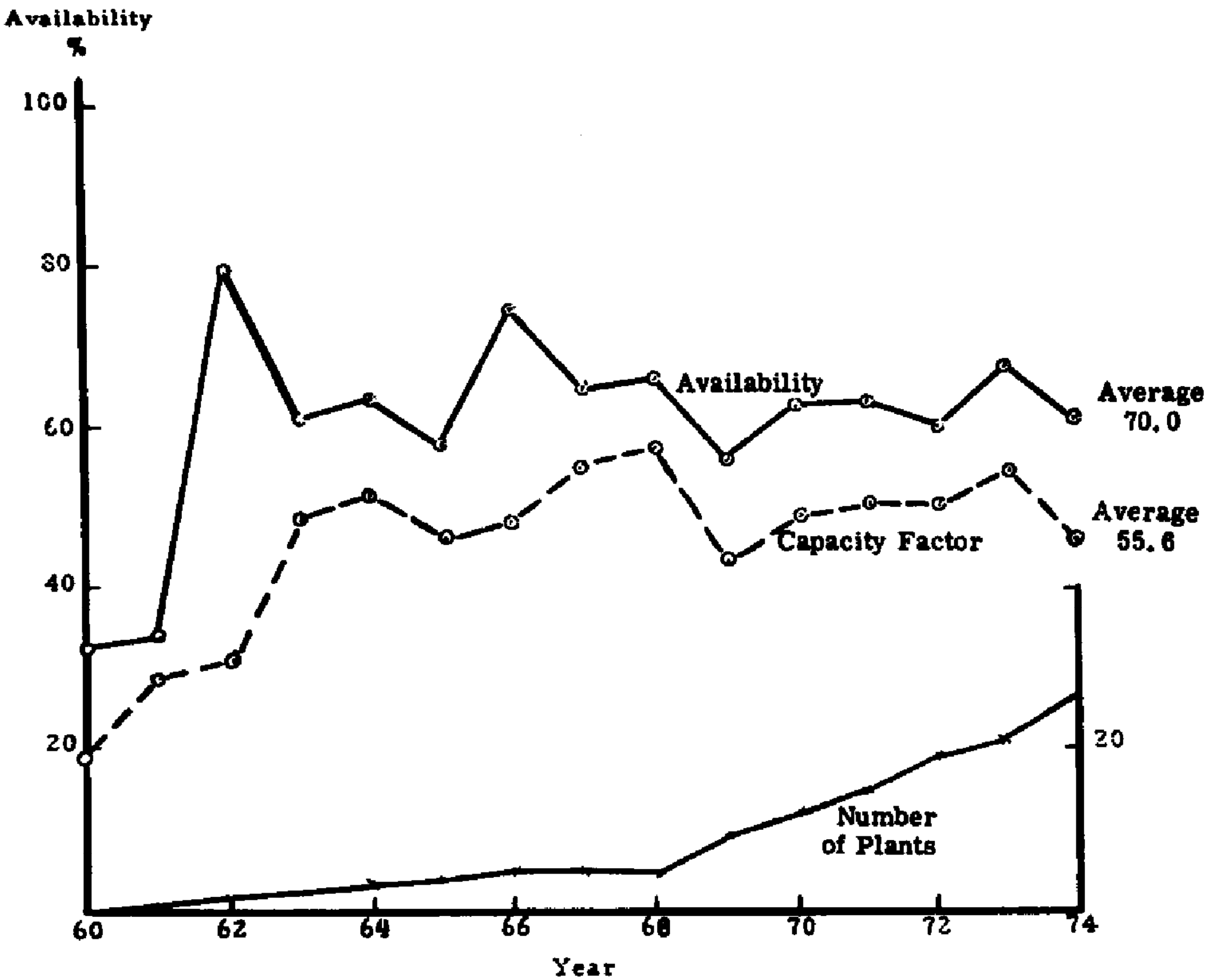


Figure 18

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POWER HISTOGRAM OF BROWN'S FERRY 1
JANUARY 1975

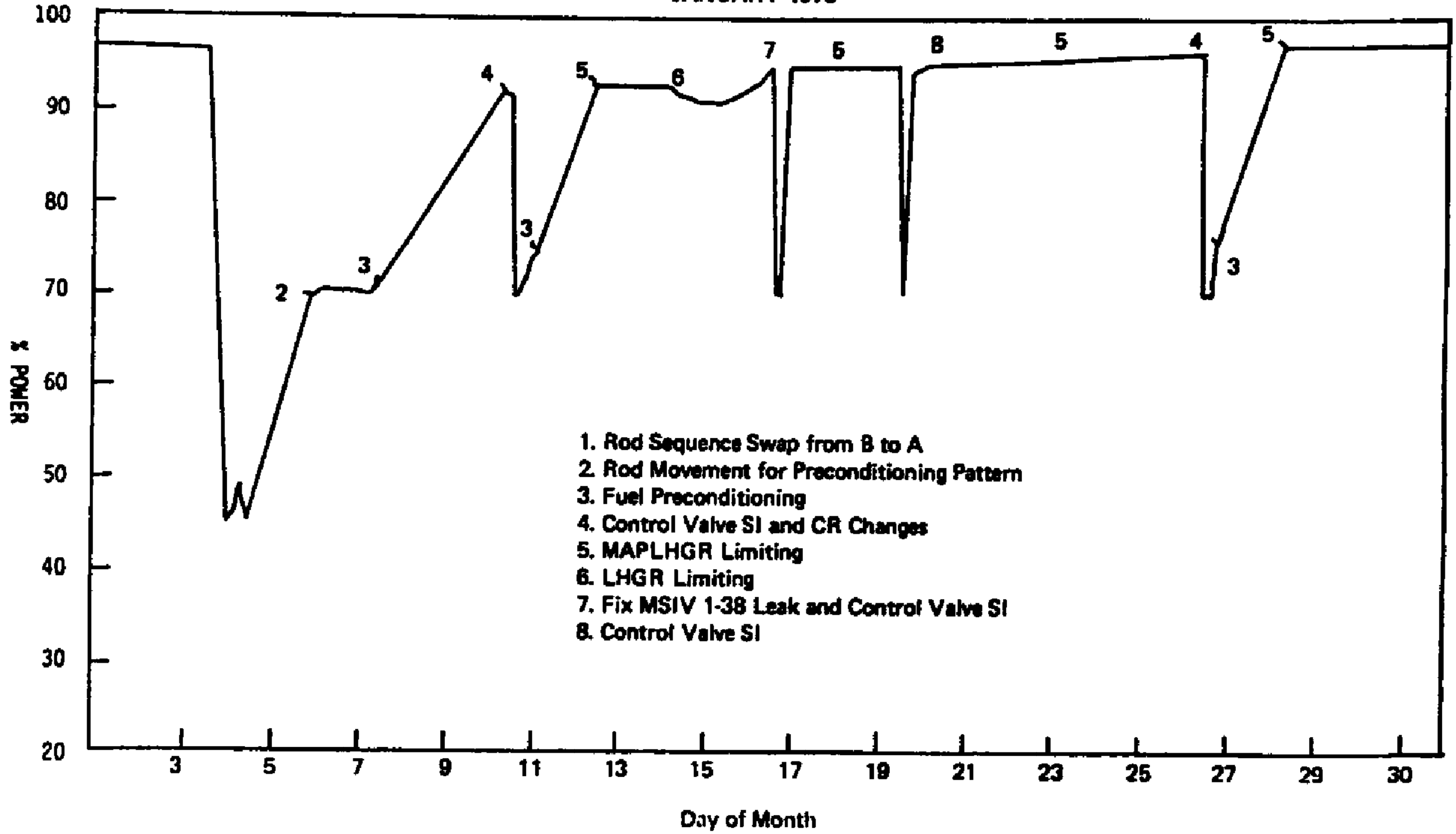

Figure 19

CUMULATIVE AVERAGE PLANT AVAILABILITY VERSUS EXPERIENCE

ALL BWR PLANTS

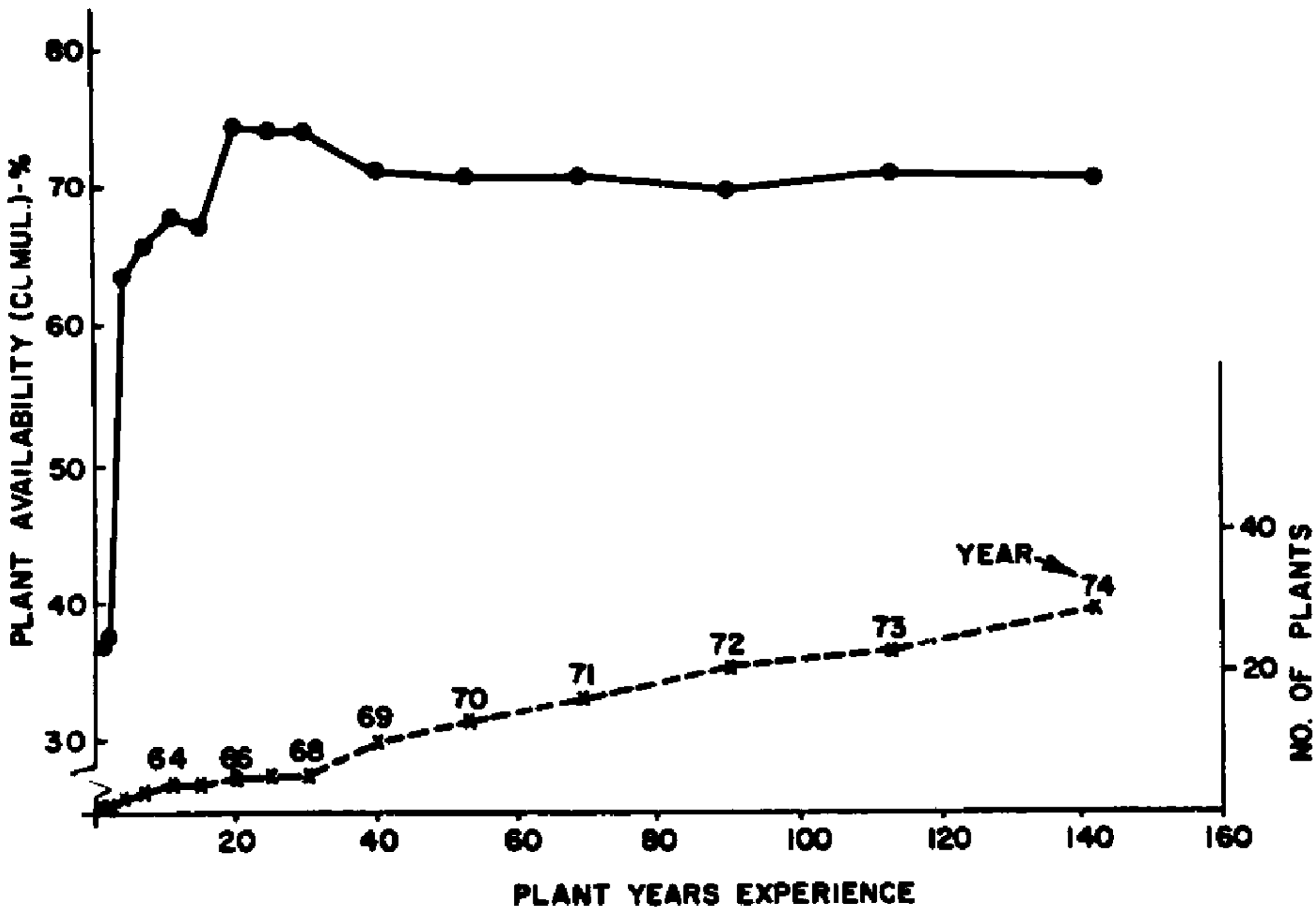

Figure 20

BWR's have accumulated a total of approximately 125 reactor-years of operation. During that period, the year-by-year average availability is shown in Figure 17. At the end of 1973, after 100 reactor-years of operation, an analysis was made of the causes of BWR unavailability or derated operation. The potential generation that could have been achieved during that 100 reactor-years (full time operation at rated capacity) was about 250 x 10^9 kilowatt hours. Generation actually achieved was about 150 x 10^9 kilowatt hours. Capacity Factor was thus about 60%. Of the 40% lost energy, 8% was due to operation at less than rated capacity and 32% was due to plant shutdowns. Some of the 8% loss from operation at less than rated capacity is attributable to load following. The remainder includes capacity lost during startup and shutdown, as well as periods of operation at forced derate due to equipment problems or regulatory restrictions. A breakdown of causes, other than load following, for the year 1973 is shown in Table 9. The 8.1% in 1973 is consistent with the 8% average for the entire time period.

The plant shutdowns accounting for the 32% capacity loss included both forced and scheduled outages. These outages occurred at various phases of plant operation as shown in Figure 21. Plant shutdowns during the period from start-up until the first reload (first cycle) accounted for a loss of 35 x 10^9 kwh, or 14% of rated capacity. Shutdowns for "annual" refueling and maintenance accounted for 32 x 10^9 kwh of lost generation, or 12.8% of capacity. Finally, shutdowns during "operating" cycles between refueling outages accounted for 13 x 10^9 kwh of lost generation, or 5.2% of capacity.

The relatively low availability of the plants during first cycle is largely attributable to a significant number of outages of longer than 30-day duration. There were 24 such outages, accounting for 2/3 of the unavailability during the first cycle. Causes of the long outages have included control rod drive modifications, suppression pool torus repairs, feedwater sparger repair, fuel repair and replacement, jet pump repair and turbine and generator problems.

The other area which has impacted upon availability is the duration of the refueling-maintenance outage. The average length of such outages was 76 days through 1973. Individual outages have ranged in length from 20 days to 273 days, as indicated by the distribution of outage duration shown in Figure 22. The long durations are, in general, due to the scheduling of major plant modifications to be done at the time of the annual refueling.

E. BWR-PWR COMPARISON

Nuclear Systems

The BWR uses the direct cycle in which steam generated in the reactor is transmitted directly to the turbine. In the PWR, steam is generated in a heat exchanger utilizing water heated in the reactor (see Figure 5). The major elements of the BWR nuclear steam supply system were described and illustrated earlier. The configuration of a Westinghouse (W) PWR steam supply system is shown in Figure 23. This system includes the reactor(1), four steam generators(2) and their associated recirculating loop piping and the pressurizer.(3) The function of the pressurizer is to maintain the high system pressure which prevents boiling of the water in the reactor core and to provide space for expansion and contraction of the water due to density changes.

As indicated earlier, the BWR turbine plant is designed to handle radioactive steam and fission gases. Because of the steam generator, the PWR turbine plant is not designed to accommodate significant radioactivity, and the occurrence of leakage of radioactive water through the steam generator tubes requires plant shutdown for repair of the steam generators. This has been a major cause of loss of availability in PWR plants and all PWR suppliers have major programs

TABLE 9

BWR CAPACITY LOSS FROM REDUCED POWER OPERATION

1973

Cause	Equivalent Annual Capacity Reduction 1973 Actual (%)
Fuel Densification	0.8
New ECCS Criteria	0
EOC SCRAM Reactivity	0.05
Fuel Channels	1.1
Feedwater Sparger	1.2
Offgas	0.8
Miscellaneous (e.g., Recirculating System)	1.1
PCIOMR	3.0
TOTAL	8.1%

PLANT SHUTDOWNS
(ALL CAUSES)
80×10^9 KWH
(32% of Total Capacity)

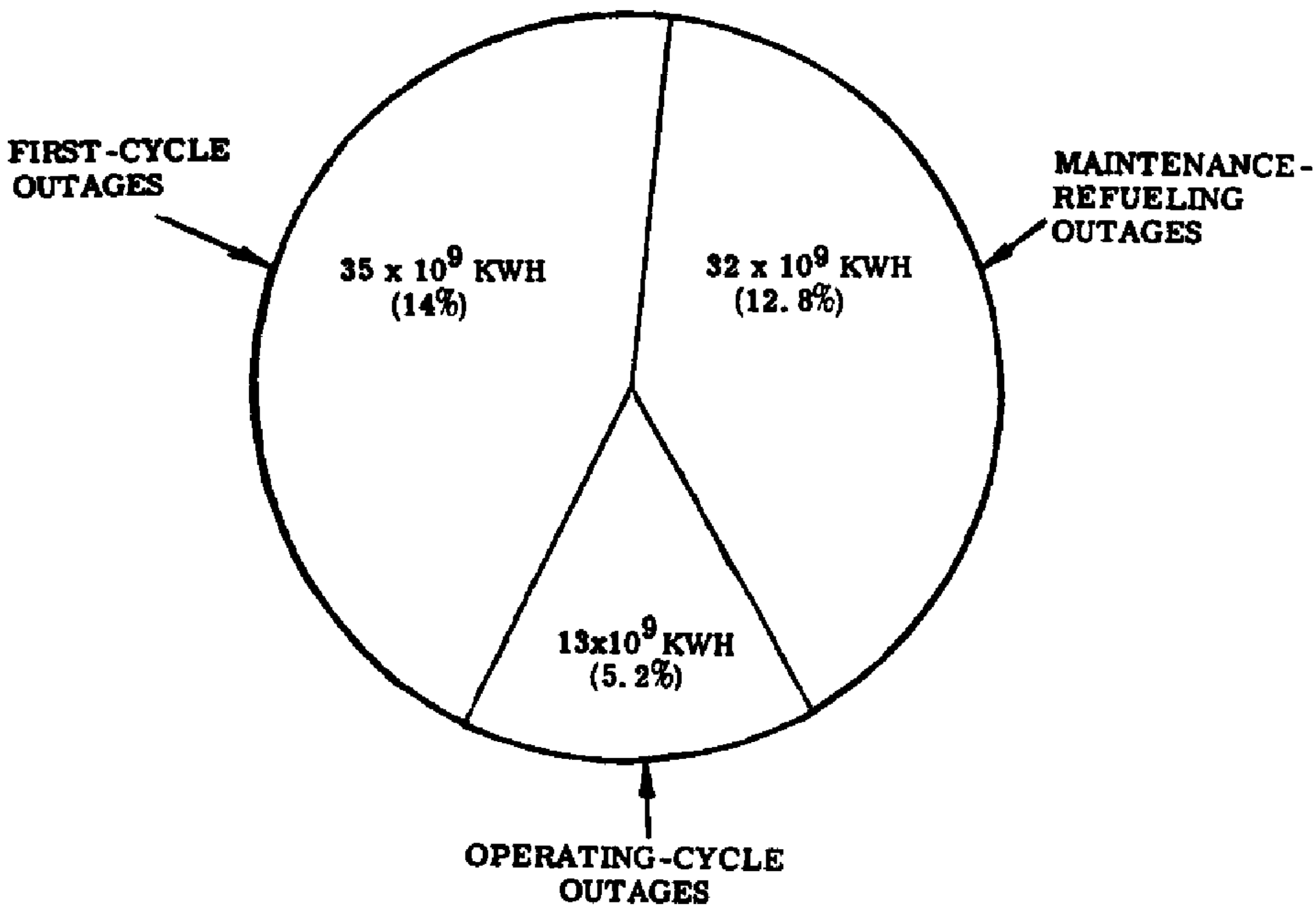

Figure 21

FREQUENCY DISTRIBUTION OF

REFUELING-MAINTENANCE OUTAGE DURATION

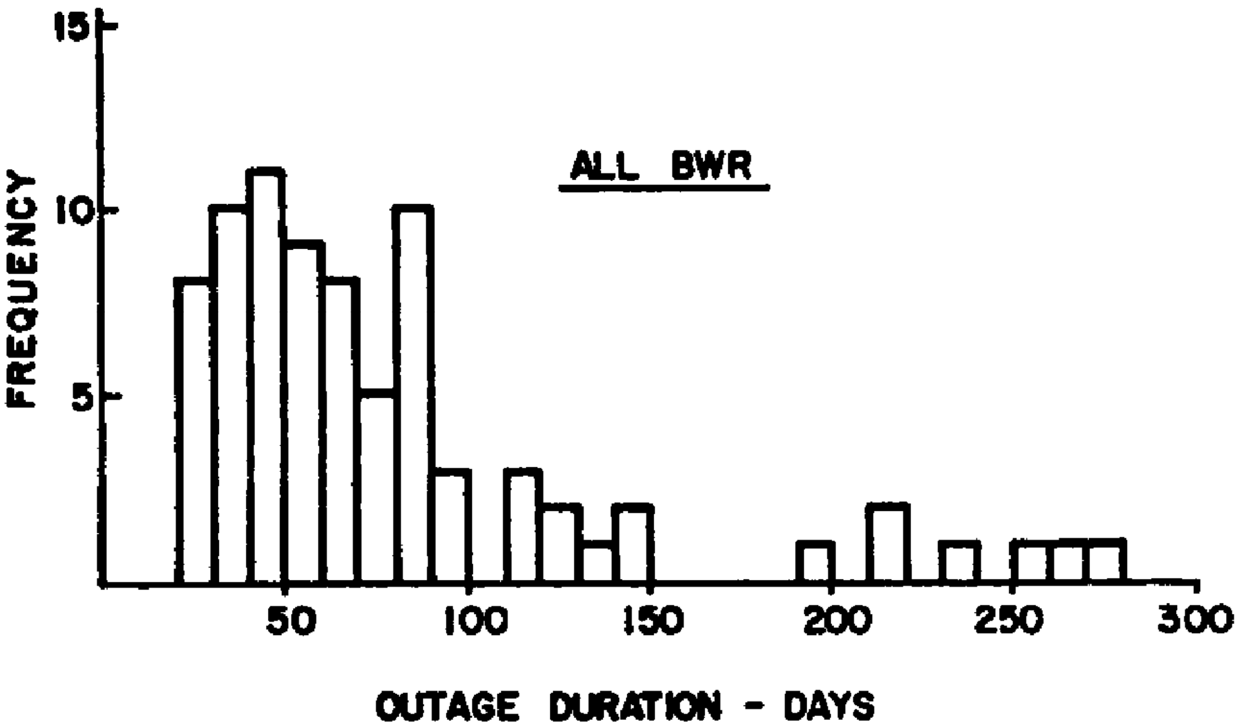

Figure 22

SIMPLIFIED DIAGRAM OF FOUR-LOOP
NUCLEAR STEAM SUPPLY SYSTEM (WESTINGHOUSE)

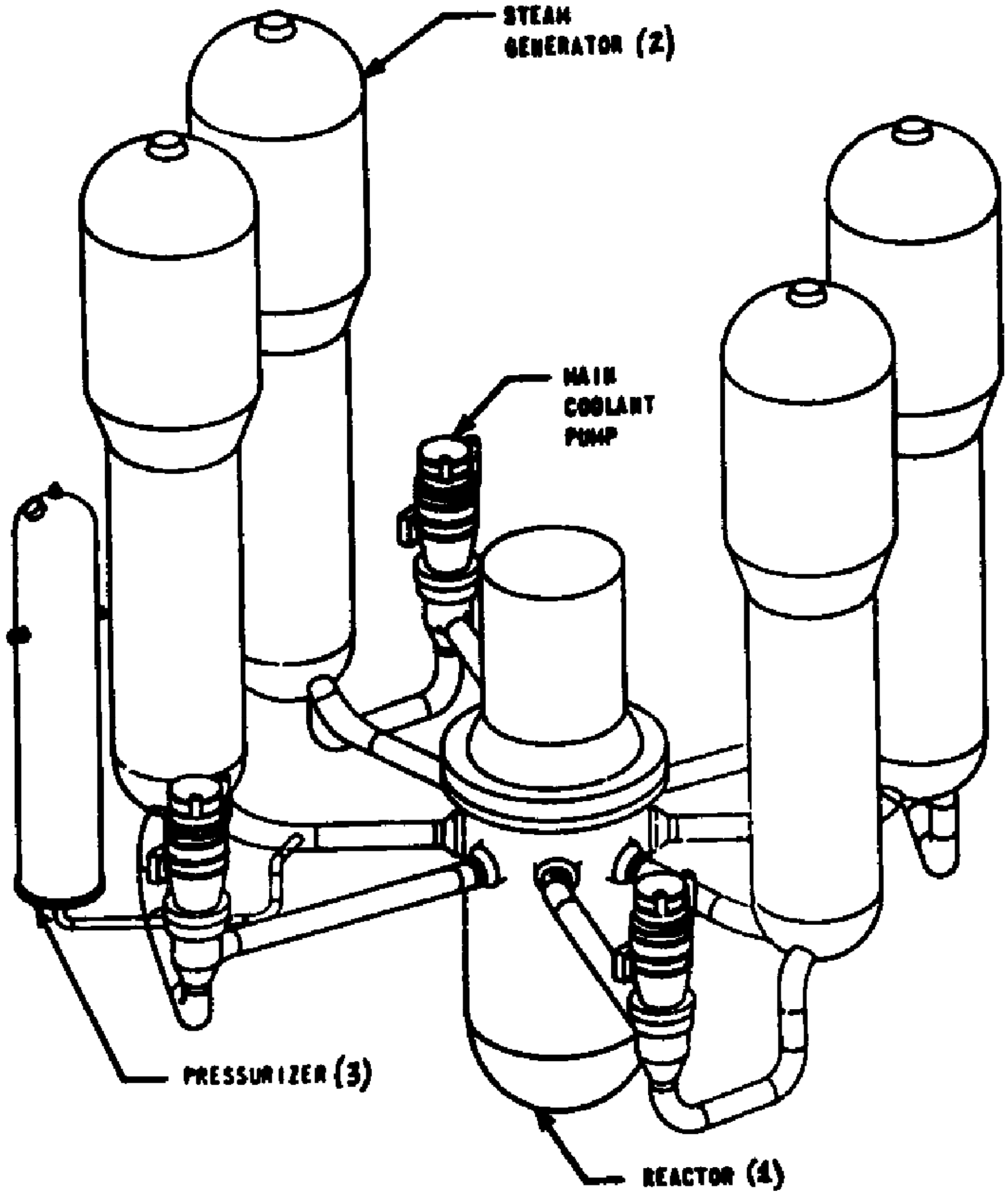

Figure 23

to improve reliability of their steam generators. One of the main changes is in improvement of water chemical treatment in the secondary side of the steam generator. The conventional phosphate treatment is being replaced with a volatile chemical agent and the feedwater is being demineralized to limit impurities carried in with the feedwater. These changes increase the capital and operating cost of the PWR, reducing its competitiveness relative to the BWR.

The presence of steam generators in the PWR account for other major differences in characteristics between the PWR and the BWR. These are summarized in Table 10.

Both plants are designed for essentially the same steam conditions, approximately 1000 psi 550°F and attain comparable thermal efficiencies. In the direct cycle BWR, the reactor system is designed for operation at this pressure and temperature. In the PWR, the peak reactor system temperature is approximately 100°F higher to provide a temperature difference for transferring heat from reactor water to steam in the steam generator. To suppress boiling in the reactor at this temperature, the reactor recirculating water is pressurized to 2250 psia, or more than twice the BWR reactor pressure.

The presence of the steam generator also accounts for differences in chemistry of the BWR and PWR reactor water. The PWR is a closed water recirculating system; the BWR is an open system with net steam output. The chemistry of the BWR reactor water was described earlier. Basically, high water purity is maintained by clean-up of the reactor feedwater and recirculating water, and no additives are used to control dissolved oxygen or pH. In the PWR, a number of additives are used to control reactor water chemistry, in addition to clean-up of the water for removal of impurities.

In the PWR, the radiolytic decomposition of water into hydrogen and oxygen is retarded by maintaining an overpressure of several atmospheres of hydrogen which promotes the recombination reaction of oxygen and hydrogen. This results in maintenance of dissolved oxygen below 0.1 ppm in the recirculating water. Also the pH is controlled in the range of 8.5 to 10 by the addition of ammonia. The low dissolved oxygen and high pH are maintained to minimize corrosion and release of corrosion products into the recirculating water.

The PWR has an additional equirement on the chemistry of the reactor water. As described below, the PWR uses a soluble neutron absorber in the reactor water to perform some reactor control functions. From the water chemistry standpoint, this represents another additive to be controlled.

It is difficult to make an evaluation of comparative advantages and disadvantages of the reactor water treatment systems since they are very different and include somewhat different functions. From the standpoint of radioactivity in the recirculating systems, the buildup rate is approximately the same in both the BWR and PWR. From the standpoint of stress corrosion cracking, the BWR has experienced more problems, but it is not substantiated that water chemistry is a dominant factor. From the standpoint of operability, the relatively simple current water chemistry controls of the BWR represent an advantage, but a modification of water conditions, if required, would be more difficult to implement in the BWR open system.

Reactor Core and Control

The suppression of boiling water within the PWR core results in a design with significantly different characteristics from the boiling water reactor. Some of the elementary characteristics are compared in Table 10.

The PWR core power density is approximately double that of the BWR, or, stated another way, for the same heat output, the PWR core is approximately half the

TABLE 10

COMPARISON OF BWR AND WESTINGHOUSE PWR CHARACTERISTICS

(3800 MW Thermal Standard Plants)

Feature	BWR	PWR
Core power density KW/liter of core	~ 54	~100
Core diameter/height - inches	197/148	133/164
Fuel weight lbs. UO_2	400,000	254,000
Number of fuel rods in core	53,500	51,000
Number of fuel bundles	848	193
Number of control elements in core	205	69
Steam conditions psia/F	985/545	1100/556
Reactor Water (maximum) psia/F	1000/546	2250/650
Reactor Water chemistry:		
pH	5.6-8.5	8.5-10
O_2 parts per million	0.2-0.3	$<$0.10

volume of the BWR. The smaller core results in a smaller reactor vessel for he PWR, but this size advantage is offset by the PWR steam generators, pressurizers, heat exchangers and higher pressure. The smaller core also results in a smaller inventory of UO_2 fuel which has an advantageous effect on fuel cycle costs. Some of the fuel weight advantage of the PWR is offset by the higher cost of the higher U-235 enrichment required in the smaller core.

Core mechanical characteristics are also summarized in Table 10. Although both cores contain about 50,000 rods, the rods are assembled into different bundle sizes, with the PWR core consisting of 193 bundles each containing 264 rods, and the BWR, 848 bundles of 63 rods each. The number of control elements per core is also significantly different.

A section showing four fuel bundles of a Westinghouse PWR core is presented in Figure 24. Each fuel bundle consists of an array of 17 x 17 (289) tube positions with 264 of these positions occupied by fuel rods and 22 by open tubes. In 69 bundle positions throughout the core, the open tubes in the bundles serve as guide thimbles for 22 individual control rods which are joined at the top into a spider assembly to form a cluster control element (shown in dotted lines in Figure 24). The open tubes which normally contain water, are distributed throughout the bundle to promote uniform heat generation among all rods in the bundle. This permits use of the same U-235 enrichment in all rods of the bundle. In contrast, it is recalled (Figure 12) that in the BWR core, the control elements are cruciform blades inserted between fuel bundles and each bundle requires varying fuel rod enrichments to improve uniformity of heat generation. The cruciform control elements arrangement emphasizes mechanical reliability at the expense of increased complexity in fuel enrichments.

Another major difference between the PWR and BWR cores is in the method of

WESTINGHOUSE 17 x 17 FUEL ASSEMBLY CROSS SECTION

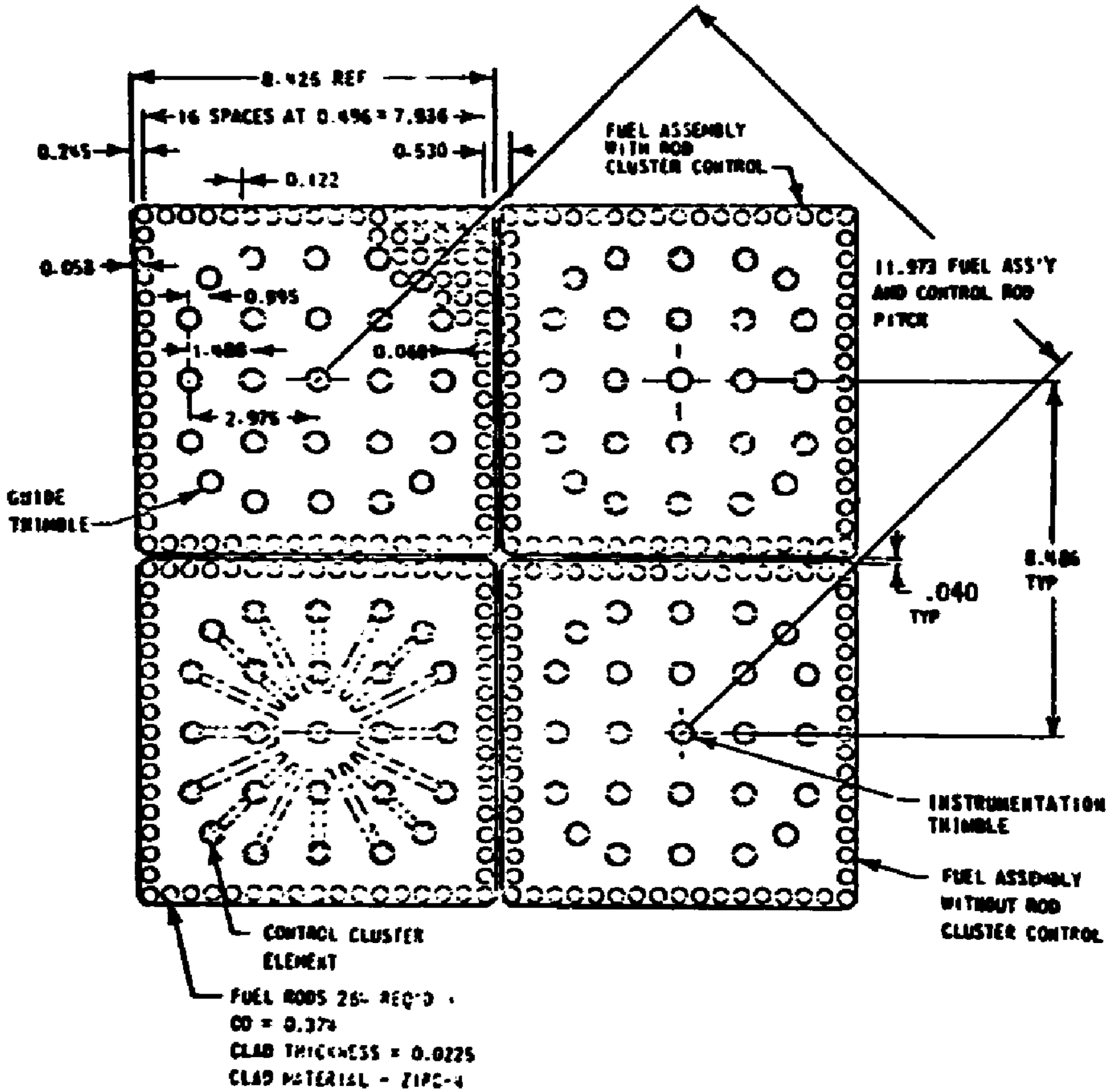

Figure 24

providing supplementary nuclear control. In the PWR, neutron absorber material, a soluble boron compound, is injected into the reactor recirculating water for reactor control. The reactor is started up and shut down using the mechanical cluster control elements. During continuous steady state operation the cluster elements can be withdrawn and control provided by the soluble poison in the reactor recirculating water. This system is too slow to respond to short term load changes which have to be accomplished through control rod movement. Slow changes in power level or fuel depletion compensation are accomplished with the soluble poison system.

For a sudden 10% power demand the PWR steam generators can provide steam from their steam drums, which serve as reservoirs, for a transient demand. The transient is followed by increased reactor output achieved by control rod movements. Response to sudden load demand is accomplished in the BWR by flow control in the flow control power range. The BWR speed of response is slower over the initial 10% load change but faster over the total range of flow control. This is illustrated in Figure 25. An additional system is being developed for the BWR to provide capability for frequency regulation, which requires fast response for several per cent load change. This will be accomplished by using the flow control system to regulate reactor steam pressure, in place of the current system in which the turbine controls reactor pressure.

Historical Evolution

The relative historical evolution of BWR and PWR reactor characteristics is illustrated in Figures 26, 27 and 28. The unit size history, summarized in Figure 26, is self-explanatory. The changes in core power density are shown in Figure 27, and the changes in fuel thermal duty (KW/ft) in Figure 28. The increases in BWR fuel thermal duty in the late 1960's occurred with the 7 x 7 fuel array. The decrease in BWR/6 over BWR/5 in 1971 was accomplished by changing the fuel array from 7 x 7 to 8 x 8. Westinghouse followed by changing their fuel array from 15 x 15 to 17 x 17.

LOAD RESPONSES

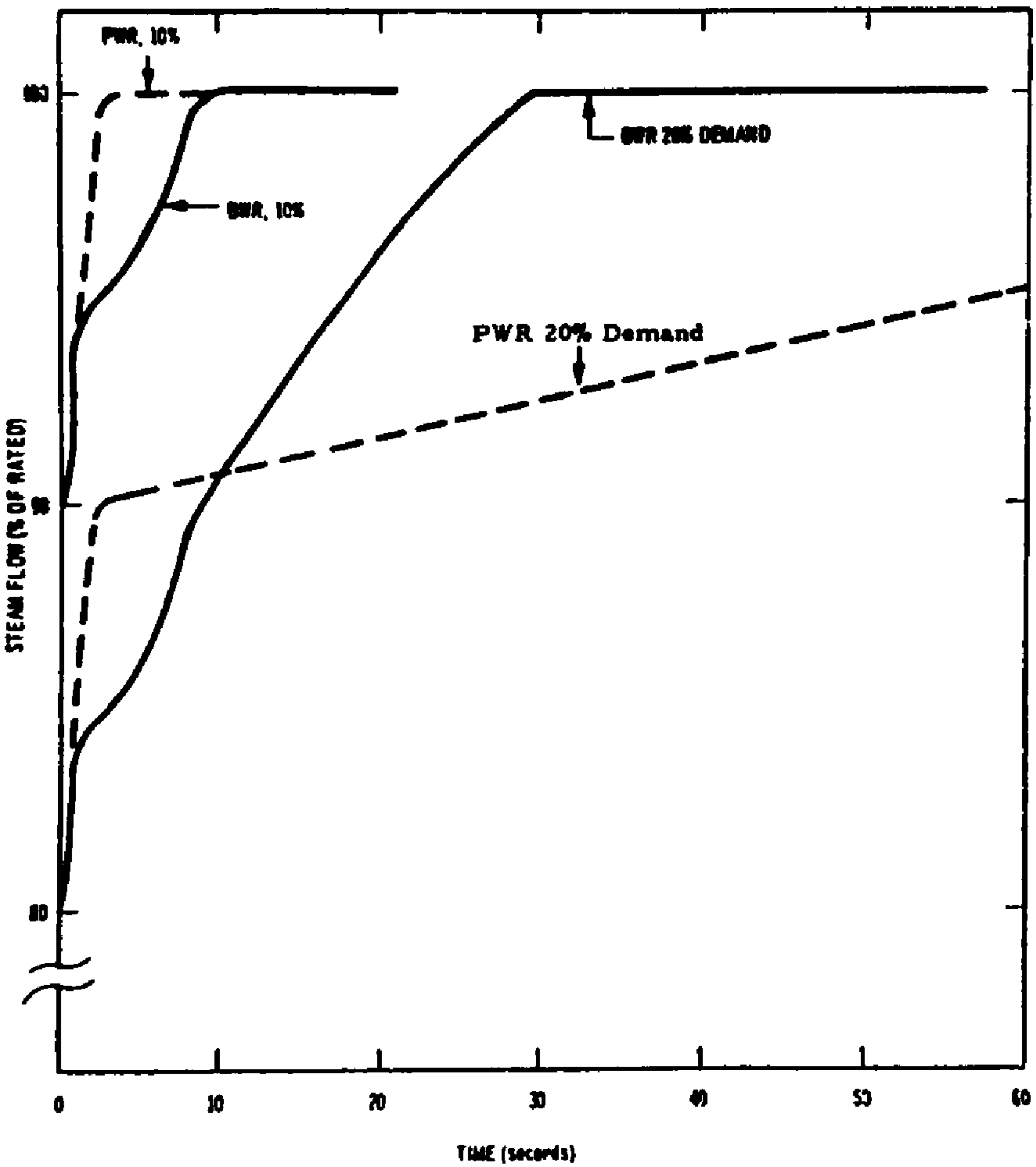

Figure 25

HISTORICAL AND FORECASTED
GROWTH OF LWR PLANT SIZES

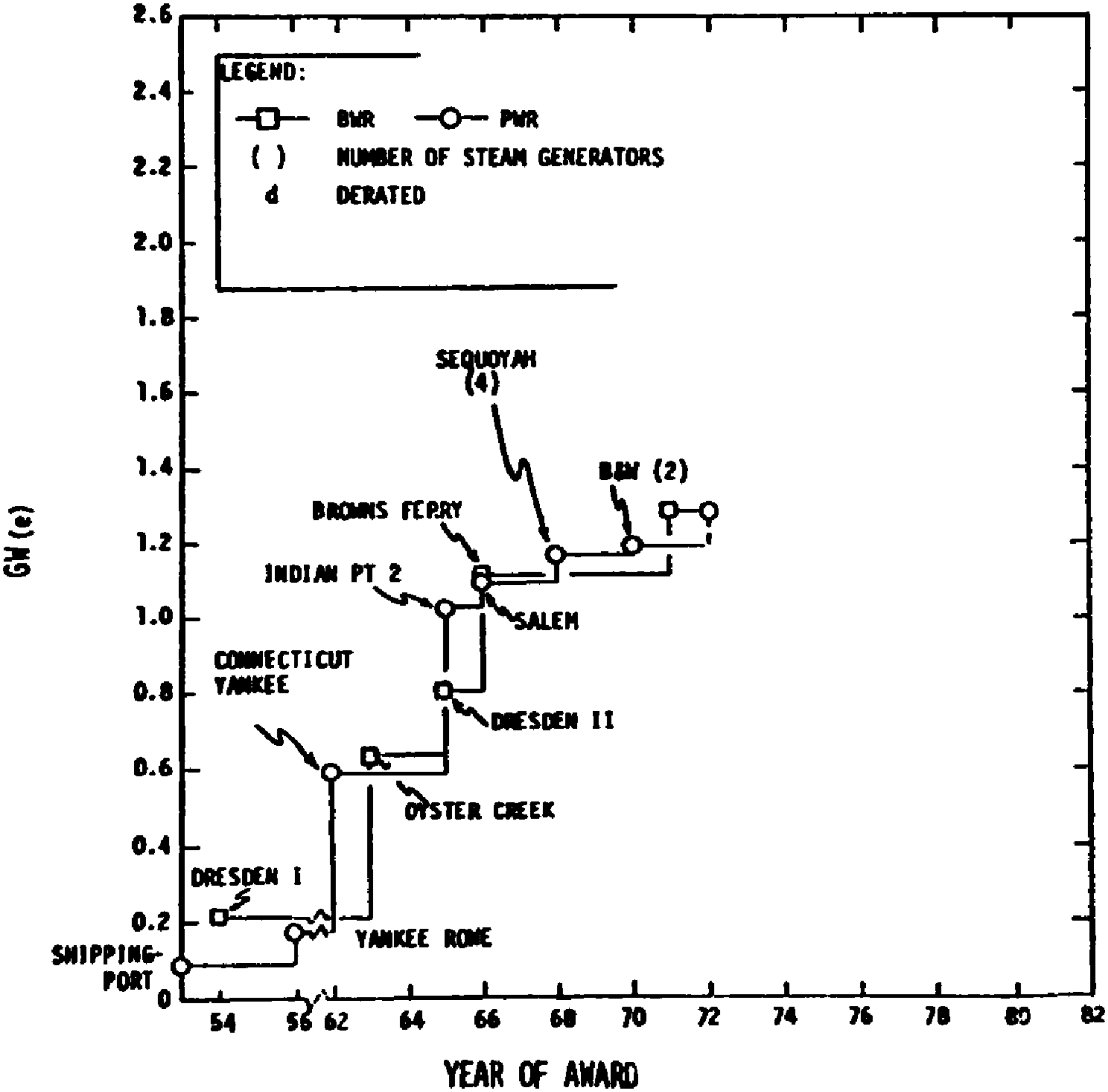

Figure 26

POWER DENSITY TREND

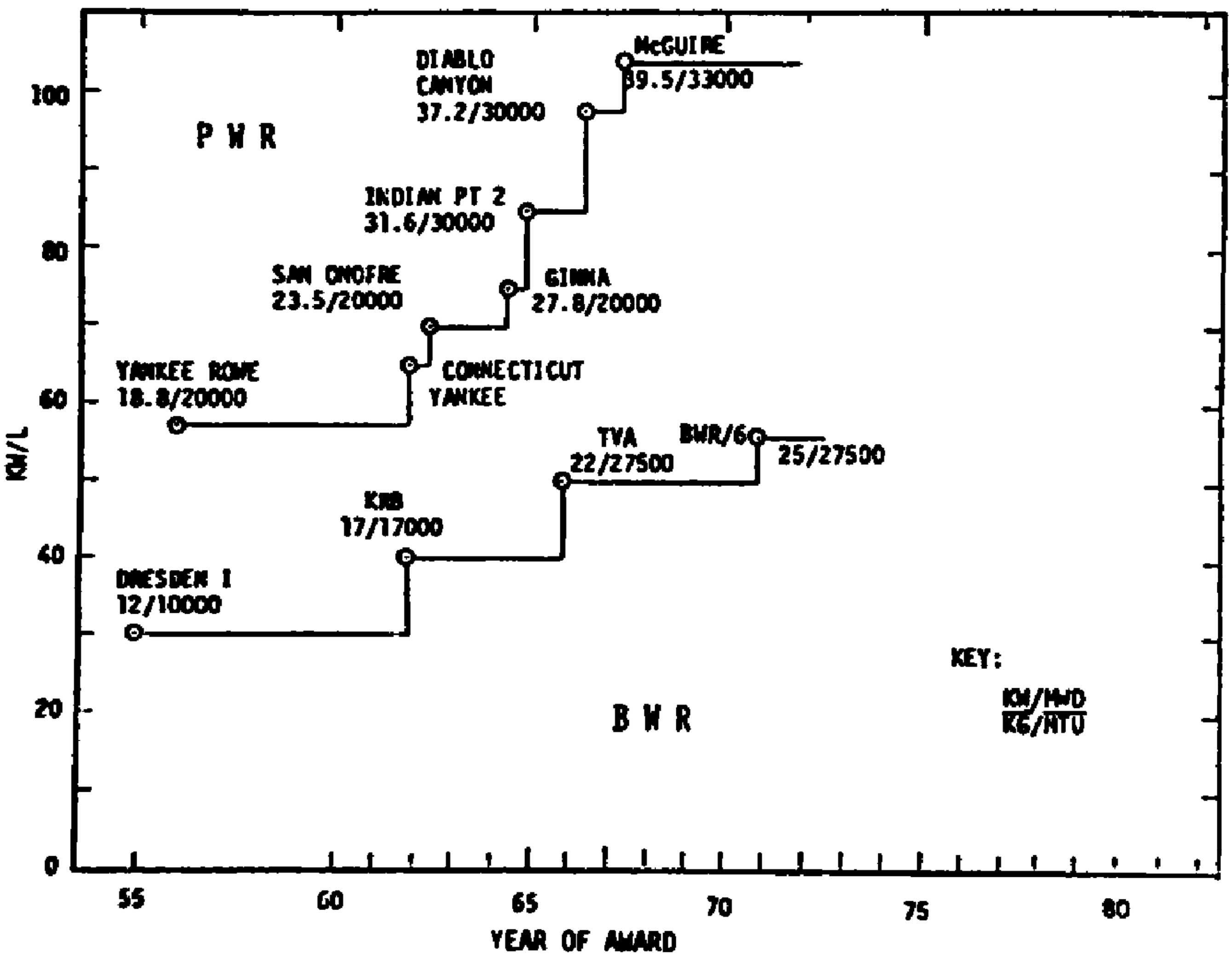

Figure 27

DESIGN AND ACTUAL PEAK POWER

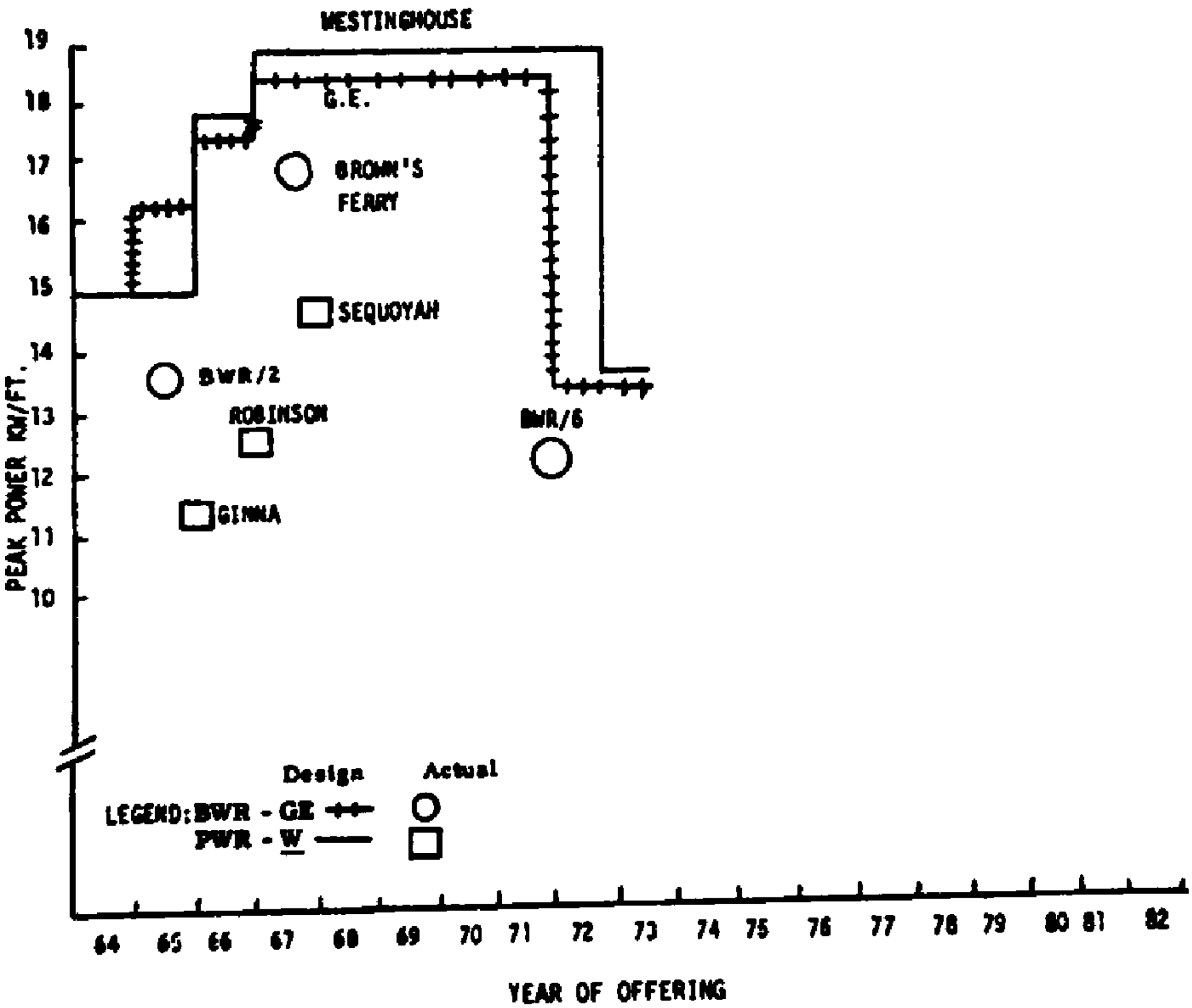

Figure 28

APPENDIX B

This section contains brief biographical sketches of Nuclear Reactor Study Group members.

Dr. C. E. Reed, Chairman

Dr. Reed holds chemical engineering degrees from Case Institute of Technology (BS, 1934) and M.I.T. (ScD, 1937) where he taught until joining the Research Laboratory as a research associate in 1942. After engineering management assignments in the Chemical Division, he became General Manager of the Silicone Products Department in 1952, of the Metallurgical Products Department in 1953 and of the Division in 1959. He was made Vice President in 1962, Vice President and Group Executive, Components and Materials Group in 1968, and Senior Vice President in 1971. He is a Fellow of the American Institute of Engineers and a member of the National Academy of Engineering.

W. W. Beardslee

A graduate of the U.S. Naval Academy, Mr. Beardslee completed a post graduate course in aeronautical engineering at the California Institute of Technology. After engineering jobs with Oldsmobile Motor Works and Lockheed Aircraft, he joined Curtiss-Wright Corporation as Assistant to the Vice President, Airplane Division. Mr. Beardslee joined General Electric in 1947 as Assistant Manager of Manufacturing of GE X-Ray Corporation. In 1948 he transferred to the Manufacturing Policy Division, and was appointed Manager-Manufacturing Engineering Service with the title of Manager-Engineering and Manufacturing Engineering Consulting Service of Corporate Consulting Services. He retired from that position as of May 31, 1975.

R. H. Beaton

Dr. Beaton, a graduate of Northeastern University, received a D.Eng. from Yale and an Honorary D.Sc. from Northeastern. His extensive industry career began at the duPont Company where he was involved in the Manhattan Project. He joined the General Electric Company in 1946 and has held management positions in the research and development, design, construction, and manufacture of nuclear plants, nuclear weapons, weather and communication satellites and other spacecraft systems, as well as sophisticated electronic systems. He was in charge of GE's Apollo Program and received a NASA Public Service Award for assisting in the successful lunar landing of Apollo 11. He assumed his present position, Vice President and General Manager, Energy Systems and Technology Division, in May 1974, and is responsible for the evaluation, planning, integration, and execution of all advanced energy-related research and development activities of the Company.

A. M. Bueche

Dr. Bueche received his Bachelor of Science degree in chemistry from the University of Michigan in 1943 and was awarded his Ph.D. in physical chemistry from Cornell University in 1947. After serving as a research associate at Cornell, he joined the staff of the General Electric Research Laboratory as a physical chemist in 1950. Following a series of managerial assignments, he was named to his present position, Vice President-Research and Development, in 1965. He was elected to membership in the National Academy of Sciences and the National Academy of Engineering. He also serves on advisory committees to the National Science Foundation, the National Research Council, the National Bureau of Standards, the National Governor's Council on Science and Technology and the U.S. Air Force.

Karl Cohen

After receiving his B.A., M.A., and Ph.D. in Chemistry from Columbia University, Dr. Cohen began his career as assistant to Professor H. C. Urey at Columbia. In 1940 he was named director of the Theoretical Division of the Manhattan Project. After the war he had successive positions as physicist and advisor on atomic energy matters for Standard Oil Development Company, director of the H. K. Freguson Company (who constructed the Brookhaven reactor and radioactive laboratory) and founded the Walter Kidde Nuclear Laboratories. He joined General Electric in 1955 and was named Manager of Advance Engineering in 1956. After successive managerial positions in 1971 he was named Manager-Nuclear Energy Operational Planning. He assumed his present position, Chief Scientist, Nuclear Energy Division, in 1973. He is a Fellow of the American Nuclear Society, a member of the National Academy of Engineering and has served as a Consultant to the Atomic Energy Commission and other government agencies. He was also elected to the Board of Directors of the U.S. National Committee to the World Energy Conference.

B. R. Doyle

A graduate from the University of California at Berkeley, Mr. Doyle served as a Navy officer before joining GE in 1956 at the Hanford Atomic Products Operation. After three and a half years of rotating assignments under the Financial Management Program and various financial supervisory and internal audit assignments, he joined the Company's Corporate Audit Staff in Schenectady. He spent five and a half years as a Traveling Auditor, and was appointed Audit Administrator in 1966. In 1968 he transferred to the Nuclear Energy Division as Manager-Finance of the Domestic Turnkey Projects Operation and in 1969 he was appointed Manager-Finance of the Turnkey and Overseas Projects Operation. In 1971 Mr. Doyle transferred to corporate headquarters as Consultant-Corporate Accounting and in 1974 was appointed to his present position -- Manager, General Accounting Operation, Corporate Accounting Operation, Corporate Finance Staff.

C. W. Elston

Mr. Elston received his B.S.M.E. from Drexel Institute of Technology in 1937 and joined the Engineering Test Program. He completed five years of the Advanced Course in Engineering. After many assignments relating to turbine design and performance, he was named assistant to the engineer in charge of large steam turbine design in 1947, and in 1949 became Manager-Turbine Engineering, Large Steam Turbine-Generator Department. Following assignments as Manager-Engineering, Gas Turbine Department, General Manager, Gas Turbine Department and General Manager, Large Steam Turbine-Generator Department, he was named Manager-Steam Turbine-Generator Operational Planning in 1966, his present assignment.

L. C. Harriott

A graduate of New York University (MSEE), Mr. Harriott joined General Electric on the Test Program in 1941. After military service and teaching at New York University, he worked in the General Engineering Laboratory and Knolls Atomic Power Laboratory. He was named Manager-Information Engineering Laboratory (General Engineering Laboratory) in 1959. In 1963 he joined North American Rockwell Space Division as Director, Information Systems. He returned to Corporate Engineering Staff in 1968 and was appointed Manager-Engineering Consulting Service in 1971. He assumed his present position as Manager-Productivity Programs, Corporate Consulting Services in 1975.

M. C. Hemsworth

Mr. Hemsworth graduated with a B.S.M.E. from the University of Nebraska in 1940 and was hired by General Electric on The Test Engineering Program. He joined the Company's Aircraft Engine Group in 1941 in Test Facilities Design and Development Engineering. In 1948 he became Manager of Test Facilities Design and Operation in the Flight Propulsion Plant in Evendale, Ohio. In 1951 he began a series of managerial assignments for the development of advanced engines and in 1957 was assigned responsibility for Engine Component Design for the Large Jet Engine Department. In 1960 he became Manager of Engineering and later Advanced Design Engineering in the Small Aircraft Engine Department and in 1964 returned to Evendale as Manager of TF-39 Engineering and in 1968 was named Manager of TF-39/CF-6/LM-2500 Engineering Operation. In 1971 he became Chief Engineer, Group Engineering Division.

T. H. Lee

Dr. Lee graduated from National Chiao Tung University (BSME) in 1946. He came to the United States and joined Andersen, Meyer and Company. He became a Test Engineer in Schenectady in 1948 and studied at Union College and R.P.I. (MSEE, 1950, Ph.D., 1954). He held various positions in engineering and research until 1955 when he started General Electric's vacuum interrupter development program. In 1959 he was appointed Manager of Engineering Research at the Philadelphia Laboratory Operation, and in 1967 he was appointed Manager of the entire Laboratory Operation. In 1971 he was made Manager of the Power Delivery Group's Technical Resources Operation, a position he held until 1974 when he assumed his present position as Manager-Group Strategic Planning Operation for the Power Generation Business Group. Dr. Lee is a Fellow of the Institute of Electrical and Electronic Engineers, a member of the American Vacuum Society, a member of the American Physical Society and a member of the National Academy of Engineering.

M. C. Leverett

After receiving his DSc. in Chemical Engineering from M.I.T., Dr. Leverett held a number of research and managerial positions with various companies before joining General Electric in 1951 as Engineering Manager in the Aircraft Nuclear Propulsion Department; in 1956 became Manager-Development Laboratories. He transferred to Hanford as Consulting Engineer, and served as Manager-Research and Engineering of Hanford Atomic Products Department until 1967 when he transferred to Nuclear Energy Division as Manager-Division Safety. His present assignment is Manager-Nuclear Safety Assurance, Nuclear Energy Division. Dr. Leverett is a Fellow of the American Nuclear Society, a member of the American Institute of Chemical Engineers, the American Association for the Advancement of Science, and the American Physical Society.

J. F. McAllister

After graduation from the University of Pennsylvania (BA in Physics and BSEE) Mr. McAllister joined General Electric on the Test Program and participated both as student and instructor in its Advanced Engineering Program. He subsequently served in a variety of engineering assignments and in 1956 was named Manager-Engineering of the Television Receiver Department. Before transferring to Corporate Staff he served as General Manager of the Power Tube Department. In 1963 he was named Consultant-Advanced Development and also served as Manager-Product Safety and Reliability. He was appointed Staff Executive, Product Quality Staff, Corporate Studies and Programs in 1971.

S. Neal

A graduate of the University of Kentucky with a B.S.M.E., Mr. Neal joined GE in 1935, spent three years on the Test Engineering Program and then joined the Large Steam Turbine-Generator Department where he served in a variety of design and engineering posts. In 1951 he transferred to the Small Steam Turbine Department in Fitchburg as Manager of Supercharger Engineering. Two years later he was named Manager of the Design Engineering Program. In 1960 he joined the General Engineering Laboratory as Manager of Design Engineering for Magneto-hydrodynamics. In 1961 he became head of the Mechanical Engineering Laboratory and in 1971 he was appointed Manager of Research and Development Planning and Communication. Mr. Neal is a Fellow of the American Society of Mechanical Engineers and a member of the society of Naval Architects and Marine Engineers.

M. P. O'Brien

Dr. O'Brien graduated from Massachusetts Institute of Technology in 1925, received a Doctor of Science degree from Northwestern University in 1959 and a Doctor of Engineering degree from Purdue University in 1961. He has been a Consultant (on a retainer basis) to General Electric Company since 1949, he is Dean of Engineering Emeritus and Professor of Engineering Emeritus at the University of California, Berkley, and is a member of the National Academy of Engineering.

A. W. Robinson

After receiving his M.S.E.E. in Electric Power Engineering from M.I.T., Mr. Robinson joined GE in 1940. He joined the Guided Missiles Operation in 1945, was made Manager-Guidance Engineering in 1953, Manager-Systems Engineering in 1955, Weapon Systems Engineer in 1956 and Manager of Future Growth Study, Advanced Systems Engineering and Space Business Development in 1960. After two years with the Office of the Secretary of Defense as Assistant Director of Defense Research and Engineering, he returned to the Company in 1965 as Manager-Advanced Requirements Planning Operation, Missile and Space Division. In this assignment he was responsible for development and construction of the radio isotope power supply landed on the moon in the Apollo program. From 1968 to 1969 he was Manager-Aerospace Resources Analysis, from 1969 to 1970 he was Manager-Group Integration Operation in the Information Systems Group. He was made Staff Executive in 1971.

E. Schmidt

A graduate of M.I.T. with a M.Sc. degree in thermal dynamics, Mr. Schmidt, following a stint in the U.S. Navy where he served as gunnery officer and radar officer, joined American Cyanamid, working there from 1947 to 1948. He worked at Oak Ridge National Laboratory from 1948 to 1951, leaving to join Convair. In 1952 he joined General Electric, working with Messrs. H. Paige, T. O. Paine and E. E. Hood. After working at NASA from 1967 to 1970, Mr. Schmidt became a Consultant and has worked with various General Electric businesses since that time.

H. E. Stone

Mr. Stone graduated from the University of Buffalo with a B.S. degree in Mechanical Engineering; he also obtained an M.S. degree in Engineering from Union College. He joined GE in 1948 on the Engineering Test Program and held several training assignments in Aircraft Gas Turbines and High Speed Bearing Testing. He joined the Knolls Atomic Power Laboratory in 1949 and held several supervisory positions in reactor safeguards and shielding of nuclear submarines. In 1958 he was appointed to a managerial position related to Power Plant Engineering and held this and similar positions until 1960. From 1960 to 1962 he was Manager of Plant Analysis and Mechanical Systems on an advanced submarine plant. In 1962 he became Project Manager In 1968 he was made General Manager of the Knolls Atomic Power Laboratory. After six years he was appointed Manager, Nuclear Energy Operational Planning at the Nuclear Energy Division. In May 1975 he was appointed to his present position as General Manager, Boiling Water Reactor Systems Department. Mr. Stone is a member of ASME and ANS.

G. Thornton

A native of Norway, Mr. Thornton holds a B.S. in Engineering from Tufts and a M.A. in Physics from Harvard. Before joining GE in 1951 he held a variety of positions including Nuclear Weapons Specialist, Manhattan Project at Los Alamos and Tinian Island, Flight Test Engineer for the University of California in the 1946 Bikini tests and Design Engineer at Fairchild Nuclear Aircraft Project in Oak Ridge. His General Electric assignments have included Supervisor-Nuclear Analysis, Aircraft Nuclear Propulsion Department, Manager of: HTRE1 Reactor Project, XNJ140 Nuclear Turbojet Project, Design and Projects Section and Advance Engineering Section. He was appointed Manager-Engineering, Nuclear Materials and Propulsion Operation in 1961 and joined Corporate Engineering in 1965. His present position is Consultant-Engineering Project Reviews.

A. Weinberg

Dr. Weinberg received his S.B., S.M. and Ph.D. degrees in Physics from the University of Chicago. One of the first members of the University of Chicago's wartime Metallurgical Laboratory from early 1942 until 1945, he then joined the Oak Ridge National Laboratory where he served as Director of the Physics Division, as Research Director, and as Director of the Laboratory from 1955 to 1973. As a member of the team of theoretical physicists at the Chicago Metallurgical Laboratory, he helped design the first large nuclear power reactors. For his work in the development of nuclear reactors, Dr. Weinberg shared the Atoms for Peace Award in 1960. Dr. Weinberg is a member of the National Academy of Sciences, the National Academy of Engineering, the American Academy of Arts and Sciences, a Fellow of the American Nuclear Society, Fellow-American Physical Society, member American Association for the Advancement of Science and the Council of the United Kingdom Science Policy Foundation. In 1973 he was named to the White House Energy Policy Office R&D Advisory Council. During 1974 he served as Director of the Federal Energy Administration's Office of Energy Research and Development.

J. F. Young

Mr. Young joined the Test Program and Advanced Engineering Course in 1937, the year he received a B.S.M.E. from Lafayette College. In 1949 he was named Manager-Engineering, Specialty Refrigeration Products Department and Engineering Services' Consultant-Energy Conversion in 1954. He was appointed General Manager of the General Engineering Laboratory in 1958, and of Electric Utility Engineering Operation in 1960. He became General Manager of the Nuclear Energy Division in 1963. He was elected Vice President in 1965 and named Vice President-Engineering in 1966 where he remained until appointed to his present position as Vice President and Staff Executive, Technical Resources Staff, Technology Planning and Development. Mr. Young is a member of the National Academy of Engineering, the International Electrical and Electronic Engineers (IEEE) and the National Society of Professional Engineers. He is a Fellow in the American Society of Mechanical Engineers.