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JAN 3 1990

Docket No. 50-271

Mr. L. A. Tremblay
Licensing Engineer
Vermont Yankee Nuclear Power Corporation
580 Main Street
Boston, Massachusetts 01740-1398

Dear Mr. Tremblay:

Enclosed for your information are copies of a Director's Decision, letter of transmittal, and Federal Register notice issued by the Director, Office of Nuclear Reactor Regulation (Director) in response to a Petition filed under 10 CFR 2.206 of the Commission's regulations. The Petition was filed by Ms. Anna Harlowe on behalf of the Ecology Center of Southern California.

The petitioner requested that the NRC fix or close all nuclear power reactors designed by General Electric Company. As discussed in the enclosed Director's Decision, the Petitioner's request under 10 CFR 2.206 has been denied.

The petitioner also expressed concern that GE is pursuing a "standardized" Advanced Boiling Water Reactor design which petitioner alleges fails to address many of the shortcomings identified by GE's own engineers as far back as the 1975 Reed Report. The petitioner was informed that the staff has not yet completed its safety evaluation for advanced boiling water reactor designs, nor has any utility applied for a license to build or operate an advanced boiling water reactor.

Sincerely,

Original signed by:

Richard H. Wessman, Director
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Letter dated December 4, 1989
   to Ms. Anna Harlowe
2. Director's Decision dated
   December 4, 1989
3. Federal Register Notice
   dated December 4, 1989

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Adjudicatory File (2)
Atomic Safety and Licensing Board
Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 4, 1989

Ms. Anna Harlowe
Issues Coordinator
Ecology Center of Southern California
Post Office Box 35473
Los Angeles, California 90035

Dear Ms. Harlowe:

This letter further responds to your Petition of March 8, 1989, requesting that the NRC fix or close all nuclear power reactors designed by General Electric Company (GE).

As bases for this request, you allege that (1) in 1972, a member of the NRC staff recommended that GE-designed reactors be banned in the United States; (2) in 1975, GE engineers generated the "Reed Report" that detailed dozens of safety and economic problems with GE-designed reactors and recommended that GE stop selling those reactors; (3) in 1986, an NRC official admitted that 24 GE reactors with Mark I containments had a 90 percent chance of failure in a nuclear accident; (4) in 1987, an NRC task force confirmed that Mark I containments were virtually certain to fail in an accident; (5) according to NRC safety studies, Mark II reactors have many possible scenarios for early containment failures; and (6) Mark II designs, on which the Reed Report focused, have dozens of safety and economic problems and have suffered massive cost overruns during construction as a result of design problems.

On June 5, 1989, I informed you that your request was being treated under 10 CFR 2.206 of the Commission's regulations and that a formal decision would be issued within a reasonable time.

For the reasons set forth in the enclosed Director's Decision under 10 CFR 2.206, your Petition has been denied. A copy of the Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c). The Decision will constitute final action of the Commission 25 days after the date of issuance unless the Commission, on its own motion, institutes a review of the Decision within that time. I have also enclosed a copy of a notice that is being filed with the Office of the Federal Register for publication.

Your letter also expressed concern that GE is pursuing a "standardized" Advanced Boiling Water Reactor (ABWR) design which you allege fails to address many of the shortcomings identified by GE's own engineers as far back as the 1975 Reed Report.

~~8912140069~~ 2pp

The Reed Report was a self-critical study performed by General Electric intended as a product improvement study to enhance the availability and performance of GE reactors. In February 1976, two NRC staff members reviewed a copy of the report in GE's Washington, D.C., offices and determined that the report (1) did not identify any new safety concerns, and (2) did not indicate that GE had failed to report any significant safety concerns to the NRC.

In June 1987, the NRC established a special task group to reevaluate the issues raised in the Reed Report, taking into account the advances in nuclear power technology and the many reactor years of operational experience in the 12 years since the Reed Report was written. In NUREG-1285, the special task group drew the following conclusions:

(1) The Reed Report does not identify any matters that would support a need to curtail the operation of any GE boiling water reactor currently licensed.

(2) The Reed Report does not identify any new safety issues of which the NRC staff was unaware.

(3) Although certain issues addressed by the Reed Report are still being studied by the NRC and the industry, there is a basis for permitting continued plant operations while those issues are being resolved.

These conclusions are consistent with the conclusions of our February 1976 review. Additional information, including the history of the Reed Report and the topics discussed in the Reed Report, is contained in NUREG-1285, a copy of which is enclosed for your information.

The staff has not yet completed its safety evaluation for advanced boiling water reactor designs. Nor has any utility applied for a license to build or operate an ABWR. However, the staff is implementing 10 CFR Part 52 and the Severe Accident Policy Statement and the Safety Goal Policy Statement in its review and evaluation of the severe accident issues that are being addressed in advanced light water reactor (LWR) design certification applications as well as in the conceptual design documentation on non-LWR designs. The staff's conclusions regarding these matters will be in accordance with the Commission's policy that future designs for nuclear power plants should reduce the risk from severe accidents.

Sincerely,

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Director's Decision
2. Federal Register Notice
3. NUREG-1285, "Reed Report"

NUREG-1285

NRC Staff Evaluation of the General Electric Company Nuclear Reactor Study ("Reed Report")

Manuscript Completed: July 1987
Date Published: July 1987

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Encl 3

NUREG-1285

NRC Staff Evaluation of the General Electric Company Nuclear Reactor Study ("Reed Report")

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-293, et al.*

BOSTON EDISON COMPANY, et al.*

(Pilgrim Nuclear Power Station, et al.)*

ISSUANCE OF DIRECTOR'S DECISION UNDER 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation (NRR), has issued a Director's Decision concerning a Petition dated March 8, 1989, filed by Ms. Anna Harlowe, Issues Coordinator, on behalf of the Ecology Center of Southern California. The Petition asked the Director, NRR, to take action to relieve what the Petitioner alleged to be undue risks to the public health and safety posed by the containment design of boiling water reactors (BWRs), as revealed by various NRC staff members' statements, published studies, and by the 1975 General Electric "Reed Report." The specific relief requested was to order all BWR licensees to "fix" or close all BWR reactors. Ms. Harlowe gave as grounds for the Petition that (1) in 1972, a member of the NRC staff recommended that GE-designed reactors be banned in the United States; (2) in 1975, GE engineers generated the "Reed Report" that detailed dozens of safety and economic problems with GE-designed reactors and recommended that GE stop selling those reactors; (3) in 1986, an NRC official admitted that 24 GE reactors with Mark I containments had a 90 percent chance of failure in a nuclear accident; (4) in 1987, an NRC task force confirmed that Mark I containments were virtually certain to fail in an accident; (5) according to NRC safety studies, Mark II reactors have many possible scenarios for early containment

~~8912140071~~ 63 pp

PENNSYLVANIA POWER & LIGHT CO. (Susquehanna Steam Electric Station, Units 1 and 2, Docket Nos. 50-387 and 50-388)

PHILADELPHIA ELECTRIC CO. (Peach Bottom Atomic Power Station, Units 2 and 3, Docket Nos. 50-277 and 50-278), (Limerick Generating Station, Unit 1, Docket No. 50-352)

POWER AUTHORITY OF THE STATE OF NEW YORK (James A. Fitzpatrick Nuclear Power Plant, Docket No. 50-333)

PUBLIC SERVICE ELECTRIC & GAS CO. (Hope Creek Nuclear Station, Docket No. 50-354)

TENNESSEE VALLEY AUTHORITY (Browns Ferry Nuclear Power Station, Units 1, 2, and 3, Docket Nos. 50-259, 50-260, and 50-296)

VERMONT YANKEE NUCLEAR POWER CORP. (Vermont Yankee Nuclear Power Station, Docket No. 50-271)

WASHINGTON PUBLIC POWER SUPPLY SYSTEM (WNP Unit 2, Docket No. 50-397)

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. INTRODUCTION

On March 8, 1989, Ms. Anna Harlowe, on behalf of the Ecology Center of Southern California (Petitioner), filed a Petition in accordance with 10 CFR 2.206 with the Nuclear Regulatory Commission (NRC). The Petition was referred to the Director, Office of Nuclear Reactor Regulation (NRR), for consideration.

The Petition asked the Director, NRR, to fix or close all nuclear reactors designed by the General Electric Company (GE). As a basis for this request, the Petitioner alleged the following:

(1) In 1972, a member of the NRC staff recommended that GE-designed reactors be banned in the United States; (2) in 1975, GE engineers generated the "Reed Report" that detailed dozens of safety and economic problems with

GE-designed reactors and recommended that GE stop selling those reactors; (3) in 1986, an NRC official admitted that 24 GE reactors with Mark I containments had a 90 percent chance of failure in a nuclear accident; (4) in 1987, an NRC task force confirmed that Mark I containments were virtually certain to fail in an accident; (5) according to NRC safety studies, Mark II reactors have many possible scenarios for early containment failures; and (6) Mark II designs, on which the Reed Report focused, have dozens of safety and economic problems and have suffered massive cost overruns during construction as a result of design problems. Ms. Harlowe also expressed concern that the GE Advanced Boiling Water Reactor design "fails to address many of the shortcomings identified by General Electric's own engineers as far back as the 1975 Reed Report" (Petition, p. 2).

On June 5, 1989, I acknowledged receipt of the Petition. I informed Ms. Harlowe that (1) the Petition would be treated under 10 CFR 2.206 of the Commission's regulations, and (2) appropriate action would be taken within a reasonable amount of time. For reasons discussed below, the Petition is denied.

II. BACKGROUND

The Petitioner alleges that in 1972, a Nuclear Regulatory Commission staff member recommended that GE-type reactors be banned in the United States. It appears that the Petitioner is making reference to a memorandum by Dr. Steven Hanauer dated September 20, 1972. Specifically, Dr. Hanauer was concerned that then recently highlighted safety disadvantages of pressure-suppression containments might outweigh the safety advantages. He recommended that the Atomic Energy Commission (predecessor to the Nuclear Regulatory

Commission) adopt a policy to discourage further use of pressure-suppression containments and that such designs not be accepted for construction permits filed 2 years after the policy would be adopted.

The Petitioner also refers to a 1975 GE document known as the "Reed Report." The Reed Report was a self-critical study performed by GE staff in 1975. It was intended as a product improvement study to enhance the availability and performance of GE's boiling water reactors (BWRs). The report, by its nature a candid self-analysis, was intended for GE's internal use only. It had always been held by GE to be "proprietary," and thus not subject to public disclosure. The principal author of the report was Dr. Charles E. Reed, a Senior Vice President of GE. Contributors included technical and professional personnel from a variety of GE departments. Their efforts resulted in the Nuclear Reactor Study, referred to today as the Reed Report, and a set of 10 subtask reports that provided the detailed technical information used to develop the Nuclear Reactor Study.

The Reed Report addressed operating BWRs and the design of future GE products and services in the nuclear field. For reactors in operation at the time, the report discussed ways to improve a plant's availability and its electrical generating capacity factor through improvements in plant hardware and also in service, fuel, equipment, and operating procedures. For future reactors, the report considered GE's then-new BWR design, the BWR-6, and discussed problems regarding final design details, licensing, and full-power operation of BWR-6 plants.

The Petitioner also refers to an early 1986 statement by a senior NRC official that the containment vessels on 24 GE reactors have a 90 percent chance of failure in a nuclear accident. Ms. Harlowe most likely is referring radiation release, on 24 GE reactors have a 90 percent chance of failure in a nuclear accident," and (3) that "in late 1987, a Nuclear Regulatory Commission task force confirmed the failure rate of these 24 'mark I' reactors, saying that their containments are virtually certain to fail in an accident." 1/

Petitioner does not provide any information of which the staff was unaware. In fact, similar, more specific and detailed concerns relative to alleged Mark I containment design deficiencies were previously addressed in Interim Director's Decision 87-14 concerning the Pilgrim Nuclear Power Plant on August 21, 1987. 2/ As stated in that Decision, containment structures are an integral part of the U.S. reactor designs in that they form one part of a structured, tiered approach to public safety known as defense in depth. Concisely put, defense in depth is the process implemented by the AEC (later NRC) to ensure that multiple levels of assurance and safety exist to minimize the risk to the public of exposure to ionizing radiation resulting from equipment failures, transients and postulated accidents.

A primary level of assurance are those activities to ensure that the plant is designed and constructed to high quality standards. The Commission's regulations require plant design to satisfy certain standards, as specified in the General Design Criteria (GDC) in 10 CFR Part 50 Appendix A. Specific information is provided in the NRC's Standard Review Plan (SRP) which details acceptable methods for complying with the requirements established in the GDC.

1/ Ecology Center of Southern California Petition at 1.
2/ Boston Edison Co. (Pilgrim Nuclear Generating Station), DD-87-14, 27 N.R.C. 87 (1987).

discussion of the methods used in the risk analyses, additional discussion on specific technical issues important in the analyses, and responses to comments received on the earlier draft.

Petitioner also alleges that Mark II reactors (eight of which are operating) still have many possible scenarios for early containment failure according to NRC safety studies. Petitioner is most likely referring to studies conducted as part of the Containment Performance Improvement, Individual Plant Examinations, and Severe Accident Policy programs. NRC studies are ongoing and not yet complete, but the NRC has made preliminary specific assessments of Mark II containment performance.

Lastly, Petitioner alleges that "Mark II reactors on which the 1975 General Electric Reed Report was primarily focused have the aforementioned 'dozens of safety and economic problems,' and have suffered massive cost overruns during construction as a result of design problems." It is believed, based on the staff's review of the Reed Report, that Petitioner is referring to Mark III reactors, not Mark II reactors, and it is on this premise that my discussion is based.

III. DISCUSSION

A. Mark I Containment Concerns

Petitioner's alleged "facts" that she wishes placed under consideration for relief contain three items that appear to be directed at the GE Mark I containment design. These are (1) that "in 1972 a Federal Nuclear Regulatory Commission [sic] staff member recommended that General Electric-type reactors be banned in the United States," (2) that in 1986, "a top Nuclear Regulatory Commission official admitted that the containment vessels, the last barrier to to a quote from Harold Denton in Inside NRC, Vol. 8, No. 12, June 9, 1986, wherein Mr. Denton was quoted as saying: "I don't have the same warm feeling about GE containment that I do about the larger dry containments. There has been a lot of work done on those containments, but Mark I containments, especially being smaller with lower design pressure - and in spite of the suppression pool - if you look (at the) WASH 1400 reg safety study, you'll find something like a 90% probability of that containment failing."

The Petitioner also alleges that a late 1987 finding of an NRC task force confirmed that the failure rate of these 24 Mark I reactors is such that their containments are "virtually certain" to fail in an accident. Although it is not clear which specific study the Petitioner is referring to, it is presumed that she refers to the "Reactor Risk Reference Document," Draft NUREG-1150, dated February 1987. NUREG-1150 estimated the probability of total core damage frequency for the Peach Bottom reactor, which is similar in design to the typical Mark I reactor, to be 8.2 X 10^{-6} per reactor year. However, NUREG-1150 went further and evaluated Mark I and other reactor design risk scenarios given that a severe (core-melt) accident (low probability event) had already taken place. Accounting for comments received from the public and three formal peer reviews, a second draft for peer review titled "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, Summary Report, Second Draft for Peer Review," NUREG-1150, was issued in June 1989 in two volumes. Volume 1 provides summaries of the risk analysis results for the five plants studied, perspectives on these results, and a discussion of the role of these risk analyses in the NRC staff's severe accident regulatory program. Volume 2 provides a more detailed

Early in the development of commercial nuclear power, it was recognized that these complex systems could not be expected to be immune from various failures and malfunctions, regardless of the quality of design, construction, and operation. Therefore, a further level of defense was established in that the plants were required to be designed to cope successfully with various equipment failures, transients, and postulated accidents. The scenarios for postulated accidents, to which all plants are designed to adequately respond, are known as design basis accidents and are detailed in the NRC's Standard Review Plan, which is used to evaluate the design of each nuclear power plant before the granting of a construction permit or an operating license.

Design basis accidents were chosen to represent a wide spectrum of plant problems, some of which were expected to be experienced in the plant's lifetime (such as failure of power systems), as well as events considered to be quite infrequent (such as major ruptures of piping systems) and not expected to occur in the plant's lifetime.

The NRC Standard Review Plan also identifies acceptable plant protection standards for each postulated plant accident. The requirements and capabilities of plant safety systems necessary to prevent these design basis accidents from leading to unacceptable radiological releases are specifically identified. The Standard Review Plan gives acceptance criteria for judging the acceptability of the analytical results in response to these hypothetical scenarios. The resulting plant design incorporates multiple and backup safety systems that will protect the reactor during a design basis accident and a postulated single failure in each system of these various protection devices.

Notwithstanding the above, additional margins are required in the plant design to protect the public even in the event of very unlikely accidents. The reactor containment provides an additional level of safety. Design basis accidents for containment reflect a number of arbitrary accident sequences developed from postulated events. For example, the containment structural design is based upon the effects of a concurrent earthquake and a rupture of major reactor coolant system piping. Concurrently, in order to assess the effectiveness of leaktightness, the safety systems are presumed not to be effective in cooling the reactor core, resulting in the release of fission products from the reactor core. Although the design basis accidents discussed above are allowed to result in some failed fuel (less than 1 percent), they do not result in significant core damage. For the containment design, some independent failures of the protection systems are assumed to occur simultaneously with the occurrence of the accident they are intended to control. Although the purpose of other safety systems is to shut down the reactor fission process and provide emergency cooling water to the reactor core, the containment has a required function of providing an essentially leaktight barrier to "bottle up" any radioactive material released to the containment through any rupture or break in the reactor coolant system. Given the release of the radioactive material and cooling water, the containment is required to retain this material and prevent significant releases to the environment. Consequently, the assessment of containment design adequacy assumes the postulated release of fission products to the containment irrespective of the performance of the core cooling safety systems.

Although design basis accidents are used to determine the adequacy of plant systems' design and performance under postulated accident conditions, severe accidents are analyzed by imposing a set of additional assumptions to further presume that these systems will not work as designed. The containment design basis reflects a combination of parameters incorporating several design basis accidents for structural considerations coupled with an assumed release of radioactive material to containment for assessing leak-tightness.

In summary, the design purpose of the reactor containment is to protect against postulated radioactive releases from hypothetical reactor accidents up to and including major ruptures of reactor coolant piping, where such events resulted in some degree of core damage. These hypothetical events postulated a release of fission products from the reactor core to the reactor coolant system and subsequently into the containment through the pipe break. This was considered one of the less likely, but possible accidents and supplied a straightforward means of providing additional margins for containment design.

The concept of severe nuclear accidents and how these accidents fit within the framework of protection from design basis accidents must also be considered. [3/] For the last several years, the staff has been studying the likelihood and consequences of extremely low probability accidents involving multiple failures that lead to core damage. This class of accidents is

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[3/] Severe accidents are defined as those "in which substantial damage is done to the reactor core, whether or not there are serious offsite consequences." This definition is extracted from the "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 Fed. Reg. 32138, August 8, 1985.

beyond the existing design basis and is generally known as severe accidents. This evaluation was first done comprehensively by the Reactor Safety Study (WASH-1400), which is known as a probabilistic risk assessment (PRA). The types of accidents studied in this evaluation are basically those in which backup safety systems fail, eventually resulting in damage to the nuclear fuel and considerable releases of radioactive material outside the reactor cooling system into the containment. Depending on other failures and containment behavior, significant radiological releases into the environment could conceivably occur. Implicit in these scenarios is the development of a better understanding of containment performance and its failure mechanisms.

More detailed PRA studies have been conducted since the publication of WASH-1400 to better understand the probability of these unlikely events and also to better predict the magnitude of potential radiological releases into the environment, given a containment failure and attendant consequences. Considerable work has also focused on the behavior of reactor containments following a severe accident in which molten reactor fuel could potentially melt through the reactor vessel. Results of such studies have generally confirmed the very low likelihood of such accidents and the relatively low risk to the public even if such very low probability accidents were to occur. Although not originally designed to protect against some of the severe accidents, reactor containments provide considerable protection due to their ability to reduce radiological releases to the public from such accidents. For example, the results of research work indicate that the actual pressure-retaining capability of most containments is well above their original design pressures. Studies also indicate that the massive containment structures may provide substantial retention of radioactive material even if they were to fail following a core melt event. As discussed below, there exists a wide range of uncertainty regarding a Mark I containment's behavior during a core melt accident. A recent study judged the probability of some form of containment failure, assuming a core melt had occurred, to be between 10 and 90 percent.[4/] However, the total core damage frequency for the BWR Mark I design (Peach Bottom) was less than the total core damage frequency of the other four reactor designs studied by generally an order of magnitude or more.

Because of the very complex processes involved in a severe reactor accident, exact predictions of accident consequences are difficult. Considerable research is under way to provide additional information in this area. Results from such studies allow NRC staff to focus attention on areas in which improvements can be made to provide increased levels of safety from these very unlikely events. The purpose of these projects is to conduct hypothetical "what if" studies, to understand ways public risk from nuclear operations can be justifiably reduced. The results of our studies indicate that risks from these severe accidents are very low and do not warrant immediate actions.

Petitioner has expressed concerns that are based on a memorandum written on September 20, 1972, by Dr. S. H. Hanauer, a member of the staff of the Atomic Energy Commission (AEC) (the NRC succeeded the AEC in 1975). These concerns relate to the ability of the Mark I containment to respond

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[4/] The "Reactor Risk Reference Document" - Draft (NUREG-1150), February 1987 adequately to its original design function (i.e., deal with a large loss-of-coolant accident). Dr. Hanauer's memorandum raised seven concerns, all of which centered on the viability of the pressure-suppression containment concept. They relate to steam-bypass susceptibility, valve reliability, lack of adequate testing, and volume limitations causing overcrowding.

When Dr. Hanauer's seven concerns were raised, the staff evaluated each of them to determine whether adequate safety margins were being maintained on existing plants. Subsequently, the NRC staff concluded that Dr. Hanauer's concerns had been properly considered and documented its findings in NUREG-0474, "A Technical Update on Pressure Suppression Type Containments in Use in U.S. Light Water Reactor Nuclear Power Plants," issued in July 1978.

Enclosure A to NUREG-0474 summarizes NRC staff actions related to each of the seven concerns identified in Dr. Hanauer's memorandum of September 20, 1972. A copy of that enclosure is being provided to the Petitioner with this Decision. Each statement of concern was followed by a response that reflected the NRC evaluation. In each case, the response showed that the NRC no longer considered the concern an unresolved safety issue.

It should be noted that although the concerns reflected the views of Dr. Hanauer in September 1972, the NRC response reflected the status of the issues in July 1978. Moreover, by June 1978, Dr. Hanauer had changed his opinion regarding his 1972 concerns, as reflected in a memorandum dated June 20, 1978, in which he stated: "Thus while we may yearn for the greater simplicity of 'dry' containments, the problems of both 'dry' and pressure-suppression containments are solvable, in my opinion, and the design safe, therefore licensable" (NUREG-0474).

Our review of the Petitioner's concern that is based on Dr. Hanauer's memorandum indicates that this concern has been addressed in NUREG-0474. Although various changes have occurred since then, the fundamental safety conclusions stated in NUREG-0474 are essentially unchanged. The most notable of the changes has been the NRC position related to rendering the containment inert. [5/] Since NUREG-0474 was issued, the regulations relating to this issue (10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors") have been revised to require all Mark I and II containments to be rendered inert. The response to Dr. Hanauer's concern (see Item B of Enclosure A to NUREG-0474) indicates that most Mark I containments were already rendered inert. With the issuance of the revised 10 CFR 50.44, the Commission required all Mark I and II containments to be rendered inert to accommodate the degraded core accident. A review of this and other changes made since NUREG-0474 was issued, indicates that in no case have the changes altered the fundamental staff conclusions concerning safety contained in NUREG-0474.

Test programs were initiated by utilities owning Mark I plants as part of a program in response to NRC letters that were transmitted in February and April 1975 to all utilities owning BWR facilities with Mark I design containments. The letters requested that the owners quantify the hydrodynamic and safety-relief valve (SRV) discharge loads and assess the effect of these loads on the containment. (These loads had not been considered during the

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[5/] An inerted containment is one in which oxygen is replaced by enough nitrogen to preclude combustion.

licensing of the individual plants because these loads (including pool swell) were identified in the period 1972 through 1974 as part of the review of the large-scale testing of the Mark III containment system design.)

As a result of these letters from the NRC and in recognition that the evaluation effort would be very similar for all Mark I BWR plants, the utilities formed an ad hoc Mark I Owners Group. The objectives of this Owners Group were to determine the magnitude and significance of these dynamic loads as quickly as possible and to identify actions to resolve any outstanding safety concerns. A series of generic test programs was created to accomplish these objectives.

Since NUREG-0474 was issued in July 1978, the generic test programs related to the Mark I containment design and the NRC assessment of the tests have been completed. The staff evaluation of the generic test programs was reported in NUREG-0661, "Mark I Containment Long Term Program Safety Evaluation Report," issued in July 1980. NUREG-0661 describes and presents staff conclusions regarding the generic techniques for the definition of suppression pool hydrodynamic loads in a Mark I system and the related structural acceptance criteria. As part of the acceptance criteria, the staff required that a plant-specific analysis be submitted by the licensees for all 24 plants having Mark I containments. These analyses have been reviewed and approved by the staff. All modifications proposed by the licensees to satisfy the criteria contained in NUREG-0661 have been completed.

Another of Dr. Hanauer's concerns focused on the safety disadvantages of pressure-suppression containments. This issue is related to the possibility of steam bypassing the suppression pool in BWR pressure-suppression containments, and was designated Generic Issue 61, "SRV Line Break Inside the Wet Well Airspace of Mark I and II Containments." An evaluation of this issue has been completed, and the results were presented in NUREG/CR-4594, "Estimated Safety Significance of Generic Issue 61," which was issued in June 1986. On the basis of these results, the staff concluded that no new requirements were justified and no further study of this safety issue was warranted.

The Petitioner also raises concerns regarding the possibility that the BWR containments might fail in the event of a severe accident. The Petitioner cites various studies regarding a high probability that Mark I containment structures will not stand various severe accident scenarios.

As discussed previously, the NRC views probabilistic risk assessment as a structured method for investigating the likelihood and consequences of reactor accidents considered to have a very low frequency of occurrence. The perceived inability of the Mark I containment to survive a severe accident has been postulated by the Petitioner as a design flaw.

The evaluation of severe accident vulnerability involves three distinct evaluations. The first involves the probability of an accident involving core damage, the second involves the likelihood of containment failure, and the third involves an assessment of the radiological consequences and public doses resulting from the accident. All three issues must be considered in making a determination on the magnitude of severe accident risk and the actions that should prudently be taken to reduce that risk.

The studies that have been conducted emphasize that their results inherently possess large uncertainties. The draft results of NUREG-1150 present the most recent program, whose intent is to accurately reflect the severe accident risk at a number of U.S. nuclear power plants and also to properly reflect the areas of uncertainty. This study included an evaluation for Peach Bottom, a plant quite similar in design to the typical Mark I reactor and containment. The study presented the estimated mean frequency of core damage as approximately 1 chance in 100,000 per year of operation. Another comprehensive risk study conducted by the NRC staff estimated a mean core damage probability of 1 in 10,000 for the Limerick plant.

These results are consistent with NRC's belief that core melt accidents are very unlikely. Draft NUREG-1150 also investigated the probability of early containment failure following a core melt and concluded that our ability to accurately predict the response of a Mark I containment was limited for situations in which it was subjected to the harsh temperature and pressure conditions following a core melt accident. As stated earlier, the report indicated that containment failure probability (for these extremely unlikely events) could likely range from 10 to 90 percent.

These uncertainties are currently the subject of research efforts to better predict the behavior of containments during severe accidents so that a more complete risk perspective can be assembled for guiding our regulatory activities. However, it is important that these uncertainties be properly characterized. They are not identified deficiencies in the BWR Mark I containments, which have been demonstrated to satisfy their design performance requirements. Rather, these uncertainties guide our research investigations, whose goals are to provide improved understanding of very unlikely risk situations at nuclear power facilities. Results from these studies (including high containment failure probabilities) also allow us to calculate public risk estimates assuming that one element of the three in a risk assessment (containment failure) is less favorable.

Even allowing the large uncertainties that result in a high upper value for containment failure, the NUREG-1150 study estimated that the probability of a large reactor accident resulting in one or more early fatalities ranged from 1 in 1 million to 1 in 1 billion. In the event of a severe accident, both the probability of very high radiation exposures and the distances over which such exposures would occur were estimated to be reasonably small. The risk levels for each Mark I reactor would of course depend on its actual core melt probability, containment behavior, the local demography, and could vary somewhat from the results presented in NUREG-1150. The results of this and related studies do, however, support our overall conclusion of low severe accident risk at Mark I reactors. One contributing factor is that the massive reactor containment structure may retain considerable radioactive material following a core melt event even if its pressure boundary fails. In this regard, containment failures include cracks or other phenomena that result in loss of pressure integrity that can result in leaks but should not be viewed solely as catastrophic failure of the containment structure. In the event radioactive material is released inside containment, some of this material dispersed in air, e.g. radioiodine, will be deposited on surfaces inside containment. Even though NRC analysis gives no credit for this phenomenon, deposition of material within containments, even though there may be leakage, will increase the time available to implement effective protective action activities.

Although we believe that severe accident risks are low at operating nuclear plants, to assure that our risk conclusions are applicable to all operating units, a number of programs are going forward to assess severe accident likelihood and consequences. These programs include plant-specific studies to determine any severe accident vulnerabilities, both from the perspective of accident frequencies and from containment performance following a core melt. Any problems will be dealt with if identified. One program is known as the Individual Plant Examination (IPE) Program and is currently under way. This program and other related programs will be conducted to provide further assessments of severe accidents on a plant-specific basis so that appropriately low risk levels can be maintained.

Evaluations of the Mark I containment with respect to severe accidents are continuing through (1) the implementation of the Commission Policy Statement on Severe Accidents, (2) the NRC staff and industry dialogue to improve containment severe accident performance for all BWRs, and (3) the containment performance improvement program. With respect to the latter program, the staff identified a number of modifications that substantially enhance the Mark I plants' capability to both prevent and mitigate the consequences of severe accidents. The improvements identified include (1) improved hardened wetwell vent capability, (2) improved reactor pressure vessel depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) updated emergency procedures and training.

After considering the staff's proposed Mark I Containment Performance Program the Commission directed the staff to pursue Mark I enhancements on a plant-specific basis in order to account for possible unique design differences that may bear on the necessity and nature of specific safety improvements. Accordingly, the Commission concluded that the recommended safety improvements, with one exception, hardened wetwell vent capability, should be evaluated by licensees as part of the Individual Plant Examination Program. With regard to the recommended plant improvement dealing with hardened vent capability, the Commission, in recognition of the circumstances and benefits associated with this modification, has directed a different approach. Specifically, the Commission has directed the staff to approve installation of a hardened vent under the provisions of 10 CFR 50.59 for licensees who, on their own initiative, elect to incorporate this plant improvement. The staff previously inspected the design of such a system that was installed by Boston Edison Company at the Pilgrim Nuclear Power Station. The staff found the installed system and the associated Boston Edison Company's analysis acceptable.

In response to the Commission's directive, the staff issued Generic Letter 89-16, "Installation of Hardened Wetwell Vent," on September 1, 1989, to all holders of operating licenses for nuclear power reactors with Mark I containments requesting licensees to submit their plans for addressing the hardened vent issue. Licensees were encouraged to install a hardened vent under the provision of 10 CFR 50.59 or to provide installation cost estimate information in order that the staff may perform plant-specific backfit analyses.

As indicated in the discussion above on the Mark I containment, the Petitioner has not presented sufficient evidence to indicate that Mark I reactors should not operate while risk-reduction improvements are being considered. That is, there is not sufficient evidence of either design flaws in Mark I reactors or high risk to warrant suspending the operating licenses for those reactors. Therefore, this portion of the Petitioner's request is denied.

B. Mark II Containment Concerns

As stated above, Petitioner alleges that Mark II reactors, supposedly an improvement over the Mark I model, still have many possible scenarios for early containment failure according to NRC safety studies. Again, Petitioner does not provide any information of which the staff was unaware. Much of what has been already stated in the discussion of the Petitioner's concerns with respect to Mark I containments as to containment design, functional purpose, and performance during severe accident scenarios applies equally to Mark II containment types.

The NRC is currently studying Mark II containment performance. The study reviews challenges to the integrity of the BWR Mark II containment that could arise from severe accidents. The challenges are organized into two broad groups: those in which containment integrity is challenged before extensive core damage, and those in which core melt occurs first, with containment integrity not threatened until the time of reactor vessel failure or later. Also reviewed are some proposed improvements that have the potential to either prevent core damage or containment failure, or to mitigate the consequences of such failure by reducing the release of fission products, and thus the offsite consequences. For each of the proposed improvements, a preliminary qualitative analysis of the impact upon core melt frequency and risk has been performed.

during severe accidents. The results of these programs will be evaluated in accordance with the Commission's regulations to determine whether any improvements should be required as a backfit.

As stated previously, Petitioner has not presented sufficient evidence to indicate that Mark II reactors should not operate while risk-reduction improvements are being considered. That is, there is not sufficient evidence of either design flaws at Mark II reactors or high risk to warrant suspending the operating licenses for those reactors. Therefore, this portion of the Petitioner's request is denied.

C. Additional Reed Report Concerns

The Petitioner also lists two concerns related to the 1975 General Electric Company "Reed Report." These are, according to the Petition, as follows:

1. In 1975, General Electric engineers wrote an internal report highly critical of their own company's nuclear reactors. This Reed Report was kept secret by both General Electric and the Nuclear Regulatory Commission until 1987, when it was released under pressure by State and local governments in cooperation with safe energy organizations. The General Electric engineers detailed dozens of safety and economics problems with all the reactors, concluding that General Electric reactors are "not a quality product." In fact, the engineers recommend that General Electric stop selling their reactors.

2. The Mark II reactors, on which the 1975 General Electric Reed Report was primarily focused, have the aforementioned "dozens of safety and economic problems," and have suffered massive cost overruns during construction as a result of design problems.

Because of the large phenomenological uncertainties and the state of flux of the ongoing research efforts, the conclusions about potential improvements are viewed as tentative. The estimated costs for selected improvements were taken from previously published information. They were not meant to be interpreted as final estimates as no cost-benefit analysis was performed.

Among the potential improvements for the first category of containment challenges are containment pressure control, such as venting from the wetwell through a hardened vent pipe, and containment pressure control and fission product scrubbing, such as the use of containment sprays with a backup water supply.

For the secondary category of containment challenges, proposed improvements include containment pressure control, for example, a hardened vent from the wetwell; improved means to depressurize the reactor, for example, enhancements to the Automatic Depressurization System (ADS) and the safety relief valves (SRVs); containment temperature control and fission product scrubbing, for example, containment sprays with a backup water supply; enhanced operability of the suppression pool cleanup systems for removal of suppression pool water and enhanced operability of the reactor water cleanup system for decay heat removal and external cooling of the drywell head; and mitigation of the fission product release, for example, use of fire protection sprays to enhance fission product retention in the reactor building. As indicated previously in the discussion on Mark I containment performance, programs are also under way to evaluate Mark II containments for performance

The Reed Report was a self-critical study performed by the staff of the General Electric Company in 1975. It was intended as a product improvement study to enhance the availability and performance of GE's boiling water reactors. The report, by its nature a candid self-analysis, was intended for GE's internal use only. It had always been held by GE to be "proprietary" and thus was not subject to public disclosure.

The principal author of the report was Dr. Charles E. Reed, a Senior Vice President of GE. Contributors included technical and professional personnel from a variety of GE departments. Their efforts resulted in the Nuclear Reactor Study, referred to today as the Reed Report, and a set of 10 subtask reports that provided the detailed technical information used to develop the Nuclear Reactor Study. The Reed Report addressed operating BWRs and the design of future GE products and services in the nuclear field. For reactors in operation at the time, the report discussed ways to improve a plant's availability and its electrical generating capacity factor through improvements in plant hardware and also in service, fuel, equipment, and operating procedures. For future reactors the report considered GE's then-new BWR design, the BWR-6, and discussed problems regarding final design details, licensing, and full-power operation of BWR-6 plants.

The NRC first learned of the existence of the Reed Report in a casual conversation between the NRC Chairman and one other Commissioner and GE officials at the San Francisco airport on August 21, 1975. There was further mention of the report in the Congressional Joint Committee on Atomic Energy hearings held in February and March 1976. At that time, Dr. Reed testified regarding the report.

On February 23-24, 1976, two NRC staff members reviewed a copy of the report in GE's Washington, D.C., offices. They determined that the report (1) did not identify any new safety concerns, and (2) did not indicate that GE had failed to report any significant safety concerns to the NRC.

On March 6, 1978, in response to a request from Congressman John D. Dingell, the NRC asked GE to provide either a copy of the Reed Report or a list of the safety issues it addressed. On March 22, 1978, GE gave the NRC a list of 25 issues identified as having "some safety significance." On May 26, 1978, GE provided to the NRC a safety evaluation of the 25 issues it had identified.

On November 9, 1978, the NRC staff gave the Commission the results of its updated review of the Reed Report and found "no substantive disagreement with the summary status provided by GE."

The NRC first received a copy of the Reed Report on January 5, 1979, under a protective agreement, when GE gave a copy to the Atomic Safety and Licensing Board in the licensing proceedings for the Black Fox nuclear plant. GE continued to categorize the report as "proprietary" and claimed that the document was exempt from mandatory public disclosure.

The NRC then received several Freedom of Information Act (FOIA) requests for the Reed Report, beginning with a request dated September 26, 1979. After reviewing arguments for and against granting an FOIA request and after consultation with the Department of Justice, the Commission voted on October 9, 1980, to release the Reed Report to the public; however, on October 17, 1980, GE sued NRC, seeking to prohibit the release. On December 21, 1984, the U.S. Court of Appeals for the Seventh Circuit ordered a remand to the

IV. CONCLUSION

The Petitioner seeks the institution of a show cause proceeding pursuant to 10 CFR 2.202 to modify or revoke the operating license of all BWR facilities. Failing that, the Petitioner seeks, without specificity, to "fix" all BWR facilities.

The institution of proceedings pursuant to 10 CFR 2.202 is appropriate only where substantial health and safety issues have been raised. See Consolidated Edison Company of New York (Indian Point, Units 1, 2, and 3), CLI-75-8, 2 NRC 173 (1975) and Washington Public Power Supply System (WPPSS Nuclear Project No. 2), DD-84-7, 19 NRC 899, 923 (1984). This is the standard that I have applied to the concerns raised by the Petitioner in this decision to determine whether enforcement action is warranted.

For the reasons discussed above, I conclude that no substantial health and safety issues have been raised by the Petitioner. Accordingly, the Petitioner's request for action pursuant to 10 CFR 2.206 is denied.

As provided in 10 CFR 2.206(c), a copy of this Decision will be filed with the Secretary of the Commission for the Commission's review. The Decision will become final action of the Commission twenty-five (25) days after issuance unless the Commission on its own motion institutes review of the Decision within that time.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 4th day of December 1989.

(2) The Reed Report does not identify any new safety issues of which the staff was unaware.

(3) Although certain issues addressed by the Reed Report are still being studied by the NRC and industry, there is no basis for suspending plant operations while those issues are being resolved.

Since knowledge of the Reed Report became public in 1987, the staff has addressed numerous Congressional and private inquiries as to the impact of the issues raised in the report on public health and safety. As stated previously, the Reed Report did not raise any new issues of which the staff was unaware. Further, corrective actions either had been implemented or were being implemented to resolve those issues. The Petitioner has not presented any evidence or any new issues identified by the Reed Report of which the staff is unaware, nor has the Petitioner presented any evidence calling into question the adequacy of the corrective actions implemented since the Reed Report was issued. On this basis, therefore, the Petitioner's request is denied.

D. Economic Issues

Insofar as Petitioner asks for relief because of "economic problems" or "massive cost overruns during construction as a result of design problems," the NRC is without jurisdiction to grant relief. The NRC has authority to govern any activity authorized pursuant to the Atomic Energy Act of 1954, as amended, in order to protect health and to minimize danger to life or property. Because economic problems and cost overruns raise no threat to public health and safety, they do not provide the NRC with a basis on which to act. Accordingly, insofar as Petitioner bases her request on economic or cost considerations, the Petition is denied.

failures; and (6) Mark II designs, on which the Reed Report focused, have dozens of safety and economic problems and have suffered massive cost overruns during construction as a result of design problems.

On June 5, 1989, the Director, NRR, acknowledged receipt of the Petition. He informed Ms. Harlowe that (1) the Petition would be treated under 10 CFR 2.206 of the Commission's regulations, and (2) appropriate action would be taken within a reasonable time.

The Director has now determined that Ms. Harlowe's requests should be denied for the reasons set forth in the "Director's Decision Pursuant to 10 CFR 2.206" (DD-89-9). The Decision is available for inspection and copying in the Commission's Public Document Room, Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555, and at the Local Public Document Rooms near the facilities listed below. The addresses and hours of operations for the local public document rooms may be obtained by calling the following toll-free number: 1-800-638-8081.

A copy of the Decision has been filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c). As provided in 10 CFR 2.206(c), the Decision will become the final action of the Commission twenty-five (25) days after issuance unless the Commission on its own motion institutes review of the Decision within that time.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
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Dated at Rockville, Maryland,
this 4th of December 1989.

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.; Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Information Support Services, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

DD-89-9

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
Thomas E. Murley, Director

In the Matter of

BOSTON EDISON CO. (Pilgrim Nuclear Power Station, Docket No. 50-293)

CAROLINA POWER & LIGHT CO. (Brunswick Steam Electric Plant, Units 1 and 2, Docket Nos. 50-324 and 50-325)

CLEVELAND ELECTRIC ILLUMINATING CO., ET AL. (Perry Nuclear Power Plant, Unit 1, Docket No. 50-440)

COMMONWEALTH EDISON CO. (Dresden Nuclear Power Station, Units 2 and 3, Docket Nos. 50-237 and 50-249), (Quad Cities Station, Units 1 and 2, Docket Nos. 50-254 and 50-265), LaSalle County Station, Units 1 and 2, Docket Nos. 50-373 and 50-374)

CONSUMERS POWER CO. (Big Rock Point Nuclear Plant, Docket No. 50-155)

DETROIT EDISON CO. (Enrico Fermi Atomic Power Plant, Unit 2, Docket No. 50-341)

GENERAL PUBLIC UTILITIES (Oyster Creek Nuclear Power Plant, Docket No. 50-219)

GEORGIA POWER CO. (Edwin I. Hatch Nuclear Plant, Units 1 and 2, Docket Nos. 50-321 and 50-366)

GULF STATES UTILITIES CO. (River Bend Station, Docket No. 50-458)

ILLINOIS POWER CO. (Clinton Power Station, Docket No. 50-461)

IOWA ELECTRIC LIGHT & POWER CO. (Duane Arnold Energy Center, Docket No. 50-331)

LONG ISLAND LIGHTING CO. (Shoreham Nuclear Power Station, Docket No. 50-322)

MISSISSIPPI POWER & LIGHT CO. (Grand Gulf Nuclear Station, Docket No. 50-416)

NEBRASKA PUBLIC POWER DISTRICT (Cooper Nuclear Station, Docket No. 50-298)

NIAGARA MOHAWK POWER CORP. (Nine Mile Point Nuclear Station, Units 1 and 2, Docket Nos. 50-220 and 50-410)

NORTHEAST UTILITIES (Millstone Nuclear Power Station, Docket No. 50-245)

NORTHERN STATES POWER CO. (Monticello Nuclear Generating Plant, Docket No. 50-263)

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NORTHEAST UTILITIES (Millstone Nuclear Power Station, Docket No. 50-245)
NORTHERN STATES POWER CO. (Monticello Nuclear Generating Plant, Docket No. 50-263
PENNSYLVANIA POWER & LIGHT CO. (Susquehanna Steam Electric Station, Units 1 and 2, Docket Nos. 50-387 and 50-388)
PHILADELPHIA ELECTRIC CO. (Peach Bottom Atomic Power Station, Units 2 and 3, Docket Nos. 50-277 and 50-278), (Limerick Generating Station, Unit 1, Docket No. 50-352)
POWER AUTHORITY OF THE STATE OF NEW YORK (James A. Fitzpatrick Nuclear Power Plant, Docket No. 50-333)
PUBLIC SERVICE ELECTRIC & GAS CO. (Hope Creek Nuclear Station, Docket No. 50-354)
TENNESSEE VALLEY AUTHORITY (Browns Ferry Nuclear Power Station, Units 1, 2, and 3, Docket Nos. 50-259, 50-260, and 50-296)
VERMONT YANKEE NUCLEAR POWER CORP. (Vermont Yankee Nuclear Power Station, Docket No. 50-271)
WASHINGTON PUBLIC POWER SUPPLY SYSTEM (WNP Unit 2, Docket No. 50-397)

ABSTRACT

In 1975, the General Electric Company (GE) published a Nuclear Reactor Study, also referred to as "the Reed Report," an internal product-improvement study. GE considered the document "proprietary" and thus, under the regulations of the Nuclear Regulatory Commission (NRC), exempt from mandatory public disclosure. Nonetheless, members of the NRC staff reviewed the document in 1976 and determined that it did not raise any significant new safety issues. The staff also reached the same conclusion in subsequent reviews.

However, in response to recent inquiries about the report, the staff re-evaluated the Reed Report from a 1987 perspective. This re-evaluation, documented in this staff report, concluded that (1) there are no issues raised in the Reed Report that support a need to curtail the operation of any GE boiling water reactor (BWR); (2) there are no new safety issues raised in the Reed Report of which the staff was unaware; and (3) although certain issues addressed by the Reed Report are still being studied by the NRC and the industry, there is no basis for suspending licensing and operation of GE BWR plants while these issues are being resolved.

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EXECUTIVE SUMMARY

The purpose of this NRC staff evaluation of the General Electric Nuclear Reactor Study (the Reed Report) and its 10 subtask reports is to reconsider the issues and concerns identified in the report in the light of current knowledge, recent operating experience, and regulatory issues as they have developed since the report was issued in 1975.

A History of the Reed Report

The Reed Report was a self-critical study performed by the staff of the General Electric Company (GE) in 1975. It was intended as a product-improvement study to enhance the availability and performance of GE's boiling water reactors (BWRs). The report, by its nature a candid self-analysis, was intended for GE's internal use only. It has always been held by GE to be "proprietary," and thus not subject to public disclosure.

The principal author of the report was Dr. Charles E. Reed, a Senior Vice President of GE. Contributors included technical and professional personnel from a variety of GE departments. Two products resulted from their efforts. One was the Nuclear Reactor Study, referred to today as the Reed Report; the second was a set of 10 subtask reports that provided the detailed technical information used to develop the Nuclear Reactor Study.

The Structure of the Reed Report

The Reed Report addressed operating BWRs and the design of future GE products and services in the nuclear field.

For reactors in operation at the time, the report discussed ways to improve plant availability and its electrical generating capacity factor through improvements in plant hardware and through improvements in service, fuel, equipment, and operating procedures. For future reactors, the report considered GE's then-new BWR design, the BWR-6, and discussed problems regarding final design details, licensing, and full-power operation of BWR-6 plants.

The report addressed 10 general topics, as follows:

(1) nuclear systems
(2) fuel
(3) electrical, control, and instrumentation
(4) mechanical systems and equipment
(5) materials, processes, and chemistry
(6) production, procurement, and construction
(7) quality control systems overview
(8) management/information systems
(9) regulatory considerations
(10) scope and standardization

Each of these general topics was addressed in a separate subtask report, and the 10 subtask reports were used to generate the Reed Report.

History of NRC Actions Regarding the Reed Report

The Nuclear Regulatory Commission (NRC) first learned of the existence of the Reed Report in a casual conversation between the NRC Chairman and one other Commissioner and GE officials at the San Francisco airport on August 21, 1975. There was further mention of the report in the Congressional Joint Committee on Atomic Energy hearings held in February and March 1976. At that time, Dr. Reed testified regarding the report.

On February 23-24, 1976, two NRC staff members reviewed a copy of the report in GE's Washington, DC offices. They determined that the report (1) did not identify any new safety concerns and (2) did not indicate that GE had failed to report any significant safety concerns to the NRC.

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NRC Categorization of Reed Report Issues

On the basis of its reviews of the Reed Report and on information on the report supplied by GE, in November 1978 the staff grouped the 25 issues addressed in the report into six categories as follows:

- constraints on operation resulting from regulatory requirements (7 items)
- plant-specific matters to be resolved in plant-specific license reviews (4 items)

* features already deleted from GE design (1 item)
* quality assurance issues (2 items)
* issues for which final resolution was pending, but for which interim positions provided an adequate basis for allowing continued licensing of plants (8 items)
* issues already resolved by staff review (3 items)

Recent NRC Actions Regarding the Reed Report

On June 2, 1987, NRC established a special task group to evaluate again the issues raised in the Reed Report, taking into account the increased knowledge about nuclear power based on engineering studies and operational experience in the 12 years since the Reed Report was written.

This review produced three separate conclusions:

(1) The Reed Report does not identify any matters that would support a need to curtail the operation of any GE boiling water reactor plants now licensed.

(2) The Reed Report does not identify any new safety issues of which the staff was unaware.

(3) While certain issues addressed by the Reed Report are still being studied by the NRC and industry, there is a basis for permitting continued plant operations while those issues are being resolved.

1 INTRODUCTION

The purpose of this NRC staff evaluation of the General Electric Company's Nuclear Reactor Study (the Reed Report) is to reconsider the issues and concerns identified in the report in the light of current knowledge, more recent plant operating experience, and regulatory issues as they have developed since the report was issued in 1975.

This re-evaluation was prompted by concerns expressed by public officials and others regarding alleged serious weaknesses in the safety of General Electric (GE) boiling water reactors (BWRs). These statements of concern were reactions to recent accounts in the news media, particularly newspaper accounts, of a "secret" GE report written in 1975. The report referred to in news accounts is the GE Nuclear Reactor Study, which is more commonly called the Reed Report because Dr. Charles E. Reed, a Senior Vice President of GE, headed the task group whose studies culminated in the issuance of the Nuclear Reactor Study.

Because of the nature of the study, GE has always held the Reed Report to be proprietary, not to be disclosed to the public or to GE's competitors. The NRC has a copy of this GE proprietary report, along with the proprietary subtask reports and related material. In the course of performing its regulatory functions, the NRC receives and holds for review and for reference many proprietary documents from GE and from other vendors of nuclear-related products. The NRC staff had long been aware of the Reed Report and its contents.

Recently, however, in the discovery process of a lawsuit involving GE and the owners of the Zimmer facility, excerpts from the Reed Report, and other internal GE documents, apparently were included in documents being exchanged between the parties in the lawsuit. This material came into the possession of a newspaper, which purportedly disclosed some of the contents in a news article. Some newspaper articles contained accounts that stated or implied that the NRC had conspired with GE to keep this "secret" report from the public because of information that would be damaging to GE if it were disclosed. These articles, together with interest from Congress, officials from the State of Ohio, and concerned citizens, prompted the NRC staff to initiate a thorough current review re-evaluation of the Reed Report and the 10 subtask reports. The results of this current NRC staff evaluation are the subject of this report.

2 BACKGROUND

2.1 History of the Reed Report

The Reed Report was a self-critical study performed by the staff of GE in 1975, with the stated objectives 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 (BWRs)."

The principal author of the report was Dr. Reed. Contributors included technical and professional personnel from a variety of GE departments. Two products resulted from their efforts. One was the Nuclear Reactor Study, referred to as the "Reed Report"; the second was a set of 10 subtask reports that provided the detailed technical information used to develop the Nuclear Reactor Study.

The Reed Report was intended to be an internal document, not one for public disclosure because, as claimed by GE, it contained information and comments that could have an adverse effect on GE's market position with respect to its competitors.

Although GE allowed NRC to review the document on several occasions and eventually provided NRC with a copy, GE also sued NRC to prevent the agency from releasing the document to the public.

2.2 Structure and Contents of the Reed Report

The report addressed 10 general topics related to the GE nuclear power product line; these topics were:

(1) nuclear systems
(2) fuel
(3) electrical, control, and instrumentation
(4) mechanical systems and equipment
(5) materials, processes, and chemistry
(6) production, procurement, and construction
(7) quality control systems overview
(8) management/information systems
(9) regulatory considerations
(10) scope and standardization

Each of these general topics was addressed in a separate subtask report, and the 10 subtask reports were used to generate the Reed Report. The subtask reports are discussed in detail in Section 5 of this report.

For reactors in operation at the time, the report discussed ways to improve plant availability and its electrical generating capacity factor through improvements in plant hardware and through improvements in service, fuel, equipment, and operating procedures. For future reactors, the report considered GE's then-new BWR-6, and discussed problems regarding final design details, licensing, and unrestricted full-power operation.

2.3 History of NRC Actions Regarding the Reed Report

- 1975-1976

The NRC first learned of the existence of the Reed Report in a casual conversation between the NRC Chairman and one other Commissioner and GE officials at the San Francisco airport on August 21, 1975. According to testimony given at the Hearing of the Joint Committee on Atomic Energy in its Investigation of Charges Relating to Nuclear Reactor Safety, February 18, 23, and 24 and March 2 and 4, 1976, the mention of the report was oral and very general in nature.

However, because concerns were raised about the contents of the Reed Report, on February 23-24, 1976, two NRC staff members reviewed a copy of the report in GE's Washington, DC offices. They wanted to determine if the report (1) identified any new safety concerns of which the NRC was not aware, and (2) if GE had met the requirements of Section 206 of the Energy Reorganization Act of 1974 in regard to the reporting of significant safety items.

On the basis of their review, these staff members did not identify any new safety concerns or any evidence that significant safety concerns had not been reported to the NRC. A copy of their memorandum to the Director of the NRC Office of Nuclear Reactor Regulation (NRR) that documented their conclusions was incorporated into the record of the hearing of the Joint Committee on Atomic Energy's Investigation of Charges Relating to Nuclear Reactor Safety.

- 1977-1978

In December 1977, Congressman John D. Dingell asked the Commission to provide information on the Reed Report. The Chairman responded in a letter dated February 9, 1978, which described the staff's earlier review and its conclusions.

To provide further information to the Congressman, on March 6, 1978, the NRC asked GE to provide either a copy of the Reed Report or a list of the safety issues it addressed. GE responded by a letter dated March 22, 1978, which contained a list of 25 issues identified as having "some safety significance."

On April 11, 1978, two members of the NRC staff and one member of Congressman Dingell's staff reviewed the report itself at the GE offices in Washington, DC. And, on May 26, 1978, GE sent a letter to NRC that gave a status report on each of the 25 items.

On November 9, 1978, the NRC staff gave the Commission the results of its updated review of the Reed Report (SECY-78-462A). The staff review concluded: "no substantive disagreement with the summary status provided by GE." The staff also grouped the 25 issues in the report into six categories.

In a letter dated December 27, 1978, the Chairman forwarded the staff's findings and conclusions to Congressman Dingell.

- 1978-1979

On October 18, 1978, the Atomic Safety and Licensing Board (ASLB) in the Black Fox proceedings issued a subpoena to GE calling for GE to provide a copy of the

Reed Report for the proceedings. GE refused, claiming the report was "proprietary," and thus protected from mandatory public disclosure under the Commission's regulations.

GE and the ASLB were able to settle on the terms of a protective agreement, under which GE provided a copy of the report on January 5, 1979. This was the first time NRC had a copy of the report. However, under the terms of the protective agreement, the report itself was never introduced into the Black Fox proceedings.

The protective agreement did allow the following:

(1) The Reed Report was made available to the ASLB in confidence.

(2) Verbatim extractions from the report were available to counsel insofar as they related to the Intervenor's contentions and the ASLB's questions.

(3) The report was available to the Intervenor's counsel to evaluate the faithfulness of the extractions.

The parties also signed protective agreements that limited access to and use of the report.

In September 1979, the NRC received the first of several FOIA requests for the Reed Report.

- <u>1980-1984</u>

Several FOIA requests for the Reed Report were received in this period, the first actually having been made in September 1979. On October 9, 1980, after hearing arguments on a request made under the FOIA, the Commission voted to release the Reed Report to the public. However, on October 17, 1980, GE sued NRC, seeking to prohibit the release. Subsequently, on December 21, 1984, the U.S. Court of Appeals for the Seventh Circuit ordered a remand of the Commission's decision.

- <u>1986-1987</u>

In July 1986, the Commission voted to continue to withhold the Reed Report from public disclosure. This decision was based on the Commission's desire to encourage similar studies and ensure NRC access to their results. On June 3, 1987, Ohio Citizens for Responsible Energy (OCRE) filed suit in Ohio Federal District Court seeking public release of the report under the Freedom of Information Act.

To date the Commission has not released the Reed Report to the public.

2.4 <u>NRC Categorization of Reed Report Issues</u>

In its November 1978 report to the Commission (see above), the staff grouped the 25 issues addressed in the Reed Report into six categories as follows:

- constraints on operation resulting from regulatory requirements (7 items)

- plant-specific matters to be resolved in plant-specific license reviews (4 items)

- features already deleted from GE design (1 item)

- quality assurance issues (2 items)

- issues for which final resolution was pending, but for which interim positions provided an adequate basis for allowing continued licensing of plants (8 items)

- issues already resolved by staff review (3 items)

2.5 Recent NRC Actions Regarding the Reed Report

On June 2, 1987, following the appearance of newspaper stories with controversial accounts of the contents and safety implications of the report, and statements attributed to some public officials and others in these newspaper accounts and the receipt of inquiries from Congress, NRC established a special task group to re-evaluate the issues raised in the Reed Report, taking into account the increased knowledge and understanding of nuclear power issues gained in the 12 years since the Reed Report was written. Martin Virgilio was appointed task group leader. Other people were named as needs were identified for specific expertise. The people who contributed significantly to this effort are listed below.

Martin Virgilio - task group leader
Roby Bevan - technical coordinator
Ed Shomaker - legal counsel
C. Y. Cheng - technical expert
Tim Colburn - project manager, Perry Nuclear Power Plant
John Craig - technical manager
Walt Haass - technical expert
Warren Hazelton - technical expert
Wayne Hodges - technical manager
Jack Kudrick - technical expert
Oliver Lynch - technical expert
Jerry Mauck - technical expert
Robert Pettis - technical expert
Laurence Phillips - technical expert
John Ridgely - technical expert
Chen Tan - technical expert
John Thoma - technical expert
Charles Tinkler - technical expert
Robert Wright - technical expert

3 THE "TWENTY-FIVE LICENSING ISSUES" IDENTIFIED BY GENERAL ELECTRIC

As discussed above, the Reed Report was not concerned primarily with safety issues associated with GE BWRs, but with plant availability and electric generating capability and, hence, the marketability of the GE nuclear reactors. However, in response to requests by NRC, in 1978, GE's Nuclear Safety and Licensing organization reviewed the report and identified 27 safety-related items. The 27 issues were subsequently consolidated into 25 when 2 of the items identified earlier were included under other issues.

The NRC staff has again reviewed these 25 issues in the light of current knowledge of nuclear safety, and the results of that review are given below. For each of these issues, there is a statement of the issue, a statement of its safety significance, and a statement of the current status of the issue.

The staff finds that none of the 25 issues identified by GE as having some safety significance involve any safety considerations not already identified and appropriately addressed by the staff.

3.1 Degree of Completion of BWR-6 Design

- Issues

The Reed Report noted the following with regard to the BWR-6 Mark III design:

(1) The BWR-6 Mark III design was incomplete (in 1975), and several important technical problems were unresolved.

(2) The overall design of the BWR-6 Mark III is not well integrated. The design was a result of a process of evolution and reaction to competitive offerings and regulatory requirements.

(3) Future potential problems in the areas of fuels management, operational limitations, licensing, and component replacement had to be anticipated.

- Safety Significance

None. In 1975, the NRC was reviewing applications for construction permits based on preliminary BWR-6 design details, and completion of the final design details lagged significantly behind the start of construction. Accordingly, as permitted by its regulations, the NRC issued construction permits without complete or final detailed design information. As that information was later submitted during the operating license review, licensing problems sometimes resulted because some information was unsubstantiated. The end result was increased NRC review effort in some areas. This delay in the review process may have had an economic impact on the licensee, but there was no safety significance because the licensing review was simply delayed.

- Status

Before the first BWR-6 operating license was granted, the NRC reviewed and approved detailed plant design information. The following BWR-6 Mark III designs have been approved by the NRC: Clinton 1, Grand Gulf 1 and 2, Perry 1, and River Bend.

3.2 Amount of Margin Between Design Calculations for Core and Operating Limits

• **Issues**

The Reed Report noted the following with regard to the BWR-6 core at the preliminary design stage:

(1) Design thermal margin was not sufficient to avoid power derating (a reduction in allowed power level) to as low as 80% of the intended rated power to meet operating limits during portions of the core operating cycle. Such a power derating would limit the reactor to operate at only some fraction of its rated power, a substantial economic consideration.

(2) Calculational models with inadequate experimental verification could have proven to be nonconservative and might require a power derating of 5% to 10%.

• Safety Significance

None. Derating a plant to maintain adequate margin in operating limits is an economic issue, not a safety issue.

• Status

Today, cores are operating at or near the operating limits (not safety limits), as design thermal margin is maintained, while using new fuel designs, less conservative calculational models, and revised operating conditions. This generally requires revised technical specification operating limits for each operating cycle.

Nuclear power plant licensees are maintaining adequate safety margins in their operating plants by adhering to technical specification operating limits. The need for power derating is marginal, and it is generally avoided by operating plants according to cycle-dependent technical specifications that define the operating limit minimum critical power ratio for that operating cycle, using NRC staff-approved models and calculational methods.

3.3 Impact of Cold Shutdown Reactivity Margin on BWR-6 Core Design

• Issue

The Reed Report noted that the design calculation models were inadequate to ensure that the cold shutdown reactivity margin for the BWR-6 equilibrium core could meet the stuck-rod margin requirements in a plant's operating license.

• Safety Significance

None. The concern was and is economic because plant shutdown and/or limited plant availability can result when a licensee cannot demonstrate adequate shutdown margin.

• Status

All operating BWRs have technical specifications that require shutdown margin be maintained and that the plant be shut down if measured shutdown margin is inadequate.

Calculational models for the final design equilibrium core will better reflect the burnup experience with cores that contain gadolinia in order to maintain a flatter reactivity response at core equilibrium.

3.4 Impact of End of Cycle (EOC) Scram Reactivity Insertion Rate on Core Full Power Life

• Issue

The Reed Report noted that reduced scram response because of unfavorable void coefficients and the design scram reactivity curve at EOC could require derating up to 20% to meet operating limits.

• Safety Significance

None. The concern is the economic cost of derating (reduction in allowed power level) to meet regulatory limits.

• Status

GE has addressed the economic consideration of plant derating to meet operating limits at EOC operation through the following improvements:

(1) improved fuel design (fewer negative coefficients)

(2) improved calculation models

(3) design modifications to the BWR-6 scram system for more rapid insertion of rods

(4) highly cycle-dependent (and core-exposure-dependent) technical specification operating limits

(5) recirculation pump trip provisions added to all BWR product lines

3.5 Long-Term Effect of Radiation on Core Internals

• Issue

The Reed Report noted that uncertainties in estimates of radiation and corrosion damage to BWR-6 core internals did not provide assurance of a 40-year lifetime of service. Core internals might have to be replaced earlier to provide assured structural integrity for continued operation. Replacement of permanently installed core internals would result in substantial reactor downtime. Also, replacing these core internals would be difficult because access to them is difficult and workers would be exposed to high levels of radiation.

- <u>Safety Significance</u>

Two areas in the reactor internals were identified that could receive enough radiation fluence to significantly affect the material properties. These were the top guide and the mid-plane of the shroud. Although not stated directly in the discussion in the Reed Report, the apparent concern was that the material properties could be degraded to the point where the components could fail. Failure of some core internals could hinder (but not prevent) shutdown of the reactor. The GE analysis indicated that there would be sufficient margin to insert rods to achieve shutdown, even with channel interference or loss of spacing.

- <u>Status</u>

One effect of radiation on core internals and support structures that was recognized in the early 1980s is that austenitic stainless steel becomes susceptible to stress corrosion cracking. Cracks have been found in neutron monitor guide tubes in at least six BWRs. Cracks have also been found in control blade handles and sheaths.

GE has evaluated the possibility of irradiation-assisted stress corrosion cracking (IASCC) in other components, some of which were mentioned in the 1975 study. The top guide and shroud are still unlikely to last the 40-year life for which the reactor is licensed, but it is believed that the core plate will not experience enough neutron fluence to be affected. GE has been actively involved in developing non-destructive evaluation (NDE) equipment and procedures to detect IASCC, and in developing a methodology to justify continued operation with cracked components where such operation would not compromise safety.

Should the assembly become so degraded by cracking and loss of toughness that the assembly failed during a seismic event, failure could occur at several locations, and rod blockage or loss of the guide function might occur.

GE believes that the core plate is not likely to receive enough neutron fluence to become susceptible to cracking. Nevertheless, the threshold value of fluence is not yet known with certainty, and further study of this subject is being pursued.

Although Section XI of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME Code) requires visual inspection of core support structures every 10 years, the postulated crack locations may not be accessible for TV viewing. GE has been actively working on a methodology to perform remote ultrasonic inspection of the suspect locations. If this proves feasible, the top guide assembly could be inspected at selected plants with long service to determine whether a generic problem exists.

The staff believes that current monitoring, surveillance, and inspection programs will identify any incipient failure of core internals before failure, and that the radiation levels associated with plant operation are not likely to result in reactor safety problems from materials failure in BWR core internals.

3.6 Degree of Proof of Accuracy of Transient Design Methods

• Issue

The Reed Report notes that there were large calculational uncertainties because of inadequate verification of transient design methods. This inadequate verification could lead to reduction in allowed level of power operation. Application of more accurate methods, or reduction of these uncertainties by better verification programs, could result in smaller margins being permitted in thermal hydraulic transient analyses that are performed to ensure that the plant does exceed its thermal operating limit.

• Safety Significance

The concern was primarily economic, with potential power derating being required to meet regulatory operating limits.

• Status

Better calculational methods have been developed and verified against plant transient tests. In parallel with these tests, more sophisticated computer codes modeling the reactor core behavior have been developed. The problem has been resolved (the resolution of Generic Issue B-19) with the staff approval and licensee implementation of more sophisticated core modeling codes.

3.7 Impact on Fuel Integrity of Reduced Moderator Temperature due to Equipment Failure

• Issue

The Reed Report noted that excessive fuel failures due to pellet-cladding interaction (PCI) were causing power derating to reduce the leakage and dispersal of radioactivity into the reactor cooling water. Prolonged overpower transients due to loss of feedwater heating, or other coolant temperature reduction transients, could lead to PCI failures and challenge thermal hydraulic design limits.

• Safety Significance

The rapid subcooling and reactivity spike resulting from loss of feedwater heaters is reflected in fuel failures induced by PCI and leads to some increase in personnel radiation exposures. Such equipment failure and resulting fuel failure is to be avoided, and the increased exposure to plant personnel is contrary to the ALARA (as low as is reasonably achievable) exposure reduction objectives, but the reactor safety implications are minimal beyond that.

• Status

The issue of fuel integrity is not a problem because it is addressed by the following measures:

(1) preconditioning of fuel during the early phases of a new operating cycle
(2) use of new fuel design (barrier fuel)

(3) provisions for thermal power monitor for delayed overpower trip
(4) continued compliance with technical specification core operating thermal limits

3.8 Performance of Relief Valve Augmented Bypass (REVAB) System

- Issue

The Reed Report notes that the scram insertion requirèment for plants designed with the REVAB have not yet been achieved.

- Safety Significance/Status

None. This issue has no safety significance and is not relevant because the REVAB system has been deleted from the design of GE BWRs and is not used in any operating BWR.

3.9 Impact of Hydrodynamic Phenomena on Containment Designs

- Issue

A general concern over the (then) current state of containment-related issues is reflected throughout the Reed Report, with reference made to the containment issues in the Executive Summary, in the section entitled Nuclear Systems, and in the section entitled Mechanical Systems and Equipment. In several cases the same issue is discussed in different sections but with a different perspective or with emphasis on particular elements of the technical issue.

The issue of hydrodynamic phenomena and their impact on containment designs, discussed throughout the report, is identified as "Impact of Recently Discovered Phenomena on Containment Designs" in the 25 issues identified by GE. The report says: "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."

- Safety Significance

The Reed Report reflects the uncertainty present in 1975 surrounding the discovery of additional containment loads created by suppression pool phenomena related to safety relief valve (SRV) air clearing, pool swell, and high temperature steam condensation. These phenomena were identified during early testing of the Mark III design, which was initiated in 1973, and by the experience at two German BWR Mark I containments in 1972. At the German plants, severe vibratory loads on the containment structure were experienced during extended SRV operation. In 1975, concerns also were being raised by former employees of GE, and hearings were held before the Congressional Joint Committee on Atomic Energy regarding the impact of hydrodynamic loads on BWR containment designs.

The safety significance of this issue was that the additional loading created by these phenomena during an accident or transient could jeopardize the integrity of the containment structure, drywell, and/or equipment and structures near the suppression pool. Failure of the containment or drywell structures could have serious consequences during certain reactor accidents.

- Status

After the Reed Report was issued, BWR owners, working with GE, completed extensive testing and analyses, resolving all technical issues related to suppression pool hydrodynamic loads. Both generic and in-plant testing were performed to provide an expanded data base on which conservative loading definitions could be developed. To reduce loads created by SRV operation, new SRV discharge quencher designs were approved and installed. Additionally, various plant-specific modifications were made to strengthen the containment structure as needed to restore design safety margins. The NRC initiated several generic issues to guide, track and document resolution of these technical concerns, as follows:

(1) Generic Issue A-6, Mark I Short Term Program. The resolution was documented in NUREG-0408.

(2) Generic Issue A-7, Mark I Long Term Program. The resolution was documented in NUREG-0661.

(3) Generic Issue A-8, Mark II Program. The resolution was documented in NUREG-0487 and NUREG-0808.

(4) Generic Issue A-39, Determination of SRV Pool Dynamic Loads and Temperature Limits for BWR Containments. The resolution was documented in NUREG-0802.

(5) Generic Issue B-10, Behavior of BWR Mark III Containment. The resolution was documented in NUREG-0978.

3.10 Radiation Exposure from Removal of Steam Dryer/Separator Assembly

- Issue

The Reed Report noted that there was a potential for significant plant personnel radiation exposure from dryer/separator assembly handling for the BWR-6 Mark III design.

- Safety Significance

Concerns were limited to those of occupational radiation exposure. There were no reactor plant safety concerns beyond the ALARA issue. The issues involved were primarily economic considerations associated with decreased availability due to a lack of maintainability, and the ALARA issue of maintaining occupational exposure to low levels.

- Status

After the Reed Report was issued, the BWR-6 Mark III design was modified to allow underwater transfer of the dryer/separator assembly, thereby reducing occupational exposure rates, particularly during refueling. The NRC staff considers this modification an excellent example of field feedback, self-analysis, and implementation of ALARA guidelines.

3.11 Level of Testing of Mark III Containment

- Issue

The Reed Report in several sections reflects concerns over the adequacy of testing planned to investigate suppression pool phenomena for the Mark III containment. Although this concern is related to the general issue of suppression pool hydrodynamic loads, it is specifically related to questions over scaling of Mark III tests to determine pool swell loads resulting from a loss-of-coolant accident (LOCA). The concern stems from initial Mark III tests that were conducted with non-uniform scaling; a full-scale sector of the suppression pool was simulated while the drywell and boiler simulation was 1/3 scale. The Nuclear Systems section of the Reed Report recommended that "full-scale" boiler and drywell tests be performed along with consistent 1/3-scale tests. In the Mechanical Systems and Equipment section of the Reed Report, the recommendation is conditional; it recommends that 1/3-scale testing be completed as rapidly as possible, and expanded, if necessary, to resolve uncertainties.

- Safety Significance

The safety significance of this issue deals with the uncertainty over load definition for suppression pool phenomena. If the test data used to define loads were based on improperly scaled test models, then by extension the load definition used to evaluate containment structural response would be inadequate.

- Status

Pool swell tests were continued for approximately 4 years after the issuance of the Reed Report. Testing was conducted on a variety of scales and configurations in order to confirm the use of conservative scaling factors in load definition. A full-scale model of the drywell, boiler, and suppression pool was not needed. The GE technical resolution was documented in a series of reports, NEPT-13377, 20550, 21853, 13407, 13426, 13435, 21596, 24648, and 24720. The load definition report for the Mark III containments (GE document 22A 7007, February 25, 1982) was reviewed and approved by the NRC. The NRC also initiated Generic Issue B-10, "Behavior of BWR Mark III Containment," to address this issue; NRC evaluation and resolution of this generic issue was addressed in NUREG-0978 (August 1984), which documented the NRC staff acceptance of modifications and results of the load definition report on the Mark III containment.

3.12 Presence of Detectable Plutonium Inside the BWR Turbine

- Issue

The Reed Report noted that detectable amounts of plutonium produced by transmutation of uranium had migrated beyond the fuel pin boundaries and deposited inside the turbine of BWR reactors.

- Safety Significance

Plutonium is a source of long-lived alpha radiation, chemically related to calcium. When it is ingested, it tends to deposit in the bone. This subjects the tissue to long-term ionizing radiation, which can produce cancer.

• Status

Trace amounts of plutonium deposited inside the turbine are carried by steam from the reactor core to the turbine. The plutonium can be produced from tramp uranium, which are trace amounts deposited outside the fuel pins, or from leaking fuel pins. Experience has shown that essentially all of the plutonium formed in the fuel stays there. Further, analyses of reactor water show that the plutonium content is typically less than 1% of the permissible drinking water level. These trace quantities are removed by the reactor water purification system. Plutonium contamination in BWR turbines is not a significant problem.

3.13 The Effect of Sloshing of the Suppression Pool on Mark III Steel Containment Structure Design

• Issues

The Reed Report noted that testing asociated with Mark III containment was incomplete and the potential for dynamic buckling resulting from seismic sloshing of the suppression pool had to be considered in the design of the steel containment.

• Safety Significance

Buckling of the steel containment shell from sloshing of the suppression pool in a seismic event may result in failure of the containment functional capability.

• Status

The potential for buckling of the steel containment shell as a result of sloshing of the suppression pool is being handled in several different ways.

At Perry and River Bend, the annulus between the steel shell and shield building is filled with concrete up to a level above the suppression pool. Through analysis, it has been demonstrated that seismic sloshing of the pool then cannot result in buckling of the steel shell. Thus, the containment functional capability cannot then be compromised, and there is no safety significance. At the design stage, buckling of the steel shell without concrete backing was considered in the Perry and River Bend plants, and was reviewed by the NRC staff. The design was found to have met the staff's buckling criteria. In the case of Grand Gulf and Clinton, the containment structure is not a steel shell, but is concrete, not subject to potential buckling from seismic sloshing.

Buckling of steel containment shells, including consideration of dynamic responses of the shell, was studied at Lockheed Palo Alto Research Laboratory under contract with NRC. The staff's buckling criteria are based mainly on the results of this study (NUREG/CR-2836).

3.14 Evaluation of Fuel Transfer Accident in Mark III Containment

• Issue

The Reed Report noted that the potential for a fuel transfer accident in the Mark III containment had not been evaluated.

In 1975, GE had not completed the design of the Mark III containment. This containment was similar to the pressurized water reactor (PWR)-style containment where the spent fuel storage facility is located outside of the reactor building and away from the refueling floor. In the Mark III design, the spent fuel pool is located at a lower elevation than the refueling floor, whereas in the PWR designs the refueling floor and the fuel handling floor in the fuel building are at the same elevation. A concern was raised that spent fuel would have to be transported in the Mark III containment from the refueling floor elevation to the lower fuel building elevation. Since this spent fuel had recently been in the core, it would have a high rate of decay heat generation. If the fuel were to become immobile during the fuel transfer process, there might not be adequate cooling for the fuel bundle, and the radiation shine through the surrounding walls might create a new and different type hazard to plant personnel. In addition, an elaborate valving arrangement was needed to prevent the water in the upper pool (inside containment) from draining down into the spent fuel pool.

- <u>Safety Significance</u>

The potential safety significance of these postulated accidents is centered around two areas: radiation exposure considerations and the potential breaching of containment. The stuck fuel bundle in the transfer mechanism could represent a radiation exposure concern for workers in areas adjacent to the fuel transfer tube and for those on the refueling floor from gas being released from fuel bundles as they heat up because available cooling is not adequate.

The simultaneous opening of both transfer isolation valves (one at the refueling floor in the reactor building and the second in the fuel building in the spent fuel pool) could breach containment and drain the upper containment pool, flooding the spent fuel pool and the fuel handling floor. If a spent fuel bundle were to be stuck in the transfer tube at the time of the valve failures, the bundle would overheat once the upper pool was drained; this would result in a release of radioactivity to the containment atmosphere, resulting in increased exposure to the fuel handling personnel in the vicinity.

- <u>Status</u>

Since the Reed Report was issued in 1975, GE has completed an evaluation of these potential accidents. In addition, the NRC staff reviews the potential fuel handling accident as part of the licensing process. In the GE design, adequate protective measures are taken to prevent personnel from having access to areas near the transfer tube, especially during fuel transfer operations. The NRC staff has reviewed the fuel transfer system to verify that no single failure could result in a fuel handling accident, and that all aspects of the system have the appropriate alarms and interlocks. As part of this failure modes and effects analysis, the potential for inadvertent opening of both transfer tube isolation valves simultaneously was given special attention to ensure that containment will not be breached and that the upper containment pool will not be drained. Thus, the concerns raised in the Reed Report have been satisfactorily addressed to ensure that the use of the inclined fuel transfer system will not result in any significant increase in the risk to the health and safety of the public or to plant personnel.

3.15 Impact of Core Design and Licensing Criteria on BWR Capacity

- Issue

The Reed Report contained a table that identified several potential problems, some having safety significance, that could affect plant availability and capacity factor.

- Safety Significance

The concern was primarily economic, with shutdowns and power derating resulting from either equipment problems or from a licensee's inability to meet regulatory requirements.

- Status

These problems have been resolved through the following:

(1) Fuel densification problems were resolved by changes in fuel design.

(2) Emergency core cooling system criteria in Appendix K of Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) have been satisfied.

(3) Channel box wear and cracking was caused by flow-induced vibration of incore instrument and startup source tubes. The problem was resolved by eliminating bypass flow holes in the lower core plate and adding two holes in the lower tie plate of each assembly to provide an alternate flow path. (See also discussions in Sections 3.5 and 3.18 on channel box problems.)

All other problems listed affecting plant availability and capacity factor are identified and addressed elsewhere in this evaluation.

3.16 Adequacy of Design Procedures To Ensure Compliance with Licensing Criteria

- Issue

The Reed Report raised the following concerns regarding quality assurance (QA) for the BWR-6:

(1) GE had no identifiable systems engineering organization to provide independent evaluations of BWR designs at critical points in the program.

(2) GE's existing procedures for BWR systems design reviews needed improvement, and additional procedures were needed for QA for the BWR-6.

- Safety Significance

There was a lack of confidence that applicable licensing requirements would be implemented and documented.

• Status

At the time the Reed Report was issued, the GE nuclear QA program for the BWR-6 had not been completed. Since then, a program has been completed that complies with the applicable NRC requirements, codes, and standards, and the NRC has given its approval for operating license applicants to reference this QA program in the Final Safety Analysis Report for a plant. This GE report (NEDO-11209-03A and 04A, currently approved by NRC staff through Revision 6 dated July 1986) describes the approved QA program for design, fabrication, and procurement activities involving safety and safety-related structures, systems, and components of GE nuclear power plants.

3.17 Consistency of Degree of Verification of Calculational Models

• Issue

The Reed Report raised a concern that calculational models were not thoroughly reviewed and verified by comparison to experimental data to ensure adequacy.

• Safety Significance

A calculational model that is not adequately verified by comparison to results using experimental data can lead to nonconservative errors in results, and uncertainty in operating limits derived from reactor safety analyses.

• Status

GE has completed major experimental programs for verification of currently approved models, and verification problems have been resolved. The NRC staff has reviewed and approved all calculational models that are necessary to be used in licensing of operating BWR plants.

3.18 Possibility of Control Rod Binding Due to Fuel Channel Creep

• Issue

The Reed Report noted that fuel channel life was projected to be 8 to 10 years (two complete refueling cycles) rather than the desired 15 years, due to thermal creep and control rod binding.

• Safety Significance

Binding of control rods can cause slower negative reactivity addition, thereby invalidating the licensing assumptions and increasing the severity and consequences of transients and accidents.

• Status

Today, fuel channel shuffling requirements and scram-time testing technical specifications ensure against degradation in scram time.

The NRC staff has approved channel surveillance programs, in conjunction with relocation and rotation to minimize irradiation-induced channel bow, and special rod motion testing for core cells exceeding core residence program guidelines as ways to extend channel lifetime.

3.19 Compliance of Design Work and Reviews with Written Procedures

- Issue

The Reed Report identifies the following concerns regarding the BWR-6 Mark III that arose from findings of a GE internal audit:

(1) Design reviews, internal procedures, and QA audits were not always conducted in conformance with established written procedures.

(2) QA audits conducted by GE revealed instances of nonconformance with BWR Systems Department engineering practice and procedures.

(3) Staffing and organization of design assurance efforts in the BWR Systems Department did not optimize its effectiveness in departmental activities.

(4) There was a lack of coordination between the procedures and QA (P and QA) organization and GE components audited.

- Safety Significance

A proper internal audit program is needed to ensure that inadequacies in procedures and noncompliance with procedural requirements will be discovered and corrected.

- Status

As described previously, the GE QA program has been reviewed by the NRC staff, and GE now has an effective internal audit program, a part of the GE Quality Assurance system. GE audit reports are available to and inspected by the NRC. Experience has demonstrated that the GE program is effective in finding deviations and deficiencies, as it was designed to do.

3.20 Absence of Availability Goals in Design Procedures

- Issue

The Reed Report discusses instances of nonconformance with GE procedures involving issues that are basic to the achievement of design integrity and that affect plant availability. In particular, the study was concerned with achieving an optimal balance in the engineering design goals between availability and safety. The study noted in particular that many design procedures did not have availability goals.

- Safety Significance

The absence of availability goals, by itself, has no impact on safety-related design integrity. Regarding availability goals, in its licensing reviews the NRC uses safety design requirements as found in its regulations, its Standard Review Plan (NUREG-0800), its Regulatory Guidelines, and other NRC position papers, rather than availability goals.

- **Status**

Although the NRC has not established quantitative availability requirements for safety systems, unit availability can be limited by technical specifications that prevent startup and require shutdown when key safety systems are unavailable. All operating BWRs have such technical specifications, and plant availability can be affected by these technical specification limits.

3.21 Seismic Capabilities of 8 x 8 Fuel Spacer

- **Issue**

The Reed Report raised a concern related to the seismic capability of spacer design for 8 x 8 fuel. Specifically, potential loss of core coolability because of fuel spacer failure under the combined loading of an earthquake and a loss-of-coolant accident (LOCA) was envisicned as a possible impediment to licensability.

- **Safety Significance**

Maintaining the core in a coolable geometry during seismic event helps limit the consequences of a postulated LOCA to acceptable release levels.

- **Status**

GE has completed the seismic testing of the fuel assembly spacer and has reported the results in NEDE 21175-3-P-A, dated October 1984. The NRC staff has reviewed those results and accepted the design for ose in BWR cores.

3.22 Extent of Life of Position Sensor in Traversing In-Core Probe System

- **Issue**

The Reed Report addresses operational problems with the traveling in-core probe (TIP) system, including bending and contamination of the guide tubes.

- **Safety Significance**

Technical specifications and plant procedures require periodic calibration of local power range monitors that input to reactor protection systems using the TIP system. Power distribution information obtained from the TIP system is used to maintain core operating limits. Unavailability of the TIP system would prevent plant operators from obtaining certain information necessary for starting up the plant. Unavailability of the TIP system could then adversely affect plant availability.

- **Status**

Service experience with modified TIP systems designed for better availability demonstrates that longer life and improved accuracy (compared with earlier models) is being achieved. Efforts to further improve the operational usefulness and dependability of the TIP system are ongoing.

3.23 Radiation Levels Outside Biological Shield and Drywell

- Issue

The Reed Report noted that unexpected and excessively high levels of radiation outside the biological shield and/or drywell containment would constitute an occupational radiation exposure problem.

- Safety Significance

Imperfections in shielding design can result in unexpected radiation streaming through unrecognized pathways. This is a personnel radiation exposure problem. It also creates difficulties in maintaining and servicing affected parts of the plant when radiation levels are high.

- Status

High levels of shine radiation were observed during startup, particularly in early plants. However, this is no longer a problem in operating plants. To prevent such occurrences, it is standard practice to perform startup radiological surveys to confirm radiation levels and to identify unexpected ones. Licensees have identified all such pathways by actual surveys and have eliminated them. Such programs ensure that radiation exposure levels for workers do not exceed NRC established limits and conform to ALARA guidelines.

3.24 Stress Corrosion Cracking in Dresden 1 Control Rods

- Issue

The Reed Report noted that control rod lifetimes might be limited because of stress corrosion cracking in the control rod blades. This could lead to problems of

(1) limited control rod life
(2) loss of reactivity worth (leaching of absorber material)
(3) continued operability (cracking of sheath)

- Safety Significance

There is a potential for reducing the shutdown margin to below that required by technical specifications.

- Status

GE has performed an analysis of the safety implications of control rod cracking and consequent loss of rod worth. The results show that any loss of reactivity worth would be revealed by a shutdown margin test before the loss could jeopardize safe shutdown capability of the reactor. In addition

(1) Problems with control rod blades identified through operating experience were resolved by licensee actions in response to NRC IE Bulletin 79-26, Rev. 1, "Boron Loss from BWR Control Rod Blades," dated August 28, 1980.

(2) Later problems involving cracking of advanced design blades in the sheath at the handle region have been evaluated and are being addressed by a continuing surveillance program.

(3) Improved hybrid-hafnium control rod designs and better control of water chemistry have alleviated, but not eliminated, the problem of control rod blade degradation with use.

The broader issue of stress corrosion cracking in stainless steel piping associated with nuclear reactors is addressed in Section 4.6 of this report.

3.25 Peak Pressures in ATWS Calculations for BWR-3 Plants

• Issue

The Reed Report noted a potential for damage to the reactor vessel due to possible peak pressures of 1600 to 1650 psig during certain postulated events for the BWR-3, particularly the anticipated transient without scram (ATWS) event.

• Safety Significance

Overpressurization and failure of a reactor vessel would result in consequences beyond those acceptable for licensing a nuclear power plant.

• Status

Such pressures resulting from a transient event could occur only at elevated temperatures when the pressure vessel material is in a ductile state and is thus less subject to damage by an overpressure event. Further, more refined calculations by GE using better analytical methods demonstrate that peak pressures in such an event would be far less than the 1600 to 1650 psig estimated in 1975.

Interim resolution of the ATWS issue was provided by improved procedures and operator training, and through implementation of certain hardware modifications (e.g., recirculation pump trip). The ATWS issue was finally resolved when NRC issued the ATWS rule (Title 10 of the Code of Federal Regulations, Section 50.62 (10 CFR 50.62)), in July 1984. In response to this rule, plant-specific measures, including hardware modifications, have been made in all operating BWR plants, and further modifications will be made in some plants. In October 1986, the NRC accepted the GE licensing topical report NEDE-31096-P, "Anticipated Transients Without Scram; Response to NRC ATWS Rule, 10 CFR 50.67," which means that licensees may now reference this report in their plant-specific actions.

4 OTHER SAFETY-SIGNIFICANT ISSUES IDENTIFIED BY THE NRC STAFF IN THE REED REPORT

Through its most recent review and evaluation of the Reed Report, the NRC staff identified several safety-significant issues in the report that had not been highlighted by either the NRC staff or by GE in its 1978 status report on the Reed Report. These are identified and discussed below.

None of these issues involve any safety consideration not already identified and appropriately addressed by the staff.

4.1 Combination of LOCA Induced Loads and Safety Relief Valve (SRV) Actuation Loads for Mark III Containments

- Issue

The Reed Report, in the section entitled Mechanical Systems and Equipment, cites a concern that the NRC might require applicants/licensees to consider combined LOCA-induced hydrodynamic loads and SRV loads in the evaluation of suppression pool loading phenomena and design of the Mark III containment. The report further notes that it is not unreasonable to postulate SRV operation concurrent with a LOCA.

The GE status report did not explicitly identify this issue. This issue could, however, be considered a component of the overall issue of hydrodynamic phenomena identified and discussed previously. The Reed Report recommended that a high priority be assigned to the resolution of this issue, and that conservative containment design loads should be used by architect/engineers in the design and construction of plants. This approach was suggested to minimize the likelihood that future redesign or plant modifications would be needed after testing and the NRC review were completed.

- Safety Significance

The safety significance of this issue, as acknowledged in the Reed Report, is that the combination of LOCA and SRV loads could result in a higher total loading condition. The larger loads could threaten the integrity of the containment structure under accident conditions, or could reduce the safety margins in the design.

- Status

NRC now requires applicants/licensees to consider the combination of SRV and LOCA suppression pool loads; however, the NRC has evaluated and approved the GE methodology for the combination of these hydrodynamic loads. NUREG-0798 documents resolution of this issue for the Mark III containments as part of the resolution of Generic Issue B-10, "Behavior of BWR Mark III Containment"; resolution of this issue for the Mark I and Mark II containments was documented as part of the resolution of Generic Issues A-6, A-7, and A-8.

4.2 Jet Impingement on the Weir/Pool in a BWR Mark III Containment

- Issue

The Reed Report, in the section entitled Mechanical Systems and Equipment, included the following recommendation: "The possibility of a direct pipe break jet impingement on the weir/pool and its asymmetrical effects should be examined. Preliminary judgement is that this is not serious." The NRC staff was unable to locate any other clarifying information on this issue in the report. This issue was not identified or discussed in the GE status report provided in 1978.

- Safety Significance

If direct pipe break jet impingement on the weir/pool were to occur, the jet impingement loads could cause structural failure of the weir wall. Failure of the weir wall in the extreme could cause an uncovery of the suppression pool vents which, in turn, would lead to bypass of the suppression pool. For certain accidents, significant steam bypass of the suppression pool could result in overpressure failure of the containment. If the asymmetric suppression pool loads on the weir wall were sufficiently large, they would have the same consequences.

- Status

Jet impingement effects resulting from postulated pipe breaks are not unique to BWR Mark III containments and are addressed for all plants during the course of licensing review. The general consideration of jet impingement loads on structures and equipment includes those effects, if any, on the weir wall in a Mark III containment. For asymmetric suppression pool loads, the effects of such loads on the weir wall is minimal, because they are bounded by other weir wall loads (e.g., chugging load, depressurization load). Asymmetric pool swell loads were addressed in NUREG-0978, in the resolution of Generic Issue B-10, "Behavior of BWR Mark III Containment."

4.3 Main Steam Isolation Valve Leak Tightness

- Issue

The issue of leak tightness of main steam isolation valves (MSIVs) was identified in the Reed Report in the section on Mechanical Systems and Equipment, but was not discussed in the GE status report provided in 1978.

Main steam isolation valves (MSIVs) have been notorious for leaking at high rates when they are tested during the 18-month leak tightness testing that is generally required by the technical specifications. Most plants have a technical specification leak rate limit of 11.5 standard cubic feet per hour (scfh) per valve. At some plants the as-found leak rate has been as high as 4500 scfh. With such high leak rates, the MSIV-leakage control system (MSIV-LCS) probably would not be capable of performing its safety-related function of removing the leakage from between the closed MSIVs following a design-basis LOCA.

• Safety Significance

In its evaluation of the safety features of nuclear power plants, the past practice of the staff has been to give no credit for any structure, system, or component that was not safety related (sometimes referred to as safety grade). Given this past practice, following a design-basis LOCA with no credit for non-safety-related components, and assuming the single failure of one MSIV to close, the design-basis maximum allowable leakage through the MSIVs, for most plants, is 11.5 scfh. This limit on MSIV leakage is to maintain the offsite radiological consequences to within a small fraction of regulatory limits in the event of an accident. Thus, if the MSIVs were to leak at a rate greater than 11.5 scfh, and particularly at a rate that caused the MSIV-LCS to fail, the offsite consequences could exceed the regulatory limits in the event of a severe accident.

• Status

In recognition of this continuing problem of MSIV leakage, and the potential consequences in terms of offsite doses, the NRC staff early initiated Generic Issue C-8, "MSIV Leakage and Leakage Control Systems Failures." This generic issue considered the actual natural phenomena associated with the behavior and the characteristics of radioactive materials and the historical capability of "nonsafety-related" components to survive seismic events. In assessing the consequences of MSIV leakages, credit was given for fission product decay, plate-out on cold surfaces, and gravitational settling, and for a realistic evaluation of the actual materials that would be transported along the main steam line.

Because it is assumed in design-basis accient analyses that offsite power will be lost following a LOCA (as a result of the tripping of the turbine generator and failure of offsite power), no credit was given for any equipment that was not powered from the emergency diesel generator busses. The analysis performed under Generic Issue C-8 indicates that the leak rate through MSIVs could be as high as 1500 scfh without using the MSIV-LCS, and the offsite doses would be less than those specified in the regulations. The study identified a method of calculating this leakage rate, but the actual leak rate would have to be determined on a plant-by-plant basis. This information was documented in NUREG-1169, published in August 1986.

MSIV leak tightness was a concern in 1975, and it is still a concern that has not been fully resolved. The BWR Owners Group (BWROG) formed a committee to evaluate this same issue independently, with GE giving technical support to the BWROG committee. This committee generally found that the high leakage rates were attributable to valve maintenance practices. For those plants that have adopted the BWROG recommendations resulting from their evaluation, the as-found MSIV leak rates have generally been within the plant-specific technical specification limit, or within a factor of 2 or 3 of that limit. For example, Peach Bottom 3, had typical as-found leak rates of over 3000 scfh for each of the MSIVs. After following the BWROG recommendations, the next as-found leak rates were found to be less than 11.5 scfh for seven of the eight MSIVs and approximately 14.7 scfh for the eighth MSIV. This demonstrates that the MSIVs can be maintained within their respective technical specification leakage limits, and that the use of the leakage control system is not necessarily the optimum method for handling the leakage through the MSIVs in the event of a LOCA.

The technical specification limit of MSIV leakage is conservatively set to ensure that offsite dose consequences of a main steam line break are a small fraction of the regulatory limits in 10 CFR Part 100. Although MSIV leakage is an issue of continuing concern, the current state of the art and conservative limits justify continued operation of BWR plants as the MSIV leakage issue is pursued.

4.4 Control of Design of Purchased Components

• Issue

The Reed Report identifies concerns that

(1) Because GE's Nuclear Engineering Department (NED) relies almost entirely on other vendors' design expertise to produce components to purchase specifications, GE needed to develop more engineering competence and design expertise in hardware purchased from vendors, particularly valves (e.g., main steam isolation valves, safety relief valves, flow control valves, etc.).

(2) GE needed to implement a procurement policy that provides for engineering reviews and approval of design details for materials of critical components that are purchased from vendors.

• Safety Significance

The failure of purchased components used in GE safety systems or in systems important to safety could prevent those systems from performing their intended functions.

• Status

Currently, purchased components used in GE nuclear systems are appropriately considered in the GE QA program. (See also Sections 3.16 and 5.6 of this report.)

4.5 Flow-Induced Vibration of Jet Pumps

• Issue

The Reed Report raises a concern that inherently high excitation due to turbulence in the upper end of jet pumps could lead to mechanical failures caused by flow-induced vibration.

• Safety Significance

Jet pump mechanical failures could invalidate the licensing basis LOCA analyses through a failure to maintain the assumed vessel water level at the top of jet pumps during reflood.

• Status

Subsequently, tests performed by GE demonstrated that major structural components should withstand anticipated vibratory stress levels. However, operating experience revealed a problem with the holddown beams, which cracked in some operating reactors. The problem was addressed by design changes to the holddown beams and by appropriate surveillance programs and technical specification surveillance requirements to monitor jet pump operability. These recommendations and requirements were in NRC IE Bulletin 80-07, "BWR Jet Pump Assembly Failure," dated April 4, 1980; they included the use of improved holddown beam bars and a required surveillance program to anticipate incipient beam bar failure that could result in displacement of the jet pump assembly.

4.6 Stress Corrosion Cracking in Stainless Steel Piping

- Issue

The Reed Report notes that stress corrosion cracking (SCC) has occurred in type 304 stainless steel piping in several operating BWRs and that SCC has occurred in nitrided stainless steel parts, furnace-sensitized components, and in bolts that have been heavily cold-worked.

The Reed Report recommended that GE develop replacement materials, expand studies on materials, expand study on stress levels, increase efforts on environmental effects on fatigue for water chemistry control, and study the relationships between operating practices and cracking.

- Safety Significance

Several studies have shown that pipe cracking has minor safety significance. Both experience and analyses have shown that cracks in pipes caused by stress corrosion cracking will develop readily detected leaks before cracking develops to the point that complete pipe failure will occur. Nevertheless, the NRC staff has determined that reliance on this leak-before-break behavior is not sufficient. Appropriate remedial measures -- including augmented inspections to detect cracking in early stages -- and corrective actions are required where appropriate (see NRC Generic Letter 84-11, dated April 19, 1984).

- Status

Since 1975, extensive cracking has been discovered in stainless steel piping in BWRs. The NRC has established two Pipe Crack Task Groups and implemented their recommendations. The industry also has mounted an extensive effort to address the problem and develop remedies. As a result of cracking observed in large and small stainless steel pipes in recent years, all operating BWRs having susceptible piping have implemented an NRC staff-prescribed surveillance program, with staff-approved pipe repair or replacement where appropriate.

Currently, a comprehensive set of guidelines that provides the NRC positions on actions to control pipe cracking in BWRs is under development. The NRC staff has prepared a generic letter, together with a technical report (NUREG-0313, Rev. 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"), that will be issued shortly. This letter and report set forth the actions that plant owners must take to keep their plants in conformance with NRC requirements related to piping integrity.

Because construction of BWR-6 models was relatively recent, the materials and process used for their piping were highly resistant to stress corrosion cracking, and are in almost complete conformance with the proposed NRC guidelines. If, in accordance with the forthcoming generic letter, individual welds are found to be not in conformance with the materials and process guidelines, augmented inspections will be required to ensure the continued integrity of the piping.

5 THE GENERAL ELECTRIC SUBTASK GROUP REPORTS

This section contains the NRC staff evaluation of each of the subtask reports that were prepared as input to the Reed Report. The reports address the following topics:

Subtask	Topic
A:	Nuclear Systems
B:	Fuel
C:	Electrical, Control, and Instrumentation
D:	Mechanical Systems and Equipment
E:	Materials, Processes, and Chemistry
F:	Production, Procurement, and Construction
G:	Quality Control Systems Overview
H:	Management/Information System
I:	Regulatory Considerations
J:	Scope and Standardization

In its own evaluation of these reports, the NRC staff has attempted to identify any issues having safety significance, and to indicate the status of the issue so far as the NRC staff is concerned. The staff found no issues of safety significance that have not already been addressed by NRC staff initiatives, with the possible exception of a plant auxiliary power systems issue identified in the Subtask C report (Section 5.3).

5.1 Subtask A: Report on Nuclear Systems

INTRODUCTION

The subtask report on nuclear systems deals primarily with several issues expected to necessitate reducing the allowed power level of reactors (power derating) during portions of the core operating cycle. These issues stemmed from a marketing strategy that required GE to commit to designs of increasing size and performance before the designs were adequately verified via test data and field experience. Additionally the advanced designs were standardized on the basis of earlier designs before sufficient field experience feedback could be considered. The GE task force was concerned that reliability/availability considerations would be major factors in future procurement evaluations by the utilities, and that field experience with BWRs, especially with the BWR-6, would not reflect favorably on the product.

SUMMARY OF ISSUES

Most of the issues involving systems aspects of BWR NSSS design that were perceived as contributors to power derating in the 1975 study are addressed in the the Reed Report. The safety significance and current status of the following Subtask A issues are discussed in Section 3 of this report in the listed subsection:

Issue	Subsection
Amount of Margin Between Design Calculations for Core and Operating Limits	3.2

Impact of Cold Shutdown Margin on BWR-6 Core Design	3.3
Impact of EOC Scram Reactivity Insertion Rate on Core Full Power Life	3.4
Degree of Proof of Accuracy of Transient Design Methods	3.6
Impact on Fuel Integrity of Reduced Moderator Temperature due to Equipment Failure	3.7
Impact of Core Design and Licensing Criteria on BWR Capacity (New ECCS Criteria)	3.15
Consistency of Degree of Verification of Calculational Models	3.17
Radiation Exposure from Removal of Steam Dryer/Separator Assembly	3.10

In its detailed review of the subtask report, the NRC staff identified several subissues that are presented in more detail or in a different context from the discussion of the above issues in the Reed Report. A discussion of these additional issues which impact plant availability, and their safety significance, follows.

- Regulatory Backfit

  Issues: Sixteen issues expected to require backfit to plants under construction were identified.

  Safety Significance: Some backfit issues identified were necessary to meet new regulatory requirements, and some were not.

  Status: Changes were implemented where appropriate.

- Incomplete Design

  Issue: Reload cores and behavior of equilibrium cores were not factored into the design process for the early BWR-2 to -5 designs. Transient characteristics of BWR-2 to -6 designs were not assessed until after the core and circulating systems designs were frozen for hardware procurement. Seismic design analyses were performed after hardware layout was complete, and the level of effort was insufficient to complete the design properly.

  Safety Significance: The economic penalty of the failure to show design margin to operating limits in frozen designs and in reload cores creates undue pressure to compensate for design shortcomings via the application of nonconservative and unverified calculational methods, which could result in violation of fuel integrity or LOCA operating limits.

- Status: Reactors were licensed based on the results of safety analyses using NRC reviewed and approved calculational methods. The regulations require that reload core designs involving unreviewed safety questions or technical specification changes (e.g., core operating limits) be approved by the NRC staff prior to implementation.

- Uncertainties in Reactor Core Design Methods

Issue: The design thermal margin to operating limits was found to be significantly less than that predicted based on field measurements, showing discrepancies between predicted and measured void reactivity worth and a 10% underprediction of the depletion rate of gadolinium rods. The 25% margin provided in initial transient design analysis eroded to 10% by the void model error, and additional uncertainties that could further erode thermal margin were identified.

Safety Significance: These reactor core design models are used to establish technical specification operating limits for fuel integrity and LOCAs and to evaluate the consequences of transients and accidents.

Status: Improved calculation models have been developed and verified using experimental data and plant transient tests. These models have been reviewed and approved by the NRC staff and were used in the final safety analyses (and for reload core designs where appropriate) for most operating BWRs. Where uncertainties exist in these methods, NRC requires that they be quantified and applied conservatively in the licensing safety analyses and, in some cases where pre-operational verification is not feasible, requires the licensee to perform confirmatory verification.

Reactors are operating at or near the operating limits (not safety limits) during much of the core operating cycle. Extensive power derating has been avoided via new fuel designs, better modeling to minimize the use of bounding safety analyses, and detailed analyses of reload cores to ensure that core management schemes and fuel-cycle-dependent technical specifications provide maximum operating flexibility.

Licensees must maintain adequate safety margins by adhering to technical specification operating limits.

- Void Coefficient/Relief Valves

Issue: The void coefficient used in BWR transient design resulted in reactivity addition following an isolation (turbine-generator trip) that was too small by a factor of 4.3 for BWR-6 equilibrium cores as a result of changes in reactor characteristics and more realistic modeling. Design scram reactivity is reduced by a factor of 5 for the EOC equilibrium core due to the high reactivity in voids. Protection against overpressure transients of greater severity is provided by additions of relief valves, trip circuitry, and fast scram drive blades. There was concern that increase in the number of pressure relief valves and the number of challenges to these valves would significantly increase plant unavailability.

Safety Significance: Greater reliance is placed on safety relief valve performance to protect against overpressure transients that challenge pressure limits on the vessel and thermal limits on the fuel.

Status: Other design changes -- such as less negative fuel void coefficients, the fast scram drive on BWR-6s, and recirculation pump trip provisions -- in conjunction with improved scram calculation models have reduced the severity of the transient. There is no noticeable increase in plant unavailability due to pressure relief transients.

Flow Control Range

Issue: The operating flow control range was reduced for BWRs of higher core power density; for BWR-6 the nominal range was 75% to 100% versus 50% to 100% in earlier BWR-3 designs. The reduction in range was necessary to meet the design stability criterion of 0.25 decay ratio (damping factor) for equilibrium cores at EOC.

Safety Significance: The restricted flow control range reduces operating flexibility and requires more frequent control rod movement, which tends to increase fuel failures.

Status: Fuel design improvements have reduced susceptibility to PCI failure related to control rod movement. The resolution of Generic Issue B-19, "Thermal Hydraulic Stability," permits plants to operate at higher stability decay ratios, which permits removal of the design restriction on flow control range.

CONCLUSIONS

The NRC staff has reviewed the nuclear systems subtask report and finds no new issues with potential safety significance that should be addressed. The staff notes that appropriate technical specifications ensure that problems involving reactor operating flexibility and plant capacity are not alleviated at the expense of safe operating limits; such technical specifications are in place on operating reactors, and any changes in reload fuel design, which has been identified as a recommended action to avoid power derating, are subject to NRC review where required by 10 CFR 50.59 for impact on safety.

5.2 Subtask B: Report on Fuel

INTRODUCTION

The subtask report on fuel deals primarily with the design and performance limitations of the fuel and related core components in the context of their impact on the reliability and availability of BWRs. Because pellet cladding interaction (PCI) of the GE 7x7 fuel was the predominant fuel problem at the time of GE's 1975 study, fuel preconditioning operating recommendations and design changes needed to resolve the PCI problem received most of the attention. There were also concerns that regulatory requirements based on the ALARA principle could increase the obstacles to design improvement and changes through more comprehensive and conservative fuel design models for transient analysis, more extensive proof of performance for design changes, and technical specifications enforcing PCI operating recommendations.

SUMMARY OF ISSUES

The principal issues in this subtask report are addressed in the Reed Report. The safety significance and current status of the following Subtask B issues are discussed in Section 3 of this report in the listed subsection.

Issue	Subsection
Impact on Fuel Integrity of Reduced Moderator Temperature due to Equipment Failure	3.7
Impact of Core Design and Licensing Criteria on BWR Capacity	3.15
Possibility of Control Rod Binding due to Fuel Channel Creep	3.18
Seismic Capabilities of 8 x 8 Fuel Spacer	3.21

In its detailed review of this subtask report, the staff identified two additional issues that warrant attention. A discussion of these issues, and their safety significance, follows.

- End of Life Failure Modes

  Issue: Fuel performance data at the time of GE's 1975 study was limited to 15 to 20 GWD/T exposure. There was concern that after resolution of the PCI problem, failures would occur from exposure-related problems such as

  - fuel swelling due to fission products contained in the fuel
  - failure or distortion of cladding due to fission gas pressure
  - thermal fatigue of cladding
  - failure of cladding due to corrosion
  - failure of cladding due to fretting and wear by spacers
  - weld area penetration

  Status: Analytical models for design prediction of extended burnup performance have been developed and approved by the NRC staff. BWR fuel has been approved for operation to extended burnup of 40 GWD/T batch average exposure. Operating experience with BWR fuel in excess of 30 GWD/T has not revealed any significant performance problems with extended burnup fuel.

- Incipient Cracks

  Issue: Unfailed fuel of moderate exposure may contain multiple incipient cracks, which makes the fuel susceptible to failure under unusual stress.

  Safety Significance: This could cause under-prediction of core damage and radiological consequences associated with transients and accidents.

Status: Operating experience has not shown any problems associated with this failure mechanism. Conservative fuel failure criteria would bound such failures if they did occur; for example, in the licensing basis safety analysis, any fuel driven to a critical heat flux power level is assumed to fail (a conservative assumption).

CONCLUSIONS

The NRC staff has reviewed the fuel subtask report and finds no new issues with potential safety significance that should be addressed. The predominant fuel problem (PCI) at the time of GE's study has been substantially resolved, and there are no new problems associated with currently approved fuel designs or with operation at extended burnup.

5.3 Subtask C: Report on Electrical, Control, and Instrumentation Systems

INTRODUCTION

This subtask report addressed the design process for the electrical, control, and instrumentation systems to assess the adequacy of design methods and approaches to produce the required product performance, quality, and availability. In addition, design uncertainties were identified and corrective actions recommended.

SUMMARY OF ISSUES

The NRC staff review of this subtask report addressed the specific areas discussed below.

- BWR Dynamic Control System -- Dynamic Control and Load Following Capability

  Issue: The subtask report recommends that GE perform an overall systems evaluation of the technical feasibility of, and the economic justification for, modifying the BWR dynamic control system to provide increased capability for normal electrical grid frequency control duty and for coping with network disturbances (such as might lead to isolated grid operation). It also recommends that GE evaluate a joint internal effort in this regard.

  Status: Dynamic control with load-following capability is not generally approved for BWR plants, but the NRC will review applications for this capability on a case-by-case basis. This issue did not raise any new safety concerns.

- BWR Dynamic Control System -- Pressure Control System

  Issue: The subtask report recommends that GE always have on hand, in San Jose, one set of qualified pressure control system hardware, so that if problems arise overseas, there is a quick and effective way to test and evaluate solutions. In addition, the report recommends that the responsibility for at least the electrical components of the pressure control system be transferred to GE's control and instrumentation group.

  Safety Significance: This issue did not raise any new safety concerns.

• <u>BWR Dynamic Control System -- Automatic Load-Following System</u>

<u>Issue</u>: The subtask report recommends that GE's nuclear engineering group become thoroughly acquainted with the advantages and disadvantages of various electronic variable-speed pump drives for recirculation flow, to determine if they might serve as a backup for the flow control valve and to ensure themselves that the valve system is really warranted in view of potential availability advantages of the variable-speed systems. In addition, the report recommends that GE consider, and have designs for, alternatives to the non-linear 3-mode controller.

<u>Status</u>: This issue did not raise any new safety concerns.

• <u>BWR Dynamic Control System -- Feedwater Control System</u>

<u>Issue</u>: The subtask report does not provide any recommendations concerning this issue. The NRC staff has recognized that there are operational problems associated with the feedwater control system. All of these problems fall into the operational category (not safety related). All BWRs will include a feedwater trip to limit vessel high-level transients as required for the resolution of NRC's Unresolved Safety Issue A-47. Other initiatives in important-to-safety balance-of-plant systems such as feedwater systems are being considered by the NRC staff.

<u>Safety Significance</u>: This issue did not raise any new safety concerns.

• <u>BWR Dynamic Control System -- Relief Valve Augmented Bypass (REVAB)</u>

<u>Issue</u>: The subtask report recommends that GE review the ability of REVAB to meet its design objectives and consider modifying the REVAB operational objectives, in light of potential impacts on plant operational availability. In addition, the report suggests that GE review alternative means for providing the capability to accept loss of electrical load without reactor scram, and compare them with REVAB (on technical and economic bases) to form the basis for GE's future approach in this area.

<u>Status</u>: REVAB has not been installed on any GE BWR in the United States. This issue did not raise any new safety concerns.

• <u>Control Rod Drive System</u>

<u>Issues</u>: The subtask report recommends that GE

(1) continue its program for fast-scram development, ensuring 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. GE should also ensure that adequate developmental test facilities are available for testing of prototype drives with blades, under pressure, temperature, clearance, and water quality conditions to be encountered in operation.

(2) initiate a program in parallel with the present evaluation/redesign of the control rod drive. Specifically, GE should evaluate the potential for a "Vernier motion" added to the planned hydraulic fast-scram drive.

Status: This issue did not raise any new safety concerns. The design of the rod drive system for the BWR-6 has been reviewed and approved by the NRC, and the timing of the rod insertion for a scram has been taken into account in the Final Safety Analysis Report for BWR-6s, and is periodically verified through surveillance tests.

- Reactor Safety System -- Setpoint Drift

Issues: The subtask report recommends that GE

(1) continue to give the required priority to this problem and its corrective program to ensure that GE's schedule for issuance of Engineering Change Authorizations is met

(2) take the initiative with its customers, and with NRC, to ensure that the required changes are implemented on a timely basis

Status: Setpoint drift is being reviewed by the NRC staff. GE established a setpoint methodology program in the early 1980s and issued NEDC-31336, "General Electric Instrument Setpoint Methodology," which seeks to confirm the adequacy of protection system setpoints, including allowances for drift. NRC is reviewing NEDC-31336. This issue did not raise any new safety concerns.

- Reactor Safety System -- Solid-State Safety System

Issues: The subtask report recommends that GE

(1) at the proper time in the detail design stage, implement design review of measures taken to ensure acceptable electrical noise immunity in the system, using some knowledgeable people from other divisions or outside GE

(2) continue to review the relative reliability of ac solid-state drivers and contactors as output elements, to establish expected lifetimes before making a final design commitment

Status: NRC reviewed and approved the safety aspects of the solid-state reactor protection system during the Clinton operating license review. The results of this review are discussed in NUREG-0853, "Safety Evaluation Report Related to the Operation of Clinton Power Station," dated February, 1982. The system is presently operational with no ongoing safety concerns. This issue did not raise any new safety concerns.

- Neutron Monitoring System

Issues: The subtask report recommends that GE

(1) defer to its Task Force 6 for recommendations on the incore sensors

(2) review the Traveling Incore Probe designs to evaluate more effective solutions to both the position read-out and guide tube concerns

Status: This issue did not raise any new safety concerns.

- Other Instrumentation Systems

Issue: The report did not provide any recommendations; it stated that specific problems that have occurred seem to be adequately resolved.

Status: This issue did not raise any new safety concerns.

- Power Generation Control Complex (PGCC)

Issue: The report did not make any recommendations regarding the PGCC.

Status: The NRC staff has reviewed and approved the PGCC during several operating license case reviews (e.g., Susquehanna, Nine Mile Point 2, LaSalle). This issue did not raise any new safety concerns.

- NUCLENET Complex

Issues: The report recommends that GE

(1) complete two technical design reviews on display control system (DCS) in 3rd quarter of 1975 and 1st quarter 1976, utilizing some technical experts from outside the nuclear engineering department. In the future, these reviews should be done routinely using such outside experts.

(2) confirm that its staff is capable of maintaining the first NUCLENET hardware system.

(3) make maximum use of interactive computer graphics for the printed circuit board work.

(4) obtain early data on the reliability of the 4400 computer.

(5) explore the opportunities to use Honeywell-PCD standard software as a basis for DCS system.

(6) review the plans for field maintenance of NUCLENET systems to ensure that someone is doing the test and diagnostic programming and procedures work necessary to keep the equipment operating in the field.

Status: NRC reviewed the safety aspects of NUCLENET during the Clinton operating license review. The results of this review are discussed in NUREG-0853, "Safety Evaluation Report Related to the Operation of Clinton Power Station," dated February 1982. The system is presently operational, with no ongoing safety concerns. This issue did not raise any new safety concerns.

- <u>Plant Auxiliary Power Systems</u>

<u>Issues</u>: The report recommends that GE

(1) give its customers increased application engineering assistance to emphasize the need for greater main switchyard redundancy to improve plant availability.

(2) specify the redundancy and other special requirements of power supplies provided by the customer for non-safety-related GE systems affecting a plant's availability. These specifications should include electrical, pneumatic, and hydraulic supplies, at all power levels.

(3) centralize the responsibility for power supplies for all GE systems to enable an effective approach to power supply/plant unavailability problems. In addition to documenting and coordinating all power supply requirements for availability-related systems, an important part of this effort should be convincing the customer of the benefits of meeting these requirements.

<u>Status</u>: During the licensing review of recent BWR operating license applications (e.g., River Bend, Perry, Nine Mile Point 2), the NRC staff has been unable to find consistency in a utility's characterization of the Class 1E/non-Class 1E boundaries associated with the reactor protection system (RPS) power supplies. In fact, in some cases, an individual utility has been confused as to the location(s) of this boundary. This has led to various separation, physical identification, seismic, and Class 1E/non-Class 1E interface concerns regarding RPS bus A and B. The staff believes that if the third recommendation had been followed for the RPS power supplies, the confusion regarding the concerns addressed above would have been alleviated. The staff is reviewing this issue to determine if it should be considered further, possibly as a generic issue.

- <u>C&I Availability/Reliability/Maintainability Program</u>

<u>Issues</u>: The report recommends that GE

(1) show a greater concern for and preoccupation with the safety aspects of nuclear design. In non-nuclear projects, the safety aspects are easier to address and, therefore, require less utilization of resources and regulatory involvement.

(2) develop its nuclear projects to the same order of operational reliability that customers for non-nuclear projects (NASA, DOD, etc.) demand.

(3) encourage greater reliability efforts. In non-nuclear projects, the customer (NASA, DOD, etc.) demands, funds, and monitors a reliability program. In the nuclear industry, NRC provides a reliability standard for protection systems but does not fund the effort. GE's utility customers are not known to either require or fund reliability efforts.

(4) have its nuclear engineering department reliability and maintainability plan objectively reviewed by knowledgeable GE personnel outside the department.

(5) strengthen its nuclear engineering department problem/failure reporting system by consolidating the current multiple systems into a single, comprehensive system with closed-loop features to ensure accountability and satisfactory dispositions.

(6) initiate education and training courses in availability/reliability maintainability engineering so that there is a more consistent and uniform approach to these disciplines in the design engineering community.

Status: The NRC staff believes that the current industry maintenance program and technical specification surveillance requirements provide adequate assurance that safety systems will be available when required. There is an ongoing program within the Institute of Electrical and Electronics Engineers (IEEE) to provide enhanced maintenance guidelines for many types of components. In addition, several vendors have submitted technical specification improvement programs to the NRC. This issue did not raise any new safety concerns.

- C&I Component and System Qualification

Issue: The subtask report recommends that GE's standards and qualification engineering department be given additional manpower and the responsibility for reviewing and approving the qualification of all systems and components for which C&I has responsibility.

Status: The NRC has stringent component and system qualification standards. This issue did not raise any new safety concerns.

- Systems Responsibility

Issues: The report recommends that GE

(1) focus the responsibility and authority for total BWR system design specification and control as the full-time responsibility of a senior technical manager and a small group of highly qualified system engineers.

(2) establish the required management and operational policies and procedures needed to ensure that this group receives the required support from GE's design, manufacturing, marketing, and projects organizations.

Status: This issue did not raise any new safety concerns.

CONCLUSIONS

As a result of its review, the staff concludes that, with the possible exception of the plant auxiliary power systems issue, no new safety issues are addressed in this subtask report. The issues addressed either involve (1) concerns that have been resolved elsewhere or (2) concerns that do not involve design methods, performance, quality, and availability for any safety aspects of BWR safety systems.

5.4 Subtask D: Report on Mechanical Systems and Equipment

INTRODUCTION

The subtask report on mechanical systems and equipment deals primarily with the reliability of major mechanical components in the BWR-6 nuclear steam supply system and the impact on projected plant availability. Mark I, II, and III containment issues are also addressed. Flow-induced vibration problems occurring in reactors that had been operating at the time of the study are addressed, as are the corrective actions taken in response to identified problems with mechanical systems and equipment and the design qualification and adequacy of BWR-6 components that have no operating history in reactor plants.

This report includes an extensive review of nuclear plant performance in terms of availability at the time of the study and the expected impact of identified problems and design changes on BWR-6 availability.

SUMMARY OF ISSUES

The safety significance and current status of Reed Report issues relating to the containment (including main steam isolation valves), mechanical equipment failures due to flow-induced vibration, and problems with the TIP system are discussed in Sections 3 and 4 of this NRC evaluation report. A discussion of other issues in this subtask report that are of potential safety significance follows.

- Crosby Safety Relief Valves (SRVs)

  Issue: Crosby direct spring-loaded SRVs were to be used on BWR-5 and -6 systems in place of the Target Rock and Dresser valves installed on plants that were operating at the time of the 1975 study. It was expected that the Crosby valves would be more reliable because they do not employ a pilot valve system that had caused actuation problems with the other valves.

  Safety Significance: The SRVs are required to protect the integrity of the reactor coolant pressure boundary and to limit the severity of overpressure transients. The primary operational concerns relate to actuation setpoint accuracy and reseating without leakage. If SRV maintenance is required between refueling outages, it contributes to unavailability of the reactor.

  Status: Testing and limited operational experience have not revealed any significant operational reliability problems with the Crosby valves for BWR service.

- Flow Control Valve (FCV)

  Issue: BWR-5 and -6 systems use FCVs in conjunction with a constant speed pump to control recirculation flow. The 20- and 24-inch valves required for this application had not been tested, raising questions about the durability and reliability of the valves.

Safety Significance: Major operational problems could result from FCV failures, resulting in challenges to thermal limits.

Status: These valves are now performing satisfactorily in operating BWRs.

CONCLUSION

The NRC staff has reviewed the mechanical systems and equipment subtask report and finds no new issues with potential safety significance that should be addressed.

5.5 Subtask E: Report on Materials, Processes, and Chemistry

INTRODUCTION

This subtask report addressed the materials, processes, and chemical technology necessary to achieve reliability and quality in BWR systems. The report assessed the effect of materials behavior, processing, and chemistry on plant reliability, safety, performance, and lifetime; evaluated the adequacy of material selection, procurement, application, and cost; and identified critical material and chemical areas for improvement or additional development.

SUMMARY OF ISSUES

The report addressed among other things the issue of stress corrosion cracking, which is discussed in detail in Sections 3 and 4 of this report. In addition, it addressed the areas that are discussed below. Some of these subjects are also discussed less completely in various places in Section 3.

- Radioactive Contamination

  Issue: Concern was expressed that radioactive contamination of piping and other components would build up to the point where radiation exposure to plant maintenance and operations personnel would become excessive. This would require additional manpower and increased costs. The report recommended that more effort should be expended on understanding the basic mechanisms of radioactivity transport and buildup, with the aim of making modifications to reduce the problem.

  Safety Significance: This issue is related to ALARA, and is a general, industry-wide problem. Although it is not a reactor safety issue, a great deal of effort has been expended on it. It should be noted that GE has developed a procedure designed to reduce buildup of radioactive contamination in piping and surfaces containing radioactive contamination. There also have been other major industry initiatives in developing and using decontamination processes, with generally good results.

  Status: This issue did not raise any new safety conerns.

- Reactor Pressure Vessel (RPV) -- Probability of Failure

  Issue: The report estimated the probability of a sudden disruptive failure of the RPV to be less than 1×10^{-6} per reactor year. This estimate applied to all presently designed BWRs.

Status: This estimate is in accord with studies done by the staff and the Advisory Committee on Reactor Safeguards (ACRS), as delineated in the WASH-1318 and WASH-1285 reports issued by the Atomic Energy Commission. Thus, this issue did not raise any new safety concerns.

- Reactor Pressure Vessel -- LOCA Integrity

  Issue: A detailed analysis of RPV integrity in BWRs under LOCA condition was last made in 1968; it showed that RPV integrity would be maintained. Much more recent reviews by the NRC and ACRS have reached similar conclusions.

  Status: A single LOCA would produce a thermal shock event, but the lack of repressurization in a BWR would preclude loadings that could cause failure. The issue of pressurized thermal shock in BWRs was fully addressed in the NRC staff response to interrogatories during the Atomic Safety Licensing Board hearings on the Limerick plant in 1983. Thus, this issue did not raise any new safety concerns.

- Reactor Pressure Vessel -- ATWS Pressures

  Issue: Calculations of peak pressures under postulated anticipated transient without scram (ATWS) conditions have been made within the past year for various BWRs. Peak pressure in the 1600 to 1650 psig range have been calculated for certain BWR-3 plants and considerably lower values for other BWRs. These pressures are well within the capacity of the vessel.

  Status: Recent studies done at Brookhaven have indicated that the maximum pressure expected during an ATWS event in a BWR is on the order of 1300 to 1350 psig, even less than GE assumed in the Reed Report. Thus this issue did not raise any new safety concerns. (Additional discussion on this concern is provided in Section 3.)

- Reactor Pressure Vessel -- Fatigue Cracking

  Issue: GE's studies provide strong support for the idea that fatigue crack growth in vessel steel under BWR environment conditions does not have an adverse impact on RPV integrity. Other GE studies indicate that stress corrosion cracking would not occur in RPV steels in BWR water within specifications.

  Status: Fatigue cracking caused by anticipated transients, as analyzed under ASME Code rules, is very unlikely, even with the known deleterious effect of BWR coolant on fatigue strength. Recent studies also provide assurance that when RPV steel is properly heat treated and stress relieved, it is not subject to stress corrosion cracking at stress levels found in reactor vessels. The stringent controls on welding and post-weld heat treatments imposed during the manufacture of reactor pressure vessels provide assurance that the material will be in a resistant condition, and high residual stresses will not be present. Thus, this issue did not raise any new safety concerns.

- **Reactor Pressure Vessel -- Nozzle Cracking**

Issue: Cracks had been observed in the cladding around feedwater nozzles at Millstone and Dresden 2, but the cracks were small enough to be readily removed. Ultrasonic indications of possible cracks at Pilgrim were being monitored on a continuing basis. In the BWR-6, the cladding was eliminated around all nozzles.

Status: In 1975, cracks were found in feedwater nozzles of several more BWRs, and a formal inspection and repair program was initiated. GE issued Service Information Letter No. 207, addressing feedwater nozzle cracking, on November 19, 1976. All cracking events and repair operations were reviewed and approved by the NRC. The NRC initiated Generic Technical Activity A-10 to address this issue. In July 1977, the NRC published NUREG 0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking," which described the problem, probable cause, and recommended actions. The cracking in both the feedwater nozzles and the control rod drive return line nozzles was attributed to thermal cycling. Thermal cycling of the feedwater nozzles was caused by an ineffective thermal sleeve. GE performed extensive testing and analysis, which resulted in recommended changes in design of the spargers and thermal sleeve. This was documented in GE's report NEDE 21821-A, issued in February 1980. The NRC resolution of this issue was documented in NUREG-0619, which was issued in November 1980, and was implemented by NRC's Generic Letter 81-11. This problem is considered to be resolved.

- **Reactor Pressure Vessel -- Inspection Access**

Issue: The BWR-6 was designed to accommodate currently specified and reasonably anticipated future RPV inspection requirements. However, inspection of RPVs in older plants, if required, can be performed to only a limited extent with currently available equipment and methods.

Status: While access to the RPV is provided for examination equipment in the BWR-6, the equipment itself had not been fully developed at the time this subtask report was written. Further, the ASME Code-specified inspections of ligament areas between control rod penetrations in the bottom head were not then possible in any BWR. Where such inspections are not practical, NRC may grant relief from the Code requirements.

Regarding the inspectability of the shell portion of the reactor vessel, including the radiation-affected belt line region, some BWR-5s provided access for inspection. Preservice examinations of this area have been performed at plants built fairly recently; therefore the equipment for examination from the outside has proven to be practical.

For older BWRs, the NRC has granted relief from examination of the major shell welds, because the biological shield is so close to the vessel that no examination equipment can fit in the insulated area. BWR vessels cannot readily be inspected from the inside (as PWRs are) because such internal structures as jet pumps are in the way, and the internals are not designed to be completely removed.

GE has a program with an overseas utility to develop equipment and methods to remotely inspect a significant portion of beltline welds by ultrasonics from the inside of the vessel. The NRC staff expects that such methods will soon be developed for general use. However, the staff does not believe that the acoustic emission inspection techniques mentioned in the subtask report have been sufficiently developed to be considered a realistic and practical approach. Nonetheless, this issue did not raise any new safety concerns.

- Reactor Pressure Vessel -- Embrittlement

Issue: 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 may be desirable at a later date to ensure that these plants can meet hydrostatic test requirements.

Status: Dresden 1 and Humboldt Bay are of no further concern because they have been decommissioned; Big Rock Point does show a considerable radiation-induced increase in RT_{NDT}. Nonetheless, the NRC staff has had no indication that Big Rock Point has any difficulty in heating up to the required temperature for leak and hydrostatic tests. This is partially because the vessel was designed to Section I of the ASME Code, so stress levels are very low. Other later BWRs are starting to show the effects of irradiation of the vessel on testing temperatures. Some licensees have considered the use of external heat sources to help achieve the required temperatures. However, the subtask report is correct in stating that irradiation of the vessel will not limit operation; thus this issue did not raise any new safety concerns.

- Materials Information System and Control

Issue: The subtask report discussed the need for GE to establish a stronger materials engineering organization with better laboratory facilities.

Status: This issue did not raise any new safety concerns.

- Level of Materials Effort

Issue: The issues discussed above addressed 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 paints. While active work is in progress in most of the areas and no significant deficiencies have been identified, the subtask report indicated that GE should expend additional effort to meet the high availability/capability goals on which its strategy is based.

Status: Several of these items have been covered in detail elsewhere (e.g., control rod materials and failure mechanisms, corrosion, Zircaloy channel materials, quality of vendor-supplied components, and radiation damage studies). In regard to the development of improved gasket, seal, and packing materials, although fewer and smaller leaks would enhance plant availability, this is not considered to be a safety issue; leakage limits already are imposed by a plant's technical specifications. In sum, none of these issues discussed in the subtask report raised any new safety concerns.

CONCLUSIONS

On the basis of its review, the NRC staff did not find that this subtask report raised any new issues with potential safety significance.

5.6 Subtask F: Report on Production, Procurement, and Construction

INTRODUCTION

This subtask report addresses critical components manufactured by GE, components procured from outside vendors, and the field erection of the nuclear steam supply system.

SUMMARY OF ISSUES

This subtask report on production, procurement, and construction identified concerns regarding fuel rods and vendor-supplied components. A discussion of these issues follows.

- Fuel Rods

  Issue: The report identified the following concerns regarding fuel rods:

  (1) GE should manufacture one standard fuel rod and one standard fuel pellet and compensate for needed variations by using different enrichments and rod arrangements. A second rod size may be needed to reduce fuel failure (increased wall thickness and reduced pellet diameter) at the highly stressed corner position.

  (2) In light of technical problems with fuel rod leaks, GE should review its decision to reduce the BWR-6 fuel pellet diameter by 0.006 inch and reduce the fuel rod wall thickness by 0.002 inch.

  (3) GE should improve the quality of the zirconium tubing it produces for fuel rods. Although the tubing is acceptable, it is of lower quality than that produced by Sandvik. Areas to be improved include roundness (it is not consistently round), surface flaws, and inspection equipment.

  Status: Appendix K to 10 CFR 50 contains the NRC requirements for fuel rod behavior during a loss-of-coolant accident. A plant's technical specifications establish the limits on the release of fission products from fuel rods as a result of normal operations and transients. These limits translate into the acceptablility of fuel rod design as corroborated by detailed analysis and testing. This issue does not raise any new safety concerns.

- Vendor-Supplied Components

  Issue: The report identified the following concerns regarding vendor-supplied components:

  (1) Vendor-supplied components are a cause of plant outages. Specific areas that must be improved include the qualification plans and commitments of qualification facilities, management commitment for establishing an integrated reliability program, valve testing, and reliability analysis in the design process. In addition, the report suggests eliminating vendors who do not provide adequate engineering support and performing studies of sufficient depth to support the quality needed for the nuclear industry.

  (2) There was a high probability that a qualified flow control valve for the recirculation system would not be available for a 1977 startup of BWR-5 plants.

  (3) GE should consider manufacturing some components supplied by vendors.

  (4) For the PGEE/NUCLENET System, GE should eliminate onsite changes by completing fabrication of the electrical and control system in the factory rather than on the site.

  Status: Appendix B to 10 CFR 50 addresses the QA criteria for the design and manufacture of safety-related components. It also provides the basic requirements for improved reliability of performance by implementation of the criteria on design control and corrective action.

  In addition, the NRC staff conducts an extensive inspection program which reviews the utility's activities and those of its principal contractors and vendors to determine conformance with NRC requirements and regulations, including those cited above.

  This issue did not raise any new safety concerns.

CONCLUSIONS

On the basis of its review, the NRC staff finds that this subtask report did not raise any new safety concerns.

5.7 Subtask G: Report on Quality Control System Overview

INTRODUCTION

This subtask report addresses the adequacy of the quality control system utilized by GE for the design, manufacture, and operation of nuclear steam supply systems for BWRs. It compares this system with the quality control systems adopted by five other GE organizational components.

SUMMARY OF ISSUES

The subtask report recommended that GE establish a "reliability" organization to analyze failure and repair data, and it discussed a need to establish plant availability goals in terms of design-significant parameters. It also stated that the resolution of major problems experience on already-constructed plants indicated a need for improved designs in equipment, materials, processes, and system control. The report included a listing of QA audit findings that showed that calibration practices were not formally documented or controlled, design reviews and documentation were not in conformance with established requirements, hardware documentation was sometimes not clear, engineers were not familiar with manuals, and, in some instances basic to ensuring design integrity, approved engineering practices and procedures had not been followed.

All of these issues are covered by existing NRC requirements and regulations. Specifically, these requirements and regulations include Appendix B to 10 CFR 50, which delineates the QA criteria for the design, construction, and operation of nuclear power plants; 10 CFR 21, which requires the immediate reporting of manufacturing defects; 10 CFR 50.55(e), which requires the reporting of deficiencies arising during construction of a nuclear power plant; and 10 CFR 50.72, which requires the reporting of certain significant events that occur during the operation of a nuclear power plant. In addition, the NRC staff conducts an extensive inspection program that reviews a utility's activities and those of its contractors and vendors on a sampling basis to determine conformance with NRC requirements and regulations, including those listed above.

It should be noted that the Institute for Nuclear Power Operations (INPO) maintains a system for collecting and analyzing failure and repair data. Access to this information is available to utilities with nuclear power plants for use in developing availability goals and improved maintenance programs.

CONCLUSIONS

On the basis of its review, the NRC staff finds that this subtask report did not raise any new safety concerns.

5.8 Subtask H: Report on Management/Information Systems

INTRODUCTION

This subtask report addresses the adequacy of management systems and their implementation to integrate and control BWR operations in the areas of design review, construction management, startup procedures, project management, and feedback of operating plant information.

SUMMARY OF ISSUES

A discussion of the findings of this study follows.

- Design Review

  Issue: Procedures for overall BWR systems design reviews should be improved.

Status: 10 CFR 50, Appendix B, gives the NRC QA criteria for design, construction, and operation of nuclear power plants. Specific requirements include design control to ensure that designs are verified and checked and that design reviews are performed. This issue was addressed in Section 3.16 and did not raise any new safety concerns.

- Calculational Models

  Issue: Additional ways are needed to obtain experimental data to verify calculational models. In addition, calculational models should be reviewed more thoroughly to ensure consistency of predictions.

  Status: This issue was addressed in Sections 3.6 and 3.17, and no new safety issues are raised here.

- Reliability Improvement

  Issue: A positive, high-visibility reliability improvement program is needed to increase plant availability.

  Status: This issue is not directly related to plant safety. However, in related areas, NRC regulations require the following: 10 CFR 21 requires the immediate reporting of manufacturing defects; 10 CFR 50.55(e) requires the reporting of deficiencies arising during plant construction; and 10 CFR 50.72 requires the reporting of certain significant events that occur during plant operation. Thus, all involved safety issues are covered by NRC regulations, and this issue did not raise any new safety concerns.

In addition, the study cited "12 unresolved 238 GESSAR items" that had been mentioned in a then-recent (circa 1975) letter from the NRC Advisory Committee on Reactor Safeguards. However, no details of this mention were given. From the context of the report, the concern is a management and information transfer problem, and so has no apparent safety significance.

CONCLUSIONS

On the basis of its review of this subtask report, the NRC staff concludes that NRC requirements and regulations adequately address the safety issues mentioned, and finds that this report did not raise any new safety concerns.

5.9 Subtask I: Report on Regulatory Considerations

INTRODUCTION

This subtask report evaluated the impact of regulatory policies on the cost of BWR power plants, including loss of availability and capacity, and it addressed ways of reducing this impact. The report concluded that backfit requirements had added up to 5% in direct equipment costs and probably more in regulation-induced delays. The report estimated that about 15% of GE's engineering time was expended on licensing matters. In addition, the report attributed a loss of 2% to 5% in annual electrical generating capability, as well as increased plant personnel requirements, to the regulatory process.

The study concluded that GE had contributed to the regulatory costs by failing to adequately develop some of the programs required by NRC to validate assumptions made in the preliminary design. This resulted in late identification of design problems, thus requiring changes to installations already in place.

As part of the recommendations for reducing the regulatory impact, the study listed potential new regulatory requirements likely to impact BWR-6 plants. It also listed possible long-term regulatory requirements. The study recommended ways that GE could minimize the impacts of these requirements if and when they come into being.

SUMMARY OF ISSUES

A discussion of the issues raised in this report follows.

- Period of Safety of Unattended Reactor

  Issue: The study recommended that the GE product safety standards be modified to ensure that a reactor will respond automatically to a reactor upset or accident to maintain core cooling for at least 30 minutes without operator intervention. The existing standard permitted credit for operator intervention in 10 minutes.

  Safety Significance: The time available for operator response relates to the probability that the required intervention to mitigate the consequences of the event will be correctly accomplished. This is a human factors consideration.

  Status: The NRC staff has some flexibility in this area. For some actions (e.g., suppression pool cooling), the normal practice is to accept an assumption of operator action within 10 minutes if it is justified on a plant-specific basis. The NRC Standard Review Plan (NUREG-0800) permits assumed actions within 20 minutes for emergency core cooling system long-term cooling, and within 15 minutes for response to boron dilution events. For anticipated transients without scram, credit for operator intervention within 2 minutes is permitted. The NRC staff has reviewed the BWR-6 to ensure that it conforms to these criteria.

- Means To Identify and Inspect Failed Fuel

  Issue: The study concluded that the main steam line (MSL) radiation monitor, which was used for prompt detection and shutdown of the reactor for a sudden and major fuel failure, may not be sufficiently sensitive for this purpose because of gross gamma radiation (mainly N^{16}) associated with the steam. It also concluded that NRC might require an improved technique for locating failed fuel, possibly more sensitive than the "sipping" technique that requires opening of the reactor. The study recommended that GE develop an improved failed fuel sensor, but noted that an NRC requirement for location of failed fuel without opening the reactor was unlikely.

  Safety Significance: The MSL high radiation scram and isolation signals serve to limit radioactivity release in the event of fuel failures. Safety analyses take credit for the isolation function in the analyses of the control rod drop accident. It is also pertinent to ALARA considerations. The procedures and efficiency for location and removal of failed fuel also are important ALARA considerations.

Status: NRC requirements have not changed. The BWR owners have indicated an intent to propose elimination of the MSL radiation isolation and scram functions. This would be justified by analyses that take credit for gas holdup in augmented offgas systems. Sipping remains the most effective means of locating failed fuel, but techniques using this method have improved since 1975. Also, fuel failure is much less frequent than in 1975. This issue did not raise any new safety concerns.

- Functional Specifications for Power and Self-Operated Valves

Issue: The study concluded that NRC was likely to impose specific functional requirements for valves that had a history of frequent operating problems, such as safety relief valves and main steam line isolation valves. The study recommended that GE develop appropriate specifications.

Safety Significance: These valves are required to function as assumed in the safety analyses to limit the consequences of various transients and accidents.

Status: A TMI Action Plan requirement provides for acoustic monitors to detect leakage of safety-relief valves. Safety-relief and main steam isolation valve performance and surveillance requirements are normally controlled by technical specifications. These are based on functional requirements and safety analyses provided by the designer of the plant's nuclear steam supply system and the specific plant licensee.

- N-2 Safety Logic

Issue: The study postulated that NRC redundancy requirements for emergency cooling systems might be expanded to require three complete systems, each capable of cooling the core in the event of a LOCA. With N = number of available systems, this is defined as N-2 logic, which permits one system to be out of service for maintenance and testing and a second system to fail when needed without loss of the emergency cooling function. The report recommended that GE study the need for N-2 safety logic as is used in German and Swiss reactor systems.

Safety Significance: The degree of redundancy in the emergency core cooling systems is related to the system availability and probability of core melt.

Status: The staff has not identified a need for additional redundancy in this area, and this issue did not raise any new safety concern; however, N-2 logic is an approach being considered by the staff in its study of Unresolved Safety Issue IA-45, "Shutdown Decay Heat Removal Requirements."

- Removable Reactor Internals

Issue: The study considered the susceptibility of internal BWR components to radiation damage, flow-induced vibration, and other failure mechanisms that might require replacing reactor internals. The difficulty of replacing components that are welded in place was of concern because of the high level of induced radioactivity and consequent occupational exposure. The study anticipated that NRC would impose new requirements in this regard, and recommended that such a design be developed for later BWR-6 orders and for advanced designs.

Safety Significance: The safety concern of this issue is the need to ensure that component failures cannot result in unacceptable consequences and that appropriate surveillance procedures and monitoring instrumentation are in place to detect such failures before they degrade plant operating safety. Additionally, replacement of failed reactor internals components is a major ALARA concern.

Status: Many internal components -- such as feedwater spargers, jet pump holddown beams, etc. -- have degraded or failed in service and have been replaced. Occupational exposure for this type of work has been significant but occupational exposure to individuals is limited by regulations. This issue did not raise any new safety concerns.

- Core Catcher

Issue: The study noted that a core catcher was a major issue with the breeder reactor, and recommended that GE study this issue so it could respond to NRC if a new requirement were developed.

Safety Significance: Prevention of containment penetration by the molten core in the event of a severe accident is a major safety issue.

Status: Later studies have shown that containment melt-through by a molten core is less likely than previously assumed. The staff is continuing its studies of severe accidents. These studies include the feasibility and cost/benefit of passive devices such as curbs to contain a molten core. Thus this issue did not raise any new safety concerns.

CONCLUSIONS

On the basis of its review, the NRC staff finds that this subtask report did not raise any new safety concerns. Moreover, the report did not present any ideas concerning possible new regulatory requirements identified by GE that give cause for the NRC to re-examine its policy in these areas. Before imposing any new requirements, the NRC routinely considers the impact on power production in relation to the safety benefit to be gained.

5.10 Subtask J: Report on Scope and Standardization

INTRODUCTION

This subtask evaluated the GE NED scope of supply and standardization policy in terms of potential impact on overall nuclear plant availability/reliability and operation. The approach consisted of analysis of plant performance data existing at that time to determine the root causes of plant unavailability and the options available to improve the plants by providing a superior quality product and by extension of the boundaries of NED scope of supply and services. Findings of the study were that about 46% of unavailability was due to the reactor building, 19% was due to refueling and other outages, and only 35% was due to balance-of-plant (BOP) issues. With respect to power limitations and availability in 1974 only, 14.3% of total capacity was lost due to forced outages and 16.1% was due to scheduled outages. The reactor scope was identified as the highest source of unavailability; contributions by the BOP area were small. Power derating as an initial response to alleviate potential equipment failures from new identified problems and to reduce fuel failures from PCI accounted for much of the lost capacity.

The study concluded that expansion of the BWR customer service area with expanded outage service, better tools, improved operation, and special programs for identified problem areas offered the best potential for improving availability. The study concluded also that the BWR availability goal based on previously established fossil availability was unrealistic because of identified technical problems and other problems not yet identified.

The standardization effort was expected to be effective only with the BWR-6.

SUMMARY OF ISSUES

The staff examined those items listed in the report as sources of unavailability and power limitations that contributed to unavailability via power derating and/or forced outages to determine their safety significance. These issues identified in the subtask report were

- PCI operating management recommendations
- leaky relief valves
- leaky MSIV valves
- MAPLHGR limitations
- sensitized stainless steel cracks (major)
- reactivity shortfall
- feedwater sparger problems
- offgas
- channels
- operations management

All of these issues are addressed elsewhere in this staff report and, with the exception of "operations management," have been substantially resolved. NRC is continuing to review and evaluate operations management by individual licenses.

CONCLUSIONS

On the basis of its review, the staff finds that this subtask report did not raise any new safety issues.

NRC FORM 335 (7-83)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG-1285

2. Leave blank

3. TITLE AND SUBTITLE
NRC Staff Evaluation of the General Electric Company Nuclear Reactor Study ("Reed Report")

4. RECIPIENT'S ACCESSION NUMBER

5. DATE REPORT COMPLETED
MONTH July YEAR 1987

6. AUTHOR(S)
Martin J. Virgilio
Roby Bevan

7. DATE REPORT ISSUED
MONTH July YEAR 1987

8. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

9. PROJECT/TASK/WORK UNIT NUMBER

10. FIN NUMBER

11. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
same as above

12a. TYPE OF REPORT

12b. PERIOD COVERED (Inclusive dates)

13. SUPPLEMENTARY NOTES

14. ABSTRACT (200 words or less)

ABSTRACT

In 1975, the General Electric Company (GE) published a Nuclear Reactor Study, also referred to as "the Reed Report," an internal product-improvement study. GE considered the document "proprietary" and thus, under the regulations of the Nuclear Regulatory Commission (NRC), exempt from mandatory public disclosure. Nonetheless, members of the NRC staff reviewed the document in 1976 and determined that it did not raise any significant new safety issues. The staff also reached the same conclusion in subsequent reviews.

However, in response to recent inquiries about the report, the staff re-evaluated the Reed Report from a 1987 perspective. This re-evaluation, documented in this staff report, concluded that (1) there are no issues raised in the Reed Report that support a need to curtail the operation of any GE boiling water reactor (BWR); (2) there are no new safety issues raised in the Reed Report of which the staff was unaware; and (3) although certain issues addressed by the Reed Report are still being studied by the NRC and the industry, there is no basis for suspending licensing and operation of GE BWR plants while these issues are being resolved.

15a. KEY WORDS AND DOCUMENT ANALYSIS
"Reed Report"
General Electric

15b. DESCRIPTORS

16. AVAILABILITY STATEMENT
Unlimited

17. SECURITY CLASSIFICATION (This report)
Unclassified

18. NUMBER OF PAGES

19. SECURITY CLASSIFICATION (This page)

20. PRICE
$