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)

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 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 8.2×10^{-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 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 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).

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 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.

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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.

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 during severe accidents. The results of these programs will be evaluated in accordance with the Commission's regulations tc 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.

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

Commission for its decision whether to release the Report [6/]. Subsequently, in July 1986, the Commission voted to withhold the Reed Report from public disclosure. GE subsequently released the Reed Report in July 1987 in a two-volume document titled "12 Years Later... An Update Report on the Nuclear Reactor Safety Study." The updated report describes how earlier NRC reviews of 1976 and 1978 confirmed how all safety issues mentioned in the Reed Report had been disclosed to the NRC previously. It also describes how the study was performed early in the BWR-6 (Mark III containment) design cycle and how the recommendations from that report were implemented before BWR-6 Mark III plants went into operation.

Nonetheless, as public interest in the "newly discovered" Reed Report heightened, and notwithstanding their earlier reviews of the document, 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.

The purpose of this review was to place these issues in a 1987 perspective to ensure that the NRC staff truly had been aware of all safety issues discussed within the report and that the issues were either resolved or programs were under way to address those issues not yet resolved.

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.

6/ General Electric Co. v. U.S. Nuclear Regulatory Commission, 750 F.2d 1394 (7th Cir. 1984).

(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.

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.