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February 28, 1990

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SECY	PDI-1 Rdg*	GPA/CA

The Honorable Frank Horton
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Horton:

I am responding to your letter of February 6, 1990, in which you requested that we look into some of the concerns expressed by the group "Retire Nine Mile One" regarding the Nine Mile Point Nuclear Station, Unit 1 (NMP-1).

The "Retire Nine Mile One" group raised issues concerning the use of a Mark I containment design, the corrosion of certain materials comprising part of the containment (the torus), whether the refueling program should be stopped and the holding of a public hearing on NMP-1. The NRC staff has prepared the enclosed response on these issues.

In summary, I believe that these issues are being addressed appropriately by the NRC staff and the licensee for the NMP-1 facility. Consequently, I know of no reason at this time why the licensee's activities in preparation for restart should not proceed.

The NRC staff will continue to monitor the licensee's progress in this area, and we will not make a decision on restart until the licensee's preparatory activities are essentially complete. Preparations are expected to be complete later this spring. Before NMP-1 is restarted, the licensee and the NRC staff will brief the Commission on the status of NMP-1 in a meeting that will be part of the public record.

I trust that this information will be useful to you in responding to the concerns of the "Retire Nine Mile One" group.

Sincerely,

Original Signed By:
James M. Taylor

James M. Taylor
Executive Director
for Operations

Enclosure:
1. Response of NRC staff
2. NRC Generic Letter 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities 10 CFR 50.54(f), "November 23, 1988
3. NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," September 1, 1989
4. Response to Public Meeting Comments

PDI-1	PDI-1	Tech Ed	PDI-1	PDI-1:AD	DRP:D	NRR:D
CVogan	RMartin:rsc	RCapra for	RACapra	BBoger	SVarga	JPartlow
2/23/90	2/23/90	2/23/90	2/23/90	2/23/90	2/23/90	2/23/90
		Comments incorporated				

NRR:D	EDO	
TMurley	JTaylor	
2/26/90	3/1/90	

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The "Retire Nine Mile One" group expressed concern in their letter to you dated January 8, 1989, that are very similar to the concerns this group communicated to Congressman James T. Walsh. Accordingly, we are enclosing a copy of our response to Congressman Walsh on this matter.

As we indicated to Congressman Walsh, we believe that the identified concerns are being addressed appropriately and that there is no reason at this time why the licensee's activities in preparation for restart should not proceed.

The NRC staff will continue to monitor the licensee's progress in this area, and we will not make a decision on restart until the licensee's preparatory activities are essentially complete. Preparations are expected to be complete later this spring. Before NMP-1 is restarted, the licensee and the NRC staff will brief the Commission on the status of NMP-1 in a meeting that will be part of the public record.

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Enclosure:
Response to Congressman Walsh

OFC	PDI-1	PDI-1	Tech Ed	PDI-1	PDI-1	DRP:DIR
NAME	CVogan	RMartin		RACapra	BBoger	SVarga
DATE	2/16/90	2/15/90	2/15/90		2/16/90	2/16/90

OFC	NRR:D	ADP:NRR	EDO			
NAME	JPartlow	TMurley	JTaylor			
DATE	2/16	2/21				

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OFC	NRR:D	ADP:NRR	EDO				
NAME	JPartlow	TMurley	JTaylor				
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ENCLOSURE 1

RESPONSE OF NRC STAFF TO TOPICS IN

"RETIRE NINE MILE ONE" LETTER DATED

JANUARY 8, 1989

The letter of the "Retire Nine Mile One" group expresses concern about the capability of the Mark I containment design at Nine Mile Point Unit 1 (NMP-1) to withstand the challenges of severe accidents. The issue of performance of the containment during severe accidents, in addition to other technical issues, has been a subject of the NRC's integration plan for closure of severe accident issues. Severe accidents are considered to be those accident scenarios that are more extensive, or severe, in their impact than the spectrum of design-basis accidents that have been analyzed, pursuant to specific regulatory requirements, as the basis for licensing decisions.

The NRC has had severe accidents under consideration for many years. Severe accident evaluations and research progressed to the point that the Commission issued a Severe Accident Policy Statement (50 FR 32138) on August 8, 1985, which concluded that existing plants posed no undue risk to the public. However, based on NRC and industry experience with plant-specific probabilistic risk assessments, the NRC recognized that systematic examinations would be beneficial in identifying plant-specific vulnerabilities to severe accidents. Accordingly, a severe accident closure implementation program was developed. The program includes several major elements that are directly pertinent to containment performance. One of these elements, the Individual Plant Examination for Severe Accident Vulnerabilities (IPE), will utilize probabilistic risk assessment or other systematic examination methodology to develop for each plant a better understanding of severe accident behavior and the specific accident sequences, including their probabilities. Then, if necessary, modifications would be made to help prevent or to mitigate severe accidents. Licensees for nuclear power plants have been requested to perform an IPE for their plant by an NRC generic letter dated November 23, 1988, and to submit the results of the IPE to NRC by August 1992. This generic letter is provided for information as Enclosure 2.

The second major element of the severe accident closure implementation program to be discussed here is the containment performance improvement program (CPI). This program is related to the IPE effort and is considered complementary to it as the CPI program is primarily focussed on the potential generic vulnerabilities of specific containment classes, whereas the IPE effort is focussed on plant-unique vulnerabilities. The CPI program includes both pressurized and boiling water reactor containment types. Based on the results of this program for the boiling water reactor (BWR) Mark I containment class of plants, 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 potential safety improvement candidates identified up to that point, with one exception, should be evaluated as part of the IPE Program.

With respect to the one exception, the Commission directed that improvements dealing with a hardened wetwell venting capability be approved, under the provisions of 10 CFR 50.59, for those licensees electing, on their own initiative, to implement them and that backfit analyses on the efficacy of requiring such improvements be undertaken for the remaining licensees. This issue of the hardened wetwell venting capability has been communicated to licensees by a generic letter dated September 1, 1989, which is provided as Enclosure 3 for information. The NRC staff understands from the licensee's letter of October 30, 1989 that the licensee for NMP-1 plans to install such a venting capability; however, the schedule for completing this action is still under discussion with the licensee.

The purpose of the preceding discussion is to highlight some of the present regulatory program efforts directed at assessing and improving, where needed, the performance capabilities of the Mark I containment. The statement regarding the high probability of Mark I containment failure referred to in the "Retire Nine Mile One" letter derives from the results of the WASH-1400 probabilistic risk assessment published in 1975. This assessment was based on a plant design and operating and emergency procedures as they stood at that time. As indicated by the discussion above about currently ongoing programs, much is being done to assess and, where needed, to improve the performance capabilities of the Mark I containment.

In summary, on this issue it is important to recognize that although the need has been identified to develop a severe accident closure program, which includes specific actions for the Mark I containment design, the Commission reached the conclusion in its policy statement on severe accidents in 1985 that, based on available information, the existing plants pose no undue risk to the public health and safety.

The "Retire Nine Mile One" group also asked that refueling of the NMP-1 be stopped. The licensee completed fuel loading NMP-1 on January 18, 1990, as part of its scheduled restart activities associated with a refueling outage that began in January 1988. On July 24, 1988, during this outage, the NRC Region I Administrator issued a Confirmatory Action Letter (CAL) to the licensee that confirmed that certain actions would be taken and that the NRC Regional Administrator's approval would be obtained before further operation of the unit. The NRC staff's concerns at that time evolved from consideration of the effectiveness that the licensee was achieving in responding to a number of issues.

One of these issues concerned the corrosion of the torus portion of the containment. This corrosion has been monitored for several years by the licensee. The licensee has prepared an assessment of the design margin remaining in the torus as a basis for supporting operation in the forthcoming fuel cycle. The licensee is also considering several potential corrective action approaches that it can take to resolve the problem on a long-term basis. The NRC staff has conducted independent measurements to confirm the licensee's measurements and currently has the licensee's response to this issue under review. The staff will require that an acceptable margin of safety for the torus structures be established before allowing operation in the forthcoming fuel cycle and in later fuel cycles.

The CAL confirmed the licensee's commitment to identify the root causes of such problems, to submit a restart action plan that addressed the root causes, and to submit a report about the readiness of the unit for restart. These actions have been taken by the licensee and have been reviewed extensively by the NRC staff. The NRC staff has completed its review and has approved the Restart Action Plan for Nine Mile Point Unit 1. The NRC staff concluded that the licensee's plan contains the essential elements to effect overall performance improvements. The staff has also concluded that different aspects of the plan were being implemented with varying degrees of effectiveness. These findings have been responded to by the licensee, and the NRC staff is monitoring the licensee's performance in this interim period as the licensee completes its planned activities to prepare the plant for restart.

The NRC staff is aware of the status of the issues that pertain to a decision on restart. The NRC staff has not identified any need to order the suspension of activities in preparation for restart of the plant, and the "Retire Nine Mile One" group provides no specific bases for its request in this regard. Accordingly, the staff does not plan to order suspension of the licensee's startup activities.

In response to the matter of a public hearing, it may be useful to consider that the NRC staff has previously provided an opportunity for public comment at a public meeting on the licensee's Restart Action Plan. These comments were considered and were responded to as set forth in Enclosure 4.

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

ENCLOSURE 2

November 23, 1988

To All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities

SUBJECT: INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES - 10 CFR §50.54(f) (Generic Letter No. 88-20)

1. SUMMARY

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985 (50 FR 32138), the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, the Commission recognizes, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, each existing plant should perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission.

The general purpose of this examination, defined as an Individual Plant Examination (IPE), is for each utility (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant, (3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents. It is expected that the achievement of these goals will help verify that at U.S. nuclear power plants severe core damage and large radioactive release probabilities are consistent with the Commission's Safety Goal Policy Statement. Besides the Individual Plant Examinations, closure of severe accident concerns will involve future NRC and industry efforts in the areas of accident management and generic containment performance improvements. Additional discussion is provided in SECY-88-147 on the interrelationships among these three areas and the role they play in closure of severe accident issues for operating plants. The portion of that document relevant to closure is provided as Attachment 1. Attachment 2 contains a list of references of the IDCOR program technical reports and also some related NRC and NRC contractor reports.

Therefore, consistent with the stated position of the Commission and pursuant to 10 CFR §50.54(f), you are requested to perform an Individual Plant Examination of your plant(s) for severe accident vulnerabilities and submit the results to the NRC.

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2. Examination Process

The quality and comprehensiveness of the results derived from an IPE will depend on the vigor with which the utility applies the method of examination and on the utility's commitment to the intent of the IPE. Furthermore, the maximum benefit from the IPE would be realized if the licensee's staff were involved in all aspects of the examination to the degree that the knowledge gained from the examination becomes an integral part of plant procedures and training programs. Therefore, we request each licensee to use its staff to the maximum extent possible in conducting the IPE by:

1. Having utility engineers, who are familiar with the details of the design, controls, procedures, and system configurations, involved in the analysis as well as in the technical review, and

2. Formally including an independent in-house review to ensure the accuracy of the documentation packages and to validate both the IPE process and its results.

The NRC expects the utility's staff participating in the IPE to:

(1) Examine and understand the plant emergency procedures, design, operations, maintenance, and surveillance to identify potential severe accident sequences for the plant; (2) understand the quantification of the expected sequence frequencies; (3) determine the leading contributors to core damage and unusually poor containment performance, and determine and develop an understanding for their underlying causes; (4) identify any proposed plant improvements for the prevention and mitigation of severe accidents; (5) examine each of the proposed improvements, including design changes as well as changes in maintenance, operating and emergency procedures, surveillance, staffing, and training programs; and (6) identify which proposed improvements will be implemented and their schedule.

3. External Events (Treated Separately)

Licensees are requested to proceed with the examinations only for internally initiated events (including internal flooding) at the present time. Examination of externally initiated events (i.e., internal fires, high winds/tornadoes, transportation accidents, external floods, and earthquakes) will proceed separately and on a later schedule from that of internal events (1) to permit the identification of which external hazards need a systematic examination, (2) to permit development of simplified examination procedures, and (3) to integrate other ongoing Commission programs that deal with various aspects of external event evaluations, such as the Seismic Design Margins Program (SDMP), with the IPE(s) to ensure that there is no duplication of industry efforts. Utilities would be expected to examine and identify any plant-specific vulnerabilities to severe accidents due to externally initiated events. Therefore, while performing your IPE for internally initiated events, you should document and retain plant-specific data relevant to external events (e.g., data from plant walkdowns) such that they can be readily retrieved in a convenient form when needed for later external event analyses that may be required. If a licensee chooses to submit an external event examination at this time, the staff would review it on a case-by-case basis.

While current staff efforts are focused on identifying acceptable methods for examining external events, the staff encourages the industry to propose a methodology for examining external events that meets the intent of the severe accident policy; namely, that it is capable of identifying vulnerabilities to external hazards. We will work with NUMARC in developing acceptable methodologies for external hazard examinations.

4. Methods of Examination

The NRC has identified three approaches that satisfy the examination requested by this letter. The methods are:

1. A PRA, provided it is at least a Level I* and uses current methods and information, plus a containment performance analysis that follows the general guidance given in Appendix 1 to this generic letter. The staff will consider those PRAs that follow the PRA procedures described in NUREG/CR-2300, NUREG/CR-2815, or NUREG/CR-4550 to be adequate for performing the IPE, provided the assessment considers the most current severe accident phenomenological issues (as discussed in Appendix 1) and the licensee certifies that the PRA is based on the most current design.

2. The IDCOR system analysis method (front-end only), provided the enhancements identified in the NRC staff evaluation of the IDCOR method (to be issued shortly) are applied. Guidance for the back-end analysis is provided in Appendix 1 and additional guidance will be issued as described in Section 11 of this generic letter.

3. Other systematic examination methods, provided the method is described in the licensee response and is accepted by the NRC staff. For those methods with which the staff is not familiar, a staff review might be necessary to ensure that the methods are generally acceptable.

For the phase of the evaluation associated with core melting, release of molten core to the containment, and containment performance, the staff recognizes that for a few of the phenomena, notably associated with areas that affect containment performance, there is a wide range of views about their relative probability as well as their consequences. For these issues, additional research and evaluation will be needed to help reduce the wide range of uncertainties. Because of the concern over the ability of containments to perform well during some severe accidents, the staff is conducting a Containment Performance Improvements Program. This program complements the IPE program and is intended to focus on resolving generic containment challenges. Licensees are expected to correct vulnerabilities that may be identified by their IPE results but, because of the generic Containment Performance Improvements Program that complements the IPE, the

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*The PRA levels are defined as follows: Level I - determination of core-damage frequencies based on system and human-factor evaluations; Level II - determination of the physical and chemical phenomena that affect the performance of the containment and other mitigating features and the behavior and release of the fission products to the environment; and Level III - determination of the off-site transport, deposition, and health effects of fission product releases.

staff does not require industry to make any major modifications to their containments or other systems that can affect containment performance until the information associated with the containment performance generic issues has been developed by the staff. Hence, industry will not be placed in a position of having to implement improvements before all containment performance decisions have been made.

Appendix 1 provides the utility with guidance to proceed with the evaluation of containment performance to identify plant-specific factors important to containment performance. Following the Appendix 1 guidance will also enable utilities to understand and develop strategies to minimize the challenges and the consequences such severe accident phenomena may pose to the containment integrity and to recognize the role of mitigation systems while awaiting their generic resolution.

5. Resolution of Unresolved Safety/Generic Safety Issues (Relationship to USI A-45)

Because the resolution of several USI(s) and GSI(s) may require an examination of the individual plant, it is reasonable to use the current IPE process for that examination. For example, Unresolved Safety Issue (USI) A-45 entitled "Shutdown Decay Heat Removal Requirements" had as its objective the determination of whether the decay heat removal function at operating plants is adequate and if cost-beneficial improvements could be identified. We concluded that a generic resolution to the issue (e.g., a dedicated decay heat removal system for all plants) is not cost effective and that resolution could only be achieved on a plant-specific basis. To implement a plant-specific resolution would require each plant to do an examination of its decay heat removal system to identify vulnerabilities. In the IPE, each plant will do an examination of both its decay heat removal system and those systems used for the other safety functions for the purpose of identifying severe accident vulnerabilities. Therefore, we have concluded that the most efficient way to resolve A-45 is to subsume it in the IPE.

You should ensure that your IPE particularly identifies decay heat removal vulnerabilities. To achieve this assurance we have extracted insights gained from the six case studies performed for the USI A-45 program. These insights are discussed in Appendix 5 to this letter and should be considered as you conduct your IPE. In addition, if a utility (1) discovers a notable vulnerability during its IPE that is topically associated with any other USI or GSI and proposes measures to dispose of the specific safety issue or (2) concludes that no vulnerability exists at its plant that is topically associated with any USI or GSI, the staff will consider the USI or GSI resolved for a plant upon review and acceptance of the results of the IPE. Your IPE submittal should specifically identify which USIs or GSIs it is resolving.

6. PRA Benefits

The NRC recognizes that many licensees now possess plant-specific PRAs or similar analyses. Use of existing PRA analyses is encouraged in achieving the objectives of the IPE. In some cases, the licensee may have to confirm that the existing PRA analyses reflect the current state of the art regarding severe accidents.

In addition to being an acceptable method for conducting an IPE, there are a number of potential benefits in performing PRAs on those plants without one. Some examples of potential additional benefits are as follows:

Support for Licensing Actions - PRAs have been used to support arguments to justify technical specification changes, both routine and emergency. PRAs would also be useful in supporting other regulatory actions (e.g., design modifications).

License Renewals - PRAs could be a basis for utilities to establish a program to ensure that risk-significant components and systems are identified and maintained at an acceptable level of reliability during the license renewal period.

Risk Management - A PRA could be used to develop a risk management program that systematically uses the available information about risk at a nuclear power plant and identifies alternative combinations of design and operational modifications, ranks these alternatives according to the relative benefits of each, and selects an optimum from the alternatives.

Integrated Safety Assessment - The staff believes that by performing a PRA a licensee would have the benefit of having developed the technical basis for an integrated assessment. An integrated safety assessment would (1) provide integrated schedules for licensing, regulatory, and safety issues on a predictable basis, (2) evaluate licensing and generic issues on a plant-specific basis such that they are weighted against all other pending actions, (3) provide a licensee with the opportunity to demonstrate with its PRA that various issues that might be applied to other plants are not justified at that facility, (4) help improve outage planning, and (5) rank issue importance such that the most important are dealt with first. This prioritization of actions benefits the licensees and the NRC by providing a rational schedule for implementation of actions and provides a basis for the possible elimination of actions determined to have low safety significance for the individual plant.

7. Severe Accident Sequence Selection

In performing an IPE, it is necessary to screen the severe accident sequences for the potentially important ones and for reporting to the NRC. The screening criteria to determine the potentially important functional sequences* that lead to core damage or unusually poor containment performance and should be reported to the NRC with your IPE results are listed in Appendix 2. Appendix 4 describes

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*"Sequence" is used here to mean a set of faults, usually chronological, that result in the plant consequence of interest, i.e., either a damaged core or unusually poor containment performance. A functional sequence is a set of faulted functions that summarizes by function a set of systems faults which would result in the consequence of interest. Functional sequences are to be contrasted with systemic sequences. A systemic sequence is a set of faulted systems that summarizes by systems a set of component failures resulting in a damaged core or unusually poor containment performance.

the documentation needed for the accident sequence selection and the intended disposition of these sequences.

It is expected that during the course of the examination, the utility would carefully examine the results to determine if there are worthwhile prevention or mitigation measures that could be taken to reduce the core damage frequency or poor containment performance with the attendant radioactive release. The determination of potential benefits is plant specific and will depend on the frequency and consequence of the accident sequence leading to core damage and containment failure.

8. Use of IPE Results

a. Licensee

After each licensee conducts a systematic search for severe accident vulnerabilities in its plant(s) and determines whether potential improvements, both design and procedural, warrant implementation, it is expected that the licensee will move expeditiously to correct any identified vulnerabilities that it determines warrant correction. Information on changes initiated by the licensee should be provided consistent with the requirements of 10 CFR 50.59 and 10 CFR 50.90. Changes should also be reported in your IPE submittal (by reference to previous submittals under 10 CFR 50.59 or 10 CFR 50.90) that responds to this letter (see Appendix 4).

b. NRC

The NRC will evaluate licensee IPE submittals to obtain reasonable assurance that the licensee has adequately analyzed the plant design and operations to discover instances of particular vulnerability to core melt or unusually poor containment performance given a core melt accident. Further, the NRC will assess whether the conclusions the licensee draws from the IPE regarding changes to the plant systems, components, or accident management procedures are adequate. The consideration will include both quantitative measures and nonquantitative judgment. The NRC consideration may lead to one of the following assessments:

1. If NRC consideration of all pertinent and relevant factors indicates that the plant design or operation must be changed to meet NRC regulations, then appropriate functional enhancements will be required and expected to be implemented without regard to cost except as appropriate to select among alternatives.

2. If NRC consideration indicates that plant design or operation could be enhanced by substantial additional protection beyond NRC regulations, then appropriate functional enhancements will be recommended and supported with analysis demonstrating that the benefit of such enhancement is substantial and worth the cost to implement and maintain that enhancement, in accordance with 10 CFR 50.109.

3. If NRC consideration indicates that the plant design and operation meet NRC regulations, and that further safety improvements are not substantial or not cost effective, enhancements would not be suggested unless significant new safety information becomes available.

9. Accident Management

An important aspect of severe accident prevention and mitigation is the total organizational involvement. Operations personnel have key roles in the early recognition of conditions or events that might lead to core damage. The availability of procedures specifying corrective actions and the training of operators and emergency teams can have a major influence on the course of events in case of a severe accident.

Because the conclusions you will draw from the IPE for severe accident vulnerabilities (1) will depend on the credit taken for survivability of equipment in a severe accident environment, and (2) will either depend on operators taking beneficial actions during or prior to the onset of severe core damage or depend on the operators not taking specific actions that would have adverse effects, the results of your IPE will be an essential ingredient in developing a severe accident management program for your plant.

At this time you are not required to develop an accident management plan as an integrated part of your IPE. We are currently developing more specific guidance on this matter and are working closely with NUMARC to (1) define the scope and content of acceptable accident management programs, and (2) identify a plan of action that will ultimately result in incorporating any plant-specific actions deemed necessary, as a result of your IPE, into an overall severe accident management program. Nevertheless, in the course of conducting your IPE you may identify operator or other plant personnel actions that can substantially reduce the risk from severe accidents at your plant and that you believe should be immediately implemented in the form of emergency operating procedures or similar formal guidance. We encourage each licensee to not defer implementing such actions until a more structured and comprehensive accident management program is developed on a longer schedule, but rather to implement such actions immediately within the constraints of 10 CFR 50.59.

10. Documentation of Examination Results

The IPE should be documented in a traceable manner to provide the basis for the findings. This can be dealt with most efficiently by a two-tier approach. The first tier consists of the results of the examination, which will be reported to the NRC for review. The second tier is the documentation of the examination itself, which should be retained by the licensee for the duration of the license unless superseded. Appendix 4 contains the minimum information necessary for reporting and documentation.

11. Licensee Response

A document that provides additional licensee guidance for the performance of the IPE (both core damage and containment system performance) and describes the review and evaluation process that the NRC staff will use for assessing the submittals will be issued in draft form within the next few months.

Following the issuance of the draft document, workshops with utility representatives will be scheduled to discuss the IPE objectives and to answer questions that utilities might have on both the IPE generic letter and the guidance document.

Following the completion of the workshops, the NRC, as appropriate, will revise its guidance contained in the guidance documents to take into consideration comments received and will reissue them. Within 60 days of receipt of the final guidance documents, licensees are requested to submit their proposed programs for completing the IPEs. The proposal should:

1. Identify the method and approach selected for performing the IPE,

2. Describe the method to be used, if it has not been previously submitted for staff review (the description may be by reference), and

3. Identify the milestones and schedules for performing the IPE and submitting the results to the NRC.

Meetings at NRC Headquarters during the examinations will be scheduled as needed to discuss subjects raised by licensees and to provide necessary clarifications.

Licensees are expected to submit the IPE results within 3 years. The Commission encourages those plants that have not yet undergone any systematic examination for severe accidents to promptly initiate the examination.

Those utilities that choose to use an existing PRA or similar analysis on their plant should (1) certify that the PRA meets the intent of the generic letter, in particular with respect to utility staff involvement, (2) certify that it reflects the current plant design and operation, and (3) submit the results as soon as the analysis is completed but on a shorter schedule than 3 years. Utilities with plants that used the initial IDCOR system analysis in the IDCOR test applications are encouraged to submit their results on a shorter schedule than 3 years. This will ensure review and resolution of any items while the utility's examination team is easily accessible. In this regard, the staff also encourages licensees whose plants have been extensively analyzed under the NUREG-1150 program to submit their IPEs on an expedited basis. This will enable the staff to exercise its review and decision process for determining acceptability of the IPE, the adequacy of the licensee identification of plant-specific vulnerabilities, and the associated modifications using insights and experience from NUREG-1150. Finally, those licensees planning to perform a new Level II or Level III PRA may need more time. The NRC staff will consider requests for additional time for such an examination.

12. Regulatory Basis

This letter is issued pursuant to 10 CFR §50.54(f), a copy of the 10 CFR 50.54(f) evaluation which justifies issuance of this letter is in the Public Document Room. Accordingly, all responses should be under oath or affirmation. This request for information is covered by the Office of Management and Budget under

Clearance No. 3150-0011, which expires December 31, 1989. The estimated average burden hours is 8100 person-hours per licensee response, over a 3-year period including assessment of the new requirements, searching data sources, gathering and analyzing the data, and preparing the required reports. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, DC 20503.

Sincerely,

Original signed by
Dennis Crutchfield, Acting Associate
Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
Appendices 1 through 5
w/attachments 1 and 2

APPENDIX 1

GUIDANCE ON THE EXAMINATION OF CONTAINMENT SYSTEM PERFORMANCE (BACK-END ANALYSIS)

1. Background

The role of the containment as a vital barrier to the release of fission products to the environment has been widely recognized. The public safety record of nuclear power plants has been fostered by applying the "defense-in-depth" principle, which relies on a set of independent barriers to fission product release. The containment and its supporting systems are one of these barriers. Containment design criteria are based on a set of deterministically derived challenges. Pressure and temperature challenges are usually based on the design basis loss-of-coolant accident; radionuclide challenges are based on the source term of 10 CFR Part 100. Also, criteria based on external events such as earthquakes, floods, and tornadoes are considered. The margins of safety provided by such practices have been the subject of considerable research and evaluation, and these studies have shown the ability of many containment systems to survive pressure challenges of two to three times design levels. Because of these margins, the various containment types presently used in the United States have the capability to withstand, to varying degrees, many of the challenges presented by severe accidents. For each type of containment, however, there remain failure mechanisms that could lead to either early or late containment failure, depending on both the accident scenarios involved and the containment types.

This appendix discusses the key phenomena and/or processes that can take place during the evolution of a severe accident and that can have an important effect on the containment behavior. In addition, general guidance on the evaluation of containment system performance given the present state of the art of analysis of these phenomena is provided. The evaluation should be a pragmatic exploitation of the present containment capability. It should give an understanding and appreciation of severe accident behavior, should recognize the role of mitigating systems, and should ultimately result in the development of accident management procedures that could both prevent and ameliorate the consequences of some of the more probable severe accident sequences involved. The users of this appendix are referred to Chapter 7 of Volume 1 of NUREG/CR-2300, "PRA Procedures Guide," for a more detailed description of procedures and guidance on containment performance analysis. The additional information provided here summarizes some more recent developments in core melt phenomenology relevant to containment performance, identifies areas of uncertainty, and suggests ways of proceeding with the evaluation of containment performance despite uncertainties, and potential ways of improving containment performance for severe accident challenges. In this regard, the Severe Accident Prevention and Mitigation Features report (NUREG/CR-4920) summarizes insights gained from industry-sponsored PRAs, NUREG-1150, and IDCOR reference plant analyses. The report identifies plant features and operator actions that have been found to be important to either the prevention or the mitigation of severe accidents for a specific plant containment type. The report indicates what may be important to risk and suggests potential improvements in various areas of plant design and operation. These insights and suggestions may be helpful when conducting the IPE and when making decisions on plant improvements.

The systems analysis portion of the IPE identifies accident sequences that occur as a result of an initiating event followed by failure of various systems or failure of plant personnel to respond correctly to the accident. Although the number of possible core melt accident sequences is very large, the number of containment system performance analyses does not have to be as large. The number of sequences can be reduced by grouping those accident sequences that have a similar effect on the plant features that determine the release and transport of fission products.

A containment event tree (CET) could provide a structured way for the systematic analysis of containment phenomena provided:

1. The CET is quantified, i.e., branch point split fractions are propagated for each sequence based on the most recent data base regarding important severe accident phenomena including considerations of uncertainties (e.g., letters from T. Speis, NRC, to A. Buhl, ITC, "Position Papers for the NRC/IDCOR Technical Issues," dated September 22, 1986; November 26, 1986; and March 11, 1987).

2. The system analysis is integrated with the containment analysis so that initiating events and system failures (resulting in core damage) that also impair containment systems are not overlooked.

3. The duration and sequencing of the interacting events are specified, e.g., the times at which core damage and containment failure occur, the time of inventory depletion (in particular, as related to recovery from an accident), the success or failure of equipment or operator responses, and the failure or degradation of support systems that were originally available at the onset of the accident.

2. <u>Status of Containment Systems Prior to Vessel Failure</u>

The role of interfaces between the system analysis (front-end) and the containment performance analysis (back-end) is particularly important from two perspectives. First, the likelihood of core damage can be influenced by the status of particular containment systems. Second, containment performance can be influenced by the status of core cooling systems. Thus, because the influences can flow in both directions between the system analysis (front-end) and the containment performance analysis (back-end), particular attention must be given to these interfaces.

To ensure consistency within entire sequences, the analysis should include a cross-checking sheet of the following by sequence: (1) the sequence frequency, (2) whether the containment is bypassed, (3) whether the containment is isolated, (4) the containment system and reactor system availability, and (5) the approximate source term. This cross-checking sheet would be reviewed by both the systems analyst and the source term analyst to provide added assurance that the status of key systems is treated consistently in the front-end and back-end analyses. Other options to ensure adequate interfaces can be used instead of the cross-checking list identified above.

In order to examine the containment performance, the status of the containment systems and related equipment prior to core melt should be determined. The first CET nodal decision point is to determine the likelihood of whether the

containment is isolated, bypassed, intact, or failed (i.e., a branch point split fraction). This requires analyses of (1) the pathways that could significantly contribute to containment-isolation failure, (2) the signals required to automatically isolate the penetration, (3) the potential for generating the signals for all initiating events, (4) the examination of the testing and maintenance procedures, and (5) the quantification of each containment-isolation failure mode (including common mode failures).

In the early phase of an accident, steam and combustible gases are the main contributors to containment pressurization. The objective of the containment decay heat removal systems such as sprays, fan coolers, and the suppression systems is to control the evolution of accidents that would otherwise lead to containment failure and the release of fission products to the environs. The effectiveness of the several containment decay heat removal systems for accomplishing the intended mitigating function should be examined to determine the probability of successful performance under accident conditions. This includes potential intersystem dependencies as well as the identification of all the specific functions being performed and the determination of the mission time considering potential failure due to inventory depletion (coolant, control air, and control power) or environmental conditions. If, as a result of the accident sequence, the front-line containment decay heat removal systems fail to function, if their effectiveness is degraded, or if the operator fails to respond in a timely manner to the accident symptoms, the containment pressure would continue to increase. In this case, some systems that were not intended to perform a safety function might be called upon to perform that role during an accident. If the use of such systems is considered during the examination, their effectiveness and probability of success for fulfilling the needed safety function should also be examined. Part of the examination should be to determine if adequate procedures exist to ensure the effective implementation of the appropriate operator actions.

3. Phenomena After Vessel Failure

If adequate heat removal capability does not exist in a particular accident sequence, the core will degrade and the containment could potentially overpressurize and eventually fail. Efforts to stabilize the core before reactor vessel failure or to extend the time available for vessel reflood should be investigated. For certain accident groups that proceed past vessel failure, the containment pressurization rate could exceed the capability of the mitigating systems to reject the energy associated with the severe accident phenomena encountered with vessel failure. For each such accident sequence, the molten core debris will relocate, melting through and mixing with materials in its path. Depending on the particular containment geometry and the accident sequence groups, a variety of important phenomena influence the challenges to containment integrity.

The guidance provided below deals with this subject at three levels. The first provides some rather general considerations regarding the nature of these phenomena as they impact containment (Section 3.1). The second level considers the manifestation of these phenomena in more detail within the generic high and low pressure scenarios (Sections 3.1.1 and 3.1.2). Finally, the third level provides some specific guidance particularly regarding the treatment of certain important areas of uncertainty (Section 4).

3.1 General Description of the Phenomena Associated with Severe Accident Considerations

The contact of molten corium with water, referred to as fuel-coolant interaction, can occur both in-vessel and ex-vessel. If the interaction is energetic inside the reactor vessel, it may generate missiles and a rapid pressurization (steam explosion) of the primary system. Early containment failure associated with in-vessel steam explosions is generally considered to be of low enough likelihood to not warrant additional consideration (NUREG-1116). However, smaller, less energetic in-vessel steam explosions are not unlikely and their influence on fission product release and hydrogen generation are still under investigation. If the fuel-coolant interaction occurs ex-vessel, as might happen if molten fuel fell into a water-filled cavity upon vessel meltthrough, it may disperse the corium and lead to rapid pressurization (steam spike) of the containment. In any case, at one extreme, abundant presence of water would favor quenching of the corium mass and the continued dissipation of the decay heat by steaming would lead to containment pressurization. Clearly in the absence of external cooling, the containment will eventually overpressurize and fail, although the presence of extensive, passive heat sinks (structures) within the containment volume would delay the occurrence of such an event. Fuel-coolant interactions can also yield a chemical reaction between steam and the metallic component of the melt, producing hydrogen and the consequent potential for burns and/or explosions.

At the other extreme, when water is not available, the principal interaction of the molten corium is with the concrete floor of the containment. This interaction produces three challenge to containment integrity. First, the concrete decomposition gives off noncondensible gases (CO_2, CO) (of certain composition) that contribute to pressurizing the containment atmosphere. Second, concrete of certain compositions decomposes and releases CO_2 and steam, which can interact with the metallic components in the melt to yield highly flammable CO and H_2, with potential consequences ranging from benign burns at relatively low hydrogen concentrations to rapid deflagrations at high hydrogen concentrations. Third, continued penetration of the floor can directly breach the containment boundary. Also, thermal attack by the molten corium of retaining sidewalls could produce structural failure within the containment causing damage to vital systems and perhaps to failure of containment boundary.

Another type of fuel interaction is with the containment atmosphere. Scenarios can be postulated (e.g., station blackout) in which the reactor vessel and primary system remain at high pressure as the core is melting and relocating to the bottom of the vessel. Continued attack of the molten corium on the vessel lower head could eventually cause the lower head to fail. Because of a potentially high (approximately 2500 psi) driving pressure, the molten corium could be energetically ejected from the vessel. Uncertainties remain related to the effect of the following on direct containment heating: (1) vessel failure area, (2) the amount of molten corium in the lower head at the time of failure, (3) the degree to which it fragments upon ejection, (4) the degree and extent to which a path from the lower cavity to the upper containment atmosphere is obstructed, (5) the fragmented molten corium that could enter and interact with the upper containment atmosphere, and (6) cavity gas temperature. Since the containment atmosphere has small heat capacity, the energy in the fragmented corium could rapidly transfer to the containment atmosphere, causing a

rapid pressurization. The severity of such an event could be further exacerbated by any hydrogen that may be simultaneously dispersed and direct oxidation (exothermic) of any metallic components. Depending upon this and the other factors previously mentioned, this pressurization could challenge containment integrity early in the event.

The BWR Mark I and Mark II containments are normally inerted. Therefore, noncondensible gases such as hydrogen and oxygen released following a severe accident would pressurize the containment, but would not burn or rapidly deflagrate. If the containment is deinerted, additional pressurization events or dynamic loads obtained from global hydrogen burn or detonations must be considered. Local burns are also potentially important as they may degrade the seals around the various penetrations or produce a thermal environment that challenges the operability of important equipment.

Even with the above limited perspective, it should be clear that given a core melt accident, a great deal of the phenomenological progression hinges upon water availability and the outcome of the fuel-coolant interactions; specifically whether a full quench has been achieved and whether the resulting particulates will remain coolable. In general, the presence of fine particulates to any significant degree would imply the occurrence of energetic steam explosions and hence the presence of significant forces that would be expected to disperse the particulates to coolable configurations outside the reactor cavity. Otherwise, the coolability of deep corium beds of coarse particulates is the major concern. A summary of how these mechanisms interface and interact as they integrate into an accident sequence is given below.

3.1.1 Accident Sequences - High-Pressure Scenario

The core melt sequence at high primary system pressure is often due to a station blackout sequence. The high-pressure scenario also represents one of the most significant contributors to risk. The initial stages of core degradation involve coolant boiloff and core heatup in a steam environment. At such high pressures, the volumetric heat capacity of steam is a significant fraction of that of water (about one-third), and one should expect significant core (decay) energy redistribution due to natural circulation loops set up between the core and the remaining cooler components of the primary system. Consensus appears to be developing that as a result of this energy redistribution, the primary system pressure boundary could fail prior to the occurrence of large-scale core melt. The location and the size of failure, however, remain uncertain. For example, concerns have been raised about the possibility of steam generator tube failures and associated containment bypass. If the vessel lower head fails, violent melt ejection could produce large-scale dispersal and the direct containment heating phenomenon mentioned previously. A significant amount of research in the past has not yet produced definitive results on this issue.

Concerns may also be raised about the potentially energetic role of hydrogen within the blowdown process. The presence of hydrogen arises from two complementary mechanisms: (1) the metal-water reaction occurring at an accelerated pace throughout the in-vessel core heatup/meltdown/slump portion of the transient, and (2) the reaction between any remaining metallic components in the melt and the high-speed steam flow that partly overlaps and follows the melt ejection from the reactor vessel. The combined result is the release of rather large quantities of hydrogen into the containment volume within a short time

period (a few tens of seconds). The implication is that the consideration of containment atmosphere compositions and associated burning, explosion, or detonation potential becomes complicated by a whole range of highly transient regimes and large spatial gradients.

A recent independent review of uncertainties in estimates of source terms from severe accidents by an NRC-sponsored panel of experts (NUREG/CR-4883) provided an additional perspective on these issues and made recommendations for their resolution. In particular, "if direct containment heating or containment bypass through steam generator tube failure contribute importantly to risk, this may indicate a need for a hardware modification or a procedural measure to ensure depressurization before primary system failure. An early study of relative merits of the possibilities available would be valuable." The staff is in favor of adopting the panel recommendation and has initiated a research program to study the effect of depressurization on the core melt progression and the potential benefit in preventing direct containment heating.

3.1.2 Accident Sequence - Low-Pressure Scenario

At low system pressure, decay heat redistribution due to natural circulation flow (in steam) is negligible and core degradation occurs at nearly adiabatic conditions. Steam boiloff, together with any hydrogen generation, is continuously released to the containment atmosphere, where mixing is driven by natural convection currents coupled with condensation processes. The upper internals of the reactor vessel remain relatively cold, offering the possibility of trapping fission product vapor and aerosols before they are released to the containment atmosphere. Throughout this core heatup and meltdown process, the potential to significantly load the containment is small. The first possibility for significant energetic loads on the containment occurs when the molten core debris penetrates the lower core support structure and slumps into the lower plenum. The outcome of this interaction cannot be predicted precisely. Thus, a whole range of behavior must be considered in order to cover subsequent events. At the one extreme the interaction is benign, yielding no more than some steam (and hydrogen) production while the melt quickly reagglomerates on the lower reactor vessel head. At the other extreme an energetic steam explosion occurs. It may be possible to distinguish intermediate outcomes by the degree to which the vessel integrity is degraded. In analyzing this phase of the accident scenario, the important tasks are to determine the likelihood of containment failure and to define an envelope of corium relocation paths into the containment. The latter is needed to ensure the assessment of the potential for such a phenomenon as liner meltthrough.

Consideration should also be given to ex-vessel coolability as the corium can potentially interact with the concrete. The non-energetic release (vessel lower head meltthrough) and spreading upon the accessible portions of the containment floor below the vessel needs to be examined. There is a great deal of variability in accessible floor area among the various designs for some PWR cavity designs. The area over which the core debris could spread is rather small given whole-core melts and the resultant pool being in excess of 50 cm deep. In the absence of water, all these configurations would yield concrete attack and decomposition of variable intensity. In the presence of water (i.e., containment sprays), even deep pools may be considered quenchable and coolable. However, the possibility exists for insulating crusts or vapor barriers at the corium-water interface.

Both of these two extremes should be considered. The task is to estimate the range of containment internal pressures, temperatures, and gas compositions as well as the extent of concrete floor penetration and structural attack until the situation has been stabilized. In general, pressurization from continuing core-concrete interactions (dry case) would be considerably slower than from coolable debris configurations (wet case) because of the absence of steam pressurization.

As a final and crucial part of this scenario, one must address the combustible gas effect. This must include evaluation of the quantities and composition of combustible gases released to the containment, local inerting and deinerting by steam and CO_2, as well as hydrogen mixing and transport. Also included should be consideration of gaseous pathways between the cavity and upper containment volume to confirm the adequacy of communication to support natural circulation and recombination of combustible gases in the reactor cavity.

4. General Guidance on Containment Performance

In the approach outlined in this appendix, emphasis is placed on those areas that would ensure that the IPE process considers the full range of severe accidents. The IPE process should be directed toward developing a plant-specific accident management scheme to deal with the probable causes of poor containment performance at each plant. To achieve these goals, it is of vital importance to understand how reliable each of the CET estimates are, and what the driving factors are. Decisions on potential improvements should be made only after appropriately considering the sources of uncertainties. Of course, preventing failure altogether is predicated upon recovering some containment heat removal capability. Given that in either case pressurization develops on the time scale of many hours, feasible recovery actions could be planned as part of accident management.

It is the staff's view that the bulk of phenomenological uncertainties affecting containment response is associated with the high-pressure scenarios. Unless the licensee can demonstrate that the primary system can be reliably depressurized, a low probability of early containment failure should not be automatically assumed. Similarly, for BWRs it should not be assumed that the availability of the automatic depressurization system (ADS) in an event will ensure that reactor vessel failure will always occur at low pressure, since the operability of the ADS, in some plants, depends on maintaining a requisite differential pressure between containment and the reactor coolant systems.

Low-pressure sequences, by comparison, present few remaining areas of controversy. For BWRs, phenomenological uncertainties are associated with the behavior of combustibles and the spreading of the corium on the drywell floor. For PWRs, these areas include the coolability behavior of deep molten corium pools and the behavior of hydrogen (and other combustibles) in the containment atmosphere. The staff's views and guidance concerning each one of these areas is briefly summarized below.

The concerns about deep corium pools arose from experiments with top-flooded melts that exhibited crust formation and long-term isolation of the melt from the water coolant. Such noncoolable configurations would yield continuing concrete attack and a containment loading behavior significantly different from coolable ones. On the other hand, it has been pointed out that small-scale

experiments would unrealistically not favor coolability. The staff views this as an area of uncertainty and recommends that assessments be based on available cavity (spread) area and an assumed maximum coolable depth of 25 cm. For depths in excess of 25 cm, both the coolable and noncoolable outcomes should be considered. Along these lines the IPE should document the geometric details of cavity configuration and flow paths out of the cavity, including any water drain areas into it as appropriate.

With respect to hydrogen, the staff concerns are related to completeness of the current understanding of hydrogen mixing and transport. In general, combustibles accumulate very slowly and only if continuing concrete attack is postulated. For the larger dry containments, because of the large containment volume and slow release rates, compositions in the detonable range may not develop unless significant spatial concentrations exist or significant steam condensation occurs. In general, the containment atmosphere under such conditions would exhibit strong natural circulation currents that would tend to counteract any tendency to stratify. However, condensation-driven circulation patterns and other potential stratification mechanisms could limit the extent of the containment volume participating in the mixing process. For those plants with igniters (ice-condenser and Mark III plants), the buildup of combustibles from continuing corium-concrete interactions could be limited by local ignition and burning. However, oxygen availability as determined from natural circulation flows could limit the effectiveness of this mechanism. Finally, in all cases inerting/deinerting thresholds and ignition aspects need additional attention. The staff recommends that, as part of the IPE, all geometric details impacting the above phenomena (i.e., heat sink distribution, circulation paths, ignition sources, water availability, and gravity drain paths) should be documented in a readily comprehensible form, together with representative combustible source transients.

For normally inerted BWRs, the concerns with combustibles relate to potential burns and/or explosion events in deinerted Mark I or Mark II containments or in the secondary containment building following containment failure. The staff recommends that, unless deinerting can be satisfactorily ruled out by probability, its occurrence and consequences should be included in the event trees. Regarding the secondary containment, the staff believes that consideration of combustibles in it is essential with respect to the reactor building effectiveness in limiting the source term.

Finally, uncertainties arise for all plants because of lack of knowledge on how the corium will spread following discharge from the reactor vessel. For Mark I containments, such uncertainties impact the configuration of the corium-concrete interaction process and also the potential for drywell liner meltthrough. It is recommended that an assessment of the debris coolability, based on available water sources, should be performed to determine the possibility for liner meltthrough. For Mark II containments, uncertainties are associated with the retention of corium on the drywell floor (and associated corium-concrete interactions) and the extent of fuel-coolant interactions in the suppression pool. For PWR containments, the reactor cavity configuration will influence the potential for direct attack of the liner by dispersed debris, as well as the potential for basemat failure or structural failure due to thermal attack. The staff recommends that the IPE document describe the detailed geometry (including curbs, standoffs) of the drywell floor.

As discussed earlier, a CET provides a structured way for a systematic analysis of containment phenomena. Separate CETs representing the high-pressure and low-pressure sequences deal with uncertainties discussed earlier.

In general terms, and consistent with the overall IPE objectives, the staff guidance on the approach to the back-end analysis can be summarized as follows:

1. The approach should focus on containment failure mechanisms and timing. Releases should be based on corresponding release categories and associated detailed quantifications from reference plant analyses and applied to the plant being examined.

2. All severe accident sequences that meet the criteria of Appendix 2 should be considered and reported.

3. System/human response should be realistically integrated with phenomenological aspects into simplified, but realistic, containment event trees for the plant being examined. Allowance should be made for the probability of recovery or other accident management procedures (particularly for long-term responses).

4. The quantification of the containment event trees should both (a) clearly take into account the expected progression of the accident and (b) aim to envelop phenomenological behavior (i.e., account for uncertainties). This implies:

   a. Identification of the most probable list of potential containment failure mechanisms applicable to the plant under consideration (e.g., see Table 7-1, NUREG/CR-2300).

   b. Use of existing structural analyses to determine the ultimate pressure capability of the containment, i.e., the quasi-static internal pressure resulting in containment failure. These should be modified as necessary to take into account any unique aspects that could substantially modify the range of possible failure pressures.

   c. Use of available separate-effects analyses for the other potential containment failure mechanisms to determine other failure modes to which the plant might be vulnerable. As stated earlier, there are some severe accident phenomenological issues (e.g., direct containment heating and containment shell meltthrough) where research has not produced conclusive results on the challenges that these phenomena could pose to containment integrity. Consideration must be given to strategies to deal with those severe accident issues. For example, although there appears to be no consensus on whether water availability will fully quench the debris and keep it coolable and hence prevent Mark I containment shell meltthrough, there is a broad agreement that the presence of water will scrub the fission products and could substantially reduce the radionuclide released even if containment shell meltthrough were to occur. Utilities should be aware of these insights and experience when conducting the IPE and should develop appropriate strategies to deal with those phenomenological issues while awaiting their generic resolution as discussed in Section 4 of the IPE generic letter.

d. Development of a plant-specific probability distribution function of failure likelihood for the range of failure pressures.

e. Any claim of decontamination factors for the secondary containment in the analyses should consider the possibility of no natural circulation, resulting in less time for aerosol deposition, as well as localized hydrogen burns causing reactor building failure and forcing the reactor building atmosphere out into the environment.

5. Documentation should be presented concerning how any calculation was performed, what assumptions have been made, and how these phenomena couple to other aspects of the analysis. Any use of codes within the IPE to calculate accident progression up to and including the source term calculation should be described along with the circumstances under which the code was used, the version of the code used, any code revisions used, the key modeling and input assumptions, and the calculated results.

6. The insights gained from the containment performance analysis should be factored into the utility's accident management program.

APPENDIX 2

CRITERIA FOR SELECTING IMPORTANT SEVERE ACCIDENT SEQUENCES

Sequence Selection Criteria

The following screening criteria should be used to determine which potentially important functional sequences* and functional failures (based on the procedure established in NUREG/CR-2300) that might lead to core damage or unusually poor containment performance should be reported to the NRC in the IPE submittal. They do not represent a threshold for vulnerability. All numerical values given in this appendix are "expected"** values.

1. Any functional sequence that contributes 1E-6*** or more per reactor year to core damage,

2. Any functional sequence that contributes 5% or more to the total core damage frequency,

3. Any functional sequence that has a core damage frequency greater than or equal to 1E-6 per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release categories of WASH-1400,

4. Functional sequences that contribute to a containment bypass frequency in excess of 1E-7 per reactor year, or

5. Any functional sequences that the utility determines from previous applicable PRAs or by utility engineering judgment to be important contributors to core damage frequency or poor containment performance.

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*"Sequence" is used here to mean a set of faults, usually chronological, that result in the plant consequence of interest, i.e., either a damaged core or unusually poor containment performance. A systemic sequence is a set of faulted systems that summarizes by systems a set of component failures resulting in a damaged core or unusually poor containment performance. A functional sequence is a set of faulted functions that summarizes by function a set of systems faults which would result in the consequence of interest.

**For those cases where only point estimates are generated, the licensee shall propose a suitable factor that adjusts the overall value to the "expected" level.

***1E-6 denotes abbreviated scientific notation for 1×10^{-6}.

APPENDIX 3

ACCIDENT MANAGEMENT

There already is an international consensus that the cause and consequences of a severe core damage accident can be greatly influenced by the operator's actions. In addition, the ability of essential equipment to survive the environment resulting from severe accidents is an important consideration in mitigating a severe core damage accident and managing its progression. The failure of essential equipment can (1) incapacitate or remove systems needed to respond to severe accidents or (2) misinform the operator.

The NRC has initiated a research program to examine the efficacy of generic accident management strategies. We intend to periodically meet with industry (NUMARC) to compare the results of our respective programs. However, the staff has done some preliminary work in defining the key elements of a severe accident management program.

Since your IPE results will ultimately play a significant role in the development of such a program for your plant, we are providing you with the results of our work at this time. The main elements of an accident management program should address: (1) the organizational responsibilities and structure needed to direct the responses to a severe accident, (2) the instrumentation, procedures, and alarms needed to diagnose severe accidents, and the procedures and equipment needed to accomplish the functions necessary to prevent and to mitigate leading accidents, and (3) the procedures and training needed for operators to be skilled in possible remedial actions.

Suggested Elements of an Accident Management Program

1. Organization

The first element of any severe accident management program is to assign responsibilities for dealing with these accidents and to identify the necessary organizational structure.

The utility should decide which operators are to be trained to manage severe accidents or if a separate evaluation team is to be established to direct the operators. Clear lines of decisionmaking authority should be established. For example, if containment venting is an option that could conceivably be considered during the course of an accident to prevent overpressure failure, then the person responsible for making that decision should be clearly identified to all involved personnel. Analyses of ultimate containment strength, the venting pressure, and the advantages, disadvantages, and potential consequences should also have been evaluated beforehand, and the decisionmakers should be properly trained from the evaluation results to make an informed decision.

2. Instrumentation and Equipment

Practically every aspect of plant operation is likely to be involved in accident management. Coordination among the various organizational units is vital for communicating the status and the control of needed equipment. It should be clear (1) what information is needed to make decisions, (2) who is responsible

for obtaining the information, (3) what instruments plant personnel can rely on to determine the status of the plant, and (4) what essential equipment is needed to mitigate severe accidents and the time interval for which it is needed. Survivability of specific equipment needs to be evaluated by establishing whether the qualification of equipment for design basis events is sufficient to support the assumed performance of this equipment during severe accidents.

For sequences with a significant potential to progress beyond core melt, means of maintaining containment integrity is the main goal. Heat removal from the containment and retention of fission products are the most important functions. Equipment needed to accomplish these functions should have been identified and appropriate preparations made. All reasonable preparations to enable operators to recognize approaching containment failure, to assess possible remedial actions, and to accomplish the necessary functions should be provided. Potentially adverse action should be identified and evaluated. For example, recovery and initiation of containment sprays after the containment has a substantial quantity of steam and hydrogen can condense the steam and may leave a detonable mixture of hydrogen. Similarly, spraying into a containment that has been vented could result in a vacuum and possible implosion.

If special equipment might be needed to both prevent and mitigate severe accidents, provisions might be made to ensure its timely availability. The responsibility to take such action should be assigned, and the individuals responsible should know where to procure the needed equipment.

3. Procedures and Training

The accident management plan should be developed to accomplish these functions for each set of the leading accident sequences despite the degraded state of the plant. There should be consistency and smooth transition between the emergency operating procedures and the accident management plan. The plan should be checked against the existing organizational structure to ensure that responsibilities for managing each accident are clearly defined and the responsible personnel are adequately trained.

APPENDIX 4

DOCUMENTATION

At a minimum, the following information on the IPE should be documented and submitted to the NRC:

1. Certification that an IPE has been completed and documented as requested by the provisions contained in this generic letter. The certification should also identify the measures taken to ensure the technical adequacy of the IPE and the validation of the results, including any uncertainty, sensitivity, and importance analysis.

2. A list of all initiating events, the containment phenomena, and the damage states examined.

3. All function event trees and containment event trees (including quantification) as well as all data (including origin and method of analysis). The fault trees (or equivalent system failure models) for the systems identified, using the criteria of Appendix 2, as main contributors to core damage or unusually poor containment performance should also be provided.

4. The support state models for the IDCOR IPEMs, including descriptions of all applicable findings from the visual inspections.

5. A description of each functional sequence selected by the criteria of Appendix 2, including discussion of accident sequence progression, specific assumptions, and human recovery action.

6. The estimated core damage frequency and the likelihood or conditional probability of a large release. The timing of significant large releases for each of the leading functional sequences. A list of analysis assumptions with their basis should be provided along with the source of uncertainties.

7. Identification of the USI(s) and GSI(s), if applicable, that have been assessed to estimate their contribution to the core damage frequency or to unusually poor containment performance.

8. A description of the technical basis for resolving any USI or GSI when applicable.

9. A list of the potential improvements, if any (including equipment changes as well as changes in maintenance, operating and emergency procedures, surveillance, staffing, and training programs) that have been selected for implementation and a schedule for their implementation or that are already implemented. Include a discussion of the anticipated benefit as well as any drawbacks.

10. A description of the review performed by a utility party not directly involved in producing the IPE to evaluate or oversee the IPE review.

11. Documentation on the level of licensee staff involvement in the IPE.

Retained Information

The documentation pertaining to the examination that must be retained by the utility for the duration of the license or until superseded includes applicable event trees and fault trees, current versions of the system notebooks if applicable, walk-through reports, and the results of the examination. In general, all documents essential to an audit of the examination should be retained. In addition, the manner in which the validity of these documents has been ensured must be documented. For any actions taken by the operators for which credit is allowed in the IPE, the licensee should establish a plant procedure, to be used by those plant staff responsible for managing a severe accident should one occur, that provides assurance that the operators can and will take the required action. Plant owner groups are encouraged to develop generic guidelines from which utilities can develop plant-specific accident management programs and/or procedures.

APPENDIX 5

DECAY HEAT REMOVAL VULNERABILITY INSIGHTS

As part of the Unresolved Safety Issue (USI) program, six limited scope PRAs were performed under the USI A-45 project, "Shutdown Decay Heat Removal Requirements," to assess the decay heat removal (DHR) function in existing plants.* The results showed that DHR-related core damage risk is in a range, on some plants, where attention may be warranted regarding whether or not such risks can be lowered in a cost-effective manner. The results also showed that the sources of DHR-related core damage risk are highly plant specific.

The following insights have been gained as a result of those six PRAs. The insights are summarized here in order to assist licensees in the conduct of their IPEs as they relate to their search for potential core damage risk associated with DHR-related severe accident sequences. Although licensees are requested in the generic letter to proceed with the examination only for internally initiated events at the present time, insights from both internal and external events are provided in this appendix to indicate what may be important to decay heat removal function vulnerabilities when performing the IPE for externally initiated events.

Areas where such cost-effective improvements might be possible were identified for severe accident sequences initiated by transients and small-break loss-of-coolant accidents and were frequently related to lack of redundancy, separation, and physical protection in safety trains for internal fires, floods, sabotage, and seismic events.

Such areas for possible improvement were particularly apparent in plant support systems. At the support system level, there is often less redundancy, less separation and independence between trains, poorer overall general arrangement of equipment from a safety viewpoint, and much more system sharing as compared to the higher level systems. These situations suggest the possible need to investigate corrective actions that could reduce the probability that single events such as a fire, flood, or insider sabotage could disable multiple trains (or single trains with a multiple purpose) thereby creating an inability to cool the plant.

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* See the following NUREG/CR reports:

4448, "Shutdown Decay Heat Removal Analysis of a General Electric BWR3/Mark I," March 1987.

4458, "Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop Pressurized Water Reactor," March 1987.

4713, "Shutdown Decay Heat Removal Analysis of a Babcock and Wilcox Pressurized Water Reactor," March 1987.

4762, "Shutdown Decay Heat Removal Analysis of a Westinghouse 3-Loop Pressurized Water Reactor," March 1987.

4767, "Shutdown Decay Heat Removal Analysis of a General Electric BWR4/Mark I," July 1987.

4710, "Shutdown Decay Heat Removal Analysis of a Combustion Engineering Pressurized Water Reactor," July 1987.

Human errors were found to be of special significance. The six studies modeled errors of omission (e.g., delays or failures in performing specified actions), and it was found that in many cases the resulting risk was very sensitive to the assumptions made and to the way such errors were modeled.

Consequently, great care is warranted in the development of human error models. In addition, it is likely that errors of commission are also important (i.e., where the operator misdiagnoses a situation and takes an improper action that is not be related to the actual, current plant situation). Although such "cognitive" errors are much more difficult to model, efforts to take them into account will result in a more complete picture of DHR-related risk.

Of equal importance to human errors is the credit that is allowed for recovery actions, which can have a very significant effect upon the resulting risk. Some of the more important recovery actions are recovering offsite power, fixing local faults of batteries or diesel generators, actuating safety systems manually, re-aligning auxiliary feedwater steam and feedwater flowpaths, and manually opening locally failed motor-operated valves. Considering the importance of such human recovery actions, considerable effort is justified in the development of the methods and assumptions used in these areas.

Transient events that are initiated or influenced by a loss of offsite power were found to contribute significantly to risk. A new rule, 10 CFR 50.63, has been issued June 21, 1988 (53 FR 23203) as a resolution to USI A-44, "Station Blackout." Implementation of this rule will reduce the risk from such events.

For PWRs, the ability to cool the plant through "feed and bleed" operations could have a significant effect upon the DHR-related core damage risk. However, care must be taken that feed and bleed operations would actually be undertaken in a real emergency situation in sufficient time to prevent core uncovery and subsequent damage. In view of the potential benefits, significant effort might be justifiable in ensuring that procedures and training are actually in place sufficient to warrant credit for feed and bleed cooling.

Just as the origins of DHR-related risk are plant specific, the effects of corrective actions are also quite plant specific and must be evaluated on a plant-by-plant basis. In choosing which potential corrective actions to investigate in more detail, a general principle is that the modifications having the highest potential for reducing the risk, for the lowest cost, will be those that increase the redundancy or availability of systems shared between units.

In summary, both the DHR-related risk and the effects of various corrective actions are highly plant specific. The dominant risks are divided between internal and external causes, and the areas of support systems and human response are of particular significance. Studies show that various cost-effective corrective actions may be possible to reduce DHR-related core damage risk after its source has been identified.

ATTACHMENT 1

CLOSURE OF SEVERE ACCIDENT ISSUES
FOR OPERATING REACTORS

(Excerpted from SECY 88-147)

The Commission has ongoing a number of programs related to severe accident behavior in operating light water reactors. Each program addresses a specific aspect of severe accident behavior and may in fact result in a proposed specific action on the part of the staff or Commission towards the regulated industry. However, neither the staff nor Commission has yet defined for the industry which programs are critical to resolving the severe accident issues for their plants and what specific steps must be taken by each licensee to achieve this resolution.

Completion of this resolution process is termed "closure" of severe accident issues. Actions resulting from two tracks; namely, generic issues and plant-specific issues, must be taken for severe accident closure. Closure for generic severe accident issues will be obtained when the Commission takes action in the form of rulemaking, or states whatever its required approach is. Closure for plant-specific severe accident issues will be obtained when each licensee has completed certain evaluations and implemented certain programs such that events which comprise the dominant contributions to risk for each plant are identified and that practical enhancements to the design, procedures, and operation are made such that further improvements can no longer be justified by backfit analysis pursuant to 10 CFR 50.109. However, specific plant and operational improvements may be identified which do not meet the backfit rule, but if implemented, would significantly alter the risk profile of the plant, improve the balance of reliance on both prevention and mitigation, or substantively reduce uncertainties in our understanding. Any such improvements identified will be brought forward to the Commission with recommended action on a case-by-case basis. Closure of a single issue or combination of issues is achieved when the above is satisfied for that issue or those issues addressed.

It should be noted that "closure" does not imply that all severe accident activities will cease. Certain activities, such as research in the areas of severe accident phenomena and human performance will continue beyond "closure." These activities are designed to provide confirmation of previous judgments. It is expected that as a result of continuing research, experience, and other activities, additional issues or questions regarding judgments related to severe accidents may arise. These will be considered and disposed of on a case-by-case basis, and are not expected to bring into question the previous conclusions regarding closure.

The following sections describe in detail the steps that each licensee is expected to complete in order to achieve severe accident closure for each of its operating reactors.

1. Completing Individual Plant Examinations (IPEs)

The IPE program is intended to be "an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be plant specific and might be missed absent a systematic search."

Each licensee is expected to perform an IPE using a method acceptable to the staff. As will be described in the staff generic letter implementing the IPE, the staff expects that in many cases utilities, in the performance of their IPEs, may find and will voluntarily remedy uncovered vulnerabilities by making the necessary safety improvements (conforming to the requirements of 10 CFR 50.59). However, through the review of IPE submittals, the staff may find it necessary to employ established plant-specific backfit criteria to assure that justifiable corrections are made.

For the phase of the evaluation associated with identification of dominant core melt sequences (commonly referred to as the "front end" analysis of a PRA), there is little controversy regarding methods, and we expect the industry decision process with respect to potential modifications to be straightforward. For the phase of the evaluation associated with core melting, release of molten core to the containment, and containment performance, the staff recognizes that for a few of the phenomena, notably in areas which affect containment performance, there is a wide range of views about their relative probability as well as their consequences. For these issues additional research and evaluations will be needed to help reduce the wide range of uncertainties. Because of concern over the ability of containments to perform well during some severe accidents, the staff is conducting a Containment Performance Improvements Program (for more details see Item 3 below). This program complements the IPE program and is intended to focus on resolving generic containment challenges, including issues associated with the phenomena mentioned above.

The NRC and industry currently have ongoing research programs to address these few issues. However, until a sufficient understanding of these phenomena is developed, each licensee will be faced with the need to be able to understand the potential range of probabilities and consequences associated with these issues.

Accordingly, we would expect each licensee to implement a Severe Accident Management Program which provides training and guidance to their operational and technical staff on understanding and recognizing the potential consequences of these phenomena.

We do not plan to require a licensee to consider external events in its IPE at this time. The staff is currently studying methods it would find acceptable for examining plants for severe accident vulnerabilities from external events, and will be meeting with NUMARC regarding these methods as well as the scope of an external event examination. We expect completion of the methods development within 12 to 18 months. Closure with respect to external events will be achieved upon completion of an examination of each plant, as needed, for external event vulnerabilities consistent with the conclusions of the staff studies described above.

2. Accident Management

The staff has concluded that significant risk reductions can be achieved through effective severe accident management. We also believe that the IPE conclusions reached by licensees for their plants will explicitly rely on certain operator actions, or on operators not taking actions which could adversely affect both the probability and consequences of a severe accident.

Hence, a key element to severe accident closure for each plant will be the implementation of a Severe Accident Management Program. Since information on severe accident phenomena and effective accident management strategies will continue to be developed by both NRC and industry over the next several years, closure is not predicated on having a "complete" accident management program in place. Rather, closure is based on each licensee having an Accident Management Program framework in place, that can be expanded, modified, etc. to accommodate new information as it is developed.

3. Containment Performance Improvements

As a result of concerns related to the ability of containments to withstand some generic challenges associated with severe accidents, the staff has undertaken a program to determine what, if any, actions should be taken to reduce the vulnerability of containments to severe accident challenges, and to reduce the magnitude of releases that might result from such challenges.

Staff efforts have first focused on the BWR MARK I containment. The staff studies are primarily focused on the potential generic vulnerabilities of these containments, and not plant unique vulnerabilities, which is the primary focus of the IPEs. The staff schedule calls for an interim report on BWR MARK Is to be submitted to the Commission in June of this year, with final recommendations due in the fall of this year. The other types of containments are to be assessed by the fall of 1989.

The IPE generic letter is now expected to be issued by July of this year, and licensees will have approximately four months to respond identifying their plan for conducting the IPEs. Following the four-month period, it is expected they will commence with their IPEs. It is further expected that any modifications to Mark I containments that the staff may recommend will be available to the industry before they start their IPEs. For the other containment types, the fact that any staff recommendations will not be available until after they have commenced with their IPEs is a concern. However, the IPE generic letter will state that the staff does not expect the industry to make any major modifications to their containments until the information associated with the generic issues which affect containment performance has been developed by the staff. Hence, the industry will not be placed in a position of having to implement improvements before all containment performance decisions have been made.

4. Use of Safety Goal in the Closure Process

The staff expects to use safety goal policy and objectives, including the 10^{-6}/reactor-year "large release" guideline, to assist in the resolution and closure of severe accident issues. Resolution and closure of issues are expected to be of two different types, either plant unique or generic. Safety

goals and objectives are to be used only for the resolution of generic issues, i.e., severe accident issues common to a defined generic class of plants. Resolution of plant unique issues is to be accomplished on a case by case basis, using the information developed by Individual Plant Examinations (IPE) as is described in Section 1.

The staff is preparing a Safety Goal Policy Implementation Plan (Revised) that incorporates the following, as directed by the Commission (Staff Requirements Memorandum dated November 6, 1987):

(1) Information on how the staff proposes to implement OGC guidance on the use of averted on-site costs in backfit analyses.

(2) Whether averted off-site property damage costs should be included in a more explicit manner in backfit analyses.

(3) Whether $1,000/person-rem remains an appropriate cost/benefit criterion.

(4) A discussion of options for defining a "large release."

(5) A discussion of options for specifying appropriate plant performance objectives.

(6) Responses to Commissioner Bernthal's questions regarding population density considerations, and whether it would be acceptable for a plant to have no containment if it met the large release criterion by prevention of core melt (core damage) alone.

This plan will also reflect the consideration given by the staff to ACRS recommendations and the results of several meetings with the ACRS on this subject.

Resolution of severe accident generic issues using safety goal objectives is expected to proceed as follows. PRA information from a variety of sources, including both staff generated PRAs, (e.g., NUREG-1150) and utility generated PRAs (IPE) will be used to make comparisons with applicable safety goal objectives in accordance with the implementation plan. The staff will identify the reasons why particular plants appear to meet or not meet these objectives and assess these reasons in relation to current regulatory requirements. This assessment will constitute a testing of the effectiveness of these requirements or their implementation and is expected to result in the identification of potential changes to regulatory requirements that, for some plants, would be expected to result in safety enhancements. These, in turn, will be subject to appropriate regulatory analysis as provided in the Commission's backfit rule 10 CFR 50.109. Those that can be shown to provide substantial safety benefit and are cost-effective will be proposed to the Commission for backfit, possibly in the form of rulemaking. The staff expects that this process would have no impact on classes of plants for which there is reasonable assurance that safety goal objectives are met. This expectation is based upon the intent to identify those features of design and/or performance that are already in place at plants meeting safety goal objectives and to structure any new requirements such that they do not require changes or additions at these plants.

The staff's revised Safety Goal Implementation Plan is scheduled to reach the Commission in August, 1988. The first application is expected to be reflected in the staff's recommendations to the Commission in the Fall of 1988 on potential improvements to BWR MARK I severe accident containment performance.

5. Summary of Closure Process

In summary, the steps which each licensee is expected to take to achieve closure on severe accidents for its plants are as follows:

- Complete the IPEs; identify potential improvements, evaluate and fix as appropriate.

- Develop and implement a framework for an Accident Management Program that can accommodate new information as it is developed.

- Implement any Commission-approved generic requirements resulting from the staff Containment Performance Improvements Program; this should constitute closure of containment performance generic issues.

While programs for improved plant operations and research in the area of severe accidents will continue, completion of the above by a licensee is considered to constitute "closure" of the severe accident issue for the plant in question. Specific issues that may arise in the future as a result of ongoing research will be treated on a case-by-case basis and will not affect the closure process.

ATTACHMENT 2

LIST OF REFERENCES OF THE IDCOR PROGRAM REPORTS AND KEY NRC REPORTS

IDCOR Reports

Tech. Report No.	Title
1.1	Safety Goal/Evaluation Implications for IDCOR
2.1	Ground Rules for Industry Degraded Rule Making Program
3.1	Define Initial Likely Sequences
3.2	Assess Dominant Sequences
3.3	Selection of Dominant Sequences
4.1	Containment Event Trees
5.1	Human Error Effects on Dominant Sequences
6.1	Risk Significant Profile for ESF and Other Equipment
7.1	Baseline Risk Profile for Current Generation Plants
9.1	Preventive Methods to Arrest Sequences of Events Prior to Core Damage w/Revision 1
10.1	Containment Structural Capability of LWRs
11.1/11.5	Estimation of Fission Product and Core Material Characteristics
11.2	Identifying Pathways of Fission Product Transport
11.3	Fission Product Transport in Degraded Core Accidents
11.6	Resuspension of Deposited Aerosols
11.7	FAI Aerosol Correlation
12.1	Hydrogen Generation During Severe Core Damage Sequences
12.2	Hydrogen Distribution in Reactor Containment Buildings
12.3	Hydrogen Combustion in Reactor Containment Buildings
13.2-3	Evaluation of Means to Prevent, Suppress or Control Hydrogen Burning in Reactor Containments
14.1A	Key Phenomenological Models for Assessing Explosive Steam Generation Rates
14.1B	Key Phenomenological Models for Assessing Non-Explosive Steam Generation Rates
15.1	Analysis of In-Vessel Core Melt Progression
15.1A	In-Vessel Core Melt Progression Phenomena
15.1B	In Vessel Core Melt Progression Phenomena
15.2A	Effect of Core Melt Accidents on PWRs with Top Entry Instruments
15.2B	Final Report on Debris Coolability, Vessel Penetration, and Debris Dispersal
15.3	Core-Concrete Interactions
16.1	Assess Available Codes, Define Use and Follow and Support Ongoing Activities
16.1A	Review of MAAP PWR and BWR Codes
16.2-3	MAAP Modular Accident Analysis Program User's Manual, Vols. I & II
16.4	Analysis to Support MAAP Phenomenological Models
17	Equipment Survivability

17.5	Draft Final Report: An Investigation of High-Temperature Accident Conditions for Mark-1 Containment Vessels
18.1	Evaluation of Atmospheric and Liquid Pathway Dose
18.2	Completion of Conditional Complementary Cumulative Distribution Functions
19.1	Alternate Containment Concepts
20.1	Core Retention Devices
21.1	Risk Reduction Potential
22.1	Safe Stable States
23.1	Uncertainty Studies for PB, GG, Zion, Sequoyah
23.1B	Peach Bottom - Integrated Containment Analysis
23.1Z	Zion - Integrated Containment Analysis
23.1S	Sequoyah - Integrated Containment Analysis
23.1GG	Grand Gulf - Integrated Containment Analysis
23.	MAAP Uncertainty Analysis
23.5	Containment Bypass Analysis
24.4	Operator Response to Severe Accidents
85.1	IDCOR 85 Program Plan
85.2	Technical Support for Issue Resolution
85.3	IPEM A1 Thru B2
	IPE Applications PB, Susquehanna, Zion, Oconee, BWR User's Guide
85.4	Reassessment of Emergency Planning Requirements with Present Source Terms
85.5A	Revised Source Terms
85.5B	Source Terms and Emergency Planning
86.20C	Verification of IPE for Oconee
86.3A2	IPE Source Term Methodology for PWRs
86.3B2	IPE Source term Methodology for BWRs
86.20G	Verification of IPE for Grand Gulf
86.25H	Verification of IPE for Shoreham

NRC and NRC Contractor Reports

Tech. Report No.	Title
NUREG-0956	Reassessment of the Technical Bases for Estimating Source Terms
NUREG-1032	Evaluation of Station Blackout Accidents at Nuclear Power Plants
NUREG-1037	Containment Performance Working Group Report
NUREG-1079	Estimates of Early Containment Loads from Core Melt Accidents
NUREG-1116	A Review of the Current Understanding of the Potential for Containment Failure from In-Vessel Steam Explosions
NUREG-1150 Volumes 1-3	Reactor Risk Reference Document
NUREG-1265	Uncertainty Papers on Severe Accident Source Terms
NUREG/CR-2300	PRA Procedures Guide
NUREG/CR-2815	Probabilistic Safety Assessment Procedures Guide
NUREG/CR-4177 Volumes 1-2	Management of Severe Accidents
NUREG/CR-4458	Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop PWR
NUREG/CR-4550 Volumes 1-4	Analysis of Core Damage Frequency from Internal Events
NUREG/CR-4551 Volumes 1-4	Evaluation of Severe Accident Risks and the Potential for Risk Reduction
NUREG/CR-4696	Containment Venting Analysis for the Peach Bottom Atomic Power Station
NUREG/CR-4700 Volumes 1-4	Containment Event Analysis for Postulated Severe Accidents
NUREG/CR-4767	Shutdown Decay Heat Removal Analysis of a GE BWR4/Mark I
NUREG/CR-4881	Fission Product Release Characteristics into Containment Under Design Basis and Severe Accident Conditions
NUREG/CR-4883	Review of Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants
NUREG/CR-4920 Volumes 1-5	Assessment of Severe Accident Prevention and Mitigation Features
NUREG/CR-5132	Severe Accident Insights Report

Enclosure 2

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter No.	Subject	Date of Issuance	Issued To
89-16	INSTALLATION OF A HARDENED WETWELL VENT (GENERIC LETTER 89-16)	09/01/89	ALL GE PLANTS
88-20 SUPPLEMENT 1	GENERIC LETTER 88-20 SUPPLEMENT NO. 1 (INITIATION OF THE INDIVIDUAL PLANT EXAMINATION FOR SEVERE VULNERABILITIES 10 CFR 50.54(f))	08/29/89	ALL LICENSEES HOLDING OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTOR FACILITIES
89-15	EMERGENCY RESPONSE DATA SYSTEM GENERIC LETTER NO. 89-15 CORRECT ACCESSION NUMBER IS 8908220423	08/21/89	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS
89-07	SUPPLEMENT 1 TO GENERIC LETTER 89-07, "POWER REACTOR SAFEGUARDS CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS"	08/21/89	ALL LICENSEES OF OPERATING PLANTS, APPLICANTS FOR OPERATING LICENSES, AND HOLDERS OF CONSTRUCTION PERMITS
89-14	LINE-ITEMS TECHNICAL SPECIFICATION IMPROVEMENT - REMOVAL OF 3.25 LIMIT ON EXTENDING SURVEILLANCE INTERVALS (GENERIC LETTER 89-14)	08/21/89	ALL LICENSEES OF OPERATING PLANTS, APPLICANTS FOR OPERATING LICENSES, AND HOLDERS OF CONSTRUCTION PERMITS
89-13	GENERIC LETTER 89-13 SERVICE WATER SYSTEMS PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT	7/18/89	LICENSEES TO ALL POWER REACTORS BWRS, PWRS, AND VENDORS IN ADDITION TO GENERAL CODES APPLICABLE TO GENERIC LETTERS
89-12	GENERIC LETTER 89-12: OPERATOR LICENSING EXAMINATIONS	7/6/89	LICENSEES TO ALL POWER REACTORS BWRS, PWRS, AND VENDORS IN ADDITION TO GENERAL CODES APPLICABLE TO GENERIC LETTERS

ENCLOSURE 3

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

September 1, 1989

TO: ALL HOLDERS OF OPERATING LICENSES FOR NUCLEAR POWER REACTORS WITH MARK I CONTAINMENTS

SUBJECT: INSTALLATION OF A HARDENED WETWELL VENT (GENERIC LETTER 89-16)

As a part of a comprehensive plan for closing severe accident issues, the staff undertook a program to determine if any actions should be taken, on a generic basis, to reduce the vulnerability of BWR Mark I containments to severe accident challenges. At the conclusion of the Mark I Containment Performance Improvement Program, the staff identified a number of plant modifications that substantially enhance the plants' capability to both prevent and mitigate the consequences of severe accidents. The improvements that were recommended 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. The staff as part of that effort also evaluated various mechanisms for implementing of these plant improvements so that the licensee and the staff efforts would result in a coordinated coherent approach to resolution of severe accident issues in accordance with the Commission's severe accident policy.

After considering the proposed Mark I Containment Performance Program (described in SECY 89-017, January 1989), 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, that is, hardened wetwell vent capability, should be evaluated by licensees as part of the Individual Plant Examination (IPE) 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.

A copy of Boston Edison Company's description of the vent modification is enclosed for your information. For the remaining plants, the staff has been directed to initiate plant-specific backfit analyses for each of the Mark I plants to evaluate the efficacy of requiring the installation of hardened wetwell vents. Where the backfit analysis supports imposition of that requirement, the staff is directed to issue orders for modifications to install a reliable hardened vent.

3pp.

The staff believes that the available information provides strong incentive for installation of a hardened vent. First, it is recognized that all affected plants have in place emergency procedures directing the operator to vent under certain circumstances (primarily to avoid exceeding the primary containment pressure limit) from the wetwell airspace. Thus, incorporation of a designated capability consistent with the objectives of the emergency procedure guidelines is seen as a logical and prudent plant improvement. Continued reliance on pre-existing capability (non-pressure-bearing vent path) which may jeopardize access to vital plant areas or other equipment is an unnecessary complication that threatens accident management strategies. Second, implementation of reliable venting capability and procedures can reduce the likelihood of core melt from accident sequences involving loss of long-term decay heat removal by about a factor of 10. Reliable venting capability is also beneficial, depending on plant design and capabilities, in reducing the likelihood of core melt from other accident initiators, for example, station blackout and anticipated transients without scram. As a mitigation measure, a reliable wetwell vent provides assurance of pressure relief through a path with significant scrubbing of fission products and can result in lower releases even for containment failure modes not associated with pressurization (i.e., liner meltthrough). Finally, a reliable hardened wetwell vent allows for consideration of coordinated accident management strategies by providing design capability consistent with safety objectives. For the aforementioned reasons, the staff concludes that a plant modification is highly desirable and a prudent engineering solution of issues surrounding complex and uncertain phenomena. Therefore, the staff strongly encourages licensees to implement requisite design changes, utilizing portions of existing systems to the greatest extent practical, under the provisions of 10 CFR 50.59.

As noted previously, for facilities not electing to voluntarily incorporate design changes, the Commission has directed the staff to perform plant-specific backfit analyses. In an effort to most accurately reflect plant specificity, the staff herein requests that each licensee provide cost estimates for implementation of a hardened vent by pipe replacement, as described in SECY 89-017. In addition, licensees are requested to indicate the incremental cost of installing an ac independent design in comparison to a design relying on availability of ac power. In the absence of such information, the staff will use an estimate of $750,000. This estimate is based on modification of prevalent existing designs to bypass the standby gas treatment system ducting and includes piping, electrical design changes, and modifications to procedures and training.

The NRC staff requests that each licensee with a Mark I plant provide notification of its plans for addressing resolution of this issue. If the licensee elects to voluntarily proceed with plant modifications, it should be so noted, along with an estimated schedule, and no further information is necessary. Otherwise, the NRC staff requests that the above cost information be provided. In either event, it requests that each licensee respond within 45 days of receipt of this letter.

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires December 31, 1989. The estimated average burden hours are 100 person hours per licensee response, including searching data sources, gathering and analyzing the data, and preparing the required letters. These estimated average burden hours pertain only to the identified response-related matters and do not include the time for actual implementation of the requested actions. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Record and Reports Management Branch, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

If you have any questions regarding this matter, please contact the NRC Lead Project Manager, Mohan Thadani, at (301) 492-1427.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Description of Vent Modification at the Pilgrim Nuclear Power Station
2. List of Most Recently Issued Generic Letters

BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Ralph G. Bird
Senior Vice President - Nuclear

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

BECo 88-126
August 18, 1988

License DPR-35
Docket 50-293

REVISED INFORMATION REGARDING PILGRIM STATION
SAFETY ENHANCEMENT PROGRAM

Dear Sir:

Enclosed is a description of a revised design for the Direct Torus Vent System (DTVS) that was described in the "Report on Pilgrim Station Safety Enhancements" dated July 1, 1987 and transmitted to the NRC with Mr. Bird's letter (BECo 87-111) to Mr. Varga dated July 8, 1987. This revision supersedes in its entirety the Section 3.2 included in the July 1, 1987 report.

On March 7, 1988 Boston Edison Company (BECo) personnel met with Dr. Murley, Mr. Russell, and Dr. Thadani and provided a tour of SEP modifications and an informal presentation of the quantification of competing risks associated with venting the containment and conclusions drawn from these results. This presentation provided BECo the opportunity to respond to questions posed under Item 1 Section 3.2 - "Installation of A Direct Torus Vent System (DTVS)" in Mr. Varga's letter to Mr. Bird of August 21, 1987 "Initial Assessment of Pilgrim Safety Enhancement Program". The material presented was made available to the resident inspector and was included as Attachment II in NRC Inspection Report #88-12, dated May 31, 1988.

As you are aware from plant inspections we have installed the DTVS piping and portions of related control wiring. Currently, the DTVS is isolated from the Standby Gas Treatment System (SBGTS) by blind flanges installed in place of Valve AO-5025 and the DTVS rupture disk. This configuration was inspected by NRR in the performance of a technical review which focused on System, Mechanical Design and Structural Design issues. The review took place on March 2-3, 1988 as documented in NRC Inspection Report #88-07, dated May 6, 1988 and determined the installation configuration to be acceptable. We now plan to remove these blind flanges and proceed with installation of Valve AO-5025 and the DTVS rupture disk. We conclude the valve and rupture disk provide equivalent physical isolation of the DTVS piping from the SBGTS and appropriately ensure the operational integrity of the SBGTS under design basis accident conditions. Following completion of this work, we will perform a local leak rate test to verify that Valve AO-5025 is acceptably leak tight using the same method previously utilized in testing the blind flange. We also plan to complete all remaining electrical work on the DTVS in accordance with the revised design.

A001

NRR

On the basis of the revised Section 3.2, we conclude that the DTVS design as described in the enclosure does not require any change to the Technical Specifications and that we can proceed with installation without prior NRC approval.

Please feel free to contact me or Mr. J. E. Howard, of my staff at (617) 849-8900 if you have any questions pertaining to the design details of the DTVS.

R. G. Bird

Attachment: Section 3.2 Revision 1 "Installation Of A Direct Torus Vent System (DTVS)"

JEH/amm/2282

cc: Mr. D. McDonald, Project Manager
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
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Senior NRC Resident Inspector
Pilgrim Nuclear Power Station

Attachment to BECo Letter 88-126

Section 3.2 Revision 1 "Installation Of A Direct Torus Vent System (DTVS)"

pages 14, 15, 16, 17, 18, 19, 19A, 19B

3.2 INSTALLATION OF A DIRECT TORUS VENT SYSTEM (DTVS)

3.2.1 Objective of Design Change

This design change provides the ability for direct venting of the torus to the main stack. Containment venting is one core damage prevention strategy utilized in the BWR Owners Group Emergency Procedure Guidelines (EPGs) as previously approved by the NRC and is required in plant-specific Emergency Operating Procedures (EOPs). The torus vent line connecting the torus to the main stack will provide an alternate vent path for implementing EOP requirements and represents a significant improvement relative to existing plant vent capability. For 56 psi saturated steam conditions in the torus, apporoximately 1% decay heat can be vented.

3.2.2 Design Change Description

This design change (Figure 3.2-1) provides a direct vent path from the torus to the main stack bypassing the Standby Gas Treatment System (SBGTS). The bypass is an 8" line whose upstream end is connected to the pipe between primary containment isolation valves AO-5042 A & B. The downstream end of the bypass is connected to the 20" main stack line downstream of SBGTS valves AON-108 and AON-112. An 8" butterfly valve (AO-5025), which can be remotely operated from the main control room, is added downstream of 8" valve AO-5042B. This valve acts as the primary containment outboard isolation valve for the direct torus vent line and will conform to NRC requirements for sealed closed isolation valves as defined in NUREG 0800 SRP 6.2.4. The new pipe is ASME III Class 2 up to and inclusive of valve AO-5025. Test connections are provided upstream and downstream of AO-5025.

The design change replaces the existing AC solenoid valve for AO-5042B with a DC solenoid valve (powered from essential 125 volt DC) to ensure operability without dependence on AC power. The new isolation valve, AO-5025, is also provided with a DC solenoid powered from the redundant 125 volt DC source. Both of these valves are normally closed and fail closed on loss of electrical and pneumatic power. One inch nitrogen lines are added to provide nitrogen to valves AO-5042B and AO-5025. New valve AO-5025 will be controlled by a remote manual key-locked control switch. During normal operation, power to the AO-5025 DC solenoid will also be disabled by removal of fuses in the wiring to the solenoid valve. This satisfies NUREG 0800 SRP 6.2.4, Containment Isolation System acceptance criteria for a sealed closed barrier. An additional fuse will be installed and remain in place to power valve status indication for AO-5025 in the main control room.

NUREG 0800, SRP 6.2.4, Item II.6.F allows the use of sealed closed barriers in place of automatic isolation valves. Sealed closed barriers include blind flanges and sealed closed isolation valves which may be closed remote-manual valves. SRP 6.2.4 calls for administrative control to assure that sealed closed isolation valves cannot be inadvertently opened. This includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator.

Consistent with SRP 6.2.4, valve AO-5025 will be a sealed closed remote manual valve under administrative control to assure that it cannot be inadvertently opened. Administrative control will be maintained by a key-locked remote manual control switch and a fuse removed to prevent power from being supplied to the valve operator. In accordance with NUREG 0737, Item II.E.4.2.7 Position 6, AO-5025 will be sealed closed and verified as such at least every 31 days.

A 20" pipe will replace the existing 20" diameter duct between SBGTS valves AON-108, AON-112 and the existing 20" pipe to the main stack. The existing 20" diameter duct downstream of AO-5042A is shortened to allow fitup of the new vent line branch connection.

A rupture disk will be included in the 8" piping downstream of valve AO-5025. The rupture disk will provide a second leakage barrier. The rupture disk is designed to open below containment design pressure, but will be intact up to pressures equal to or greater than those which cause an automatic containment isolation during any accident conditions.

The two Primary Containment Isolation Valves (PCIVs) AO-5042B and AO-5025 are placed in series with the rupture disk. No single operator error in valve operation can activate the DTVS. The rupture disk has a rupture pressure above the automatic containment high pressure trip point. Thus, the inboard PCIV (AO-5042B) will receive an automatic isolation prior to disk rupture. The inboard PCIV (AO-5042B) requires physical electrical jumper installation to open at primary containment pressure above the automatic high pressure trip point.

Valve AO-5025 will be closed whenever primary containment integrity is required and DC power to its solenoid control valve will be disconnected. Indication of valve position will be provided in the main control room even with the valve power removed. Use of the direct torus vent will be in accordance with approved EPG requirements and controlled by EOPs in the same manner as other existing containment vent paths. Prior to opening the vent valves the SBGT system will be shutdown and valves AON-108 and AON-112 (the outlet of SBGT) placed in a closed position.

New 8" vent pipe (8"-HBB-44), including valve AO-5025 is safety related. Vent piping downstream of AO-5025, including SBGTS discharge piping to main stack, is also safety related. All safety related piping will be supported as Class I. Nitrogen piping is non-safety related and will be supported as Class II/I.

The interpretation of the Class II/I designation through this report is given below:

All Class II items which have the potential to degrade the integrity of a Class I item are analyzed. Such Class II items do not require dependable mechanical or electrical functionality during SSE, only that all of the following conditions prevail:

1. The Class II items create no missiles which impact unprotected Class I items safety functions.

2. The Class II item does not deform in a way which would degrade a Class I item.

3. If the Class II item fails, then the Class I item is protected against the full impact of all missiles generated by the assumed failure of Class II items.

All electrical portions of this design are safety related except for the indicating lights on the MIMIC panel C904, the tie-ins to the annunciator, and interface with the plant computer.

3.2.3 Design Change Evaluation

3.2.3.1 Systems/Components Affected

Containment Atmospheric Control System (CACS)

The torus purge exhaust line inboard isolation valve AO-5042B and the associated 8" pipe are the components of the CACS affected by the design modification. With incorporation of the subject modification, the CACS will depend on both essential AC (for valve AO-5042A) and essential DC (for AO-5042B) to perform its purging function.

The new 8" torus vent line will be connected to existing 8" CACS piping between valves AO-5042B and AO-5042A.

Standby Gas Treatment System (SBGTS)

The SBGTS fan outlet valves (AON-108 and AON-112), ductwork from these valves to the 20" line leading to the main stack, and the 20" line leading to the main stack are the components of this system affected by the proposed change.

Valve AON-108 is normally closed, fail-open. Valve AON-112 is normally closed, fail-closed, and these valves are provided with essential DC power and local safety related air supplies.

Primary Containment Isolation System (PCIS)

Valve AO-5042B is affected by the change from AC to DC power for the solenoid and by replacement of the existing air supply with nitrogen. The addition of containment outboard isolation valve (AO-5025) will not affect the PCIS.

Primary Containment System (PCS)

Valve AO-5025 acts as the primary containment outboard isolation valve for the direct torus vent line and will conform to NRC requirements for sealed closed isolation valves as defined in NUREG 0800 SRP 6.2.4.

3.2.3.2 Safety Functions of Affected Systems/Components

Containment Atmospheric Control System

This system has the safety function of reducing the possibility of an energy release within the primary containment from a Hydrogen-Oxygen reaction following a postulated LOCA combined with degraded Core Standby Cooling System.

Standby Gas Treatment System

This system filters exhaust air from the reactor building and discharges the processed air to the main stack. The system filters particulates and iodines from the exhaust stream in order to reduce the level of airborne contamination released to the environs via the main stack. The SBGTS can also filter exhaust air from the drywell and the suppression pool.

Primary Containment Isolation System

This system provides timely protection against the onset and consequences of design basis accidents involving the gross release of radioactive materials from the primary containment by initiating automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed pre-selected operational limits.

Primary Containment System

The primary containment system, in conjunction with other safeguard features, limits the release of fission products in the event of a postulated design basis accident so that offsite doses do not exceed the guideline values of 10 CFR 100.

3.2.3.3 Potential Effects on Safety Functions

Containment Atmospheric Control System, Standby Gas Treatment System, Primary Containment Isolation System and Primary Containment System

The improvements change the AO-5042B solenoid control from AC to DC enabling it to open (from its normally closed position) with no dependence on AC power availability. The existing air supply to AO-5042B is being replaced by nitrogen.

Ductwork at the outlet of the SBGTS is replaced with pipe and the new vent line is connected to the 20" line at the outlet of the SBGTS.

Addition of a new 8" vent line with containment isolation valve AO-5025 off the existing torus vent line could introduce a flow path under design basis conditions that could vent the containment directly to the stack bypassing the SBGTS.

3.2.3.4 Analysis of Effects on Safety Functions

An analysis of the effects on the safety functions of CACS, SBGTS, PCIS and PCS for the installation of the direct torus vent is described as follows:

The change from AC to DC control and the replacements of air with nitrogen on AO-5042B does not adversely affect the ability to open AO-5042B when the containment is being purged, or to isolate under accident conditions.

The modifications to the ductwork and 20" line leading to the main stack do not affect the design basis safety function of any of the safety related systems.

During normal plant operations, the CACS and the SBGTS do not use the torus 20" purge and vent line to perform their safety functions. The containment isolation valves are in their normally closed position, thus maintaining primary containment boundary integrity.

There are no adverse affects on the primary containment system by the addition of the DTVS. Valve AO-5025 will conform to NRC criteria for sealed closed isolation valves as defined in NUREG 0800 SRP 6.2.4 and will not affect design basis accidents. Use of the DTVS will be in accordance with the containment venting provisions of EPGs as approved by the NRC and controlled by EOPs in the same manner as other existing containment vent paths. The effects on the torus of the new 8" piping and AO-5025 have been evaluated for Mark I program loadings, using ASME BPVC Section III criteria. The remaining piping including the rupture disk was evaluated using ANSI B31.1 requirements.

During plant startup and shutdown (non-emergency condition) when the purge and vent line is in use, valve AO-5025 remains closed. In addition, the rupture disk downstream of valve AO-5025 will provide a second positive means of preventing leakage and prevent direct release up to the stack during containment purge and vent at plant startup or shutdown.

During containment high pressure conditions, the torus main exhaust line is automatically isolated by the PCIS. There is no change to the existing primary containment isolation system function for AO-5042A or AO-5042B. The sealed closed position of valve AO-5025 and the additional assurance added by the rupture disk downstream will prevent any inadvertent discharge up the stack for all design basis accident conditions.

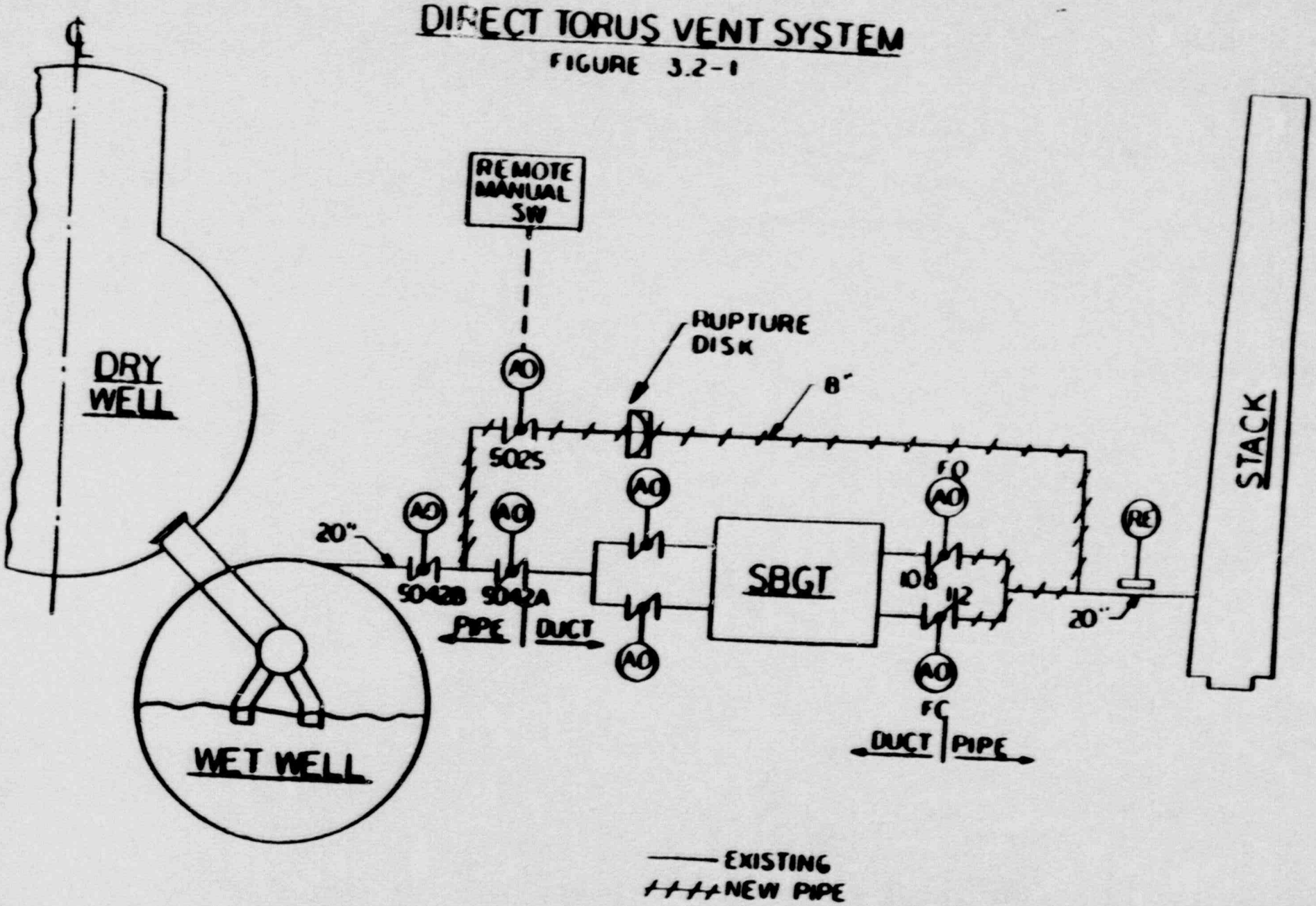
3.2.3.5 <u>Design Change Evaluation Summary Conclusions</u>

Installation of the DTVS does not adversely affect the safety functions of the CACS, SBGTS, PCIS or the integrity of primary containment or any other safety related systems.

Use of the DTVS will be in accordance with the containment venting provisions of EPGs as approved by the NRC and controlled by EOPs in the same manner as other existing containment vent paths. The DTVS provides an improved containment venting capability for decay heat removal which reduces potential onsite and offsite impacts relative to the existing containment venting capability.

DIRECT TORUS VENT SYSTEM

FIGURE 3.2-1



ENCLOSURE I

06 DEC 1989

Docket Nos. 50-220
50-410

Niagara Mohawk Power Corporation
ATTN: Mr. Lawrence Burkhardt, III
Executive Vice President
Nuclear Operations
301 Plainfield Road
Syracuse, New York 13212

Gentlemen:

Subject: Public Comments on the Restart Action Plan/Restart of Nine Mile Point Unit 1 Received from the Public - NRC Responses to Those Comments

This letter addresses the August 23, 1989, public meeting held in Oswego, New York, to receive comments on the Restart Action Plan (RAP). Based on review of the transcript of the comments from this meeting and of written comments received by mail, the NRC staff, through its Restart Assessment Panel, has concluded that no changes are needed to the RAP. This conclusion was previously noted in the NRC approval of the RAP in a letter dated September 29, 1989.

By transmittal to the Local Public Document Room, the attachment to this letter provides responses to the specific comments. A response has been provided to each comment. The responses are grouped according to those comments directly related to the RAP (29 comments) and those comments generally related to Nine Mile Point (20 comments).

Although the comments are provided for your information, comment 35 raised a concern regarding the turning off of radiation monitors during discharges and unusual events. This allegation was forwarded for your review in a separate letter dated October 12, 1989.

We appreciate your cooperation.

Sincerely,

William F. Kane, Director
Division of Reactor Projects

ttachment:
As stated

OFFICIAL RECORD COPY NINE MILE POINT COMMENTS - 0001.0.0
11/01/89

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IE01

cc w/Attachment:
C. Mangan, Senior Vice President
W. Hansen, Manager, Corporate Quality Assurance
C. Beckham, Manager, Nuclear Quality Assurance Operations
J. Perry, Vice President, Quality Assurance
J. Willis, General Station Superintendent
C. Terry, Vice President, Nuclear Engineering and Licensing
K. Dahlberg, Unit 1 Station Superintendent
R. Randall, Unit 1 Superintendent, Operations
R. Smith, Unit 2 Superintendent, Operations
R. Abbott, Unit 2 Station Superintendent
G. Wilson, Senior Attorney
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J. Warden, New York Consumer Protection Branch
Director, Power Division, Department of Public Service, State of New York
State of New York, Department of Law
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New York

bcc w/Attachment:
Region I Docket Room (with concurrences)
J. Wiggins, DRP
G. Meyer, DRP
D. Limroth, DRP
R. Barkley, DRP
S. Horwitz, PAO
M. Miller, SLO
W. Cook, SRI - Nine Mile
R. Temps, RI - Nine Mile
R. Laura, RI - Nine Mile
J. Dyer, EDO
R. Capra, NRR
M. Slosson, NRR
R. Martin, NRR

ATTACHMENT

Public Comments on the Restart Action Plan

1. Comment: The financial viability of Niagara Mohawk Power Corporation should be evaluated as part of the Restart Action Plan (RAP).

   Response: The financial viability of Niagara Mohawk is evaluated by the NRC with respect to the ability of Niagara Mohawk to safely operate Nine Mile Point. During the public Commission status briefing on August 2, 1989, the Commission questioned Niagara Mohawk with regard to its financial viability and its ability to safely operate Nine Mile Point. Niagara Mohawk assured the Commission that sufficient funds are available to safely operate the units. In summary, the Commission will continue to monitor Niagara Mohawk's activities to ensure that financial concerns do not interfere with the safe operation of the plants, but disagrees that the issue requires inclusion in the RAP.

2. Comment: The torus should be repaired prior to restart.

   Response: Both the Niagara Mohawk Power Corporation (NMPC) and the NRC have known about the corrosion of the wall of the torus at Nine Mile Point Unit 1. The thinning of the torus wall will be addressed in the resolution of RAP Specific Issue No. 7 prior to restart. NMPC will be required to perform repairs to the torus prior to restart if the torus wall has corroded beyond the minimum requirements of the American Society of Mechanical Engineers (ASME) Code governing the adequacy of this vessel. Thus, this issue is already addressed in the RAP.

3. Comment: NDE surveillance of the torus should be conducted by people other than NMPC that do not have a vested interest in keeping the plant running.

   Response: The NDE surveillance of the torus wall at Nine Mile Point Unit 1 is presently conducted by a contractor for Niagara Mohawk as well as by Niagara Mohawk employees. The results of the NDE examinations are independently reviewed by the Niagara Mohawk Quality Assurance Department and by the NRC (upon the submittal for the resolution of RAP Specific Issue No. 7). In addition, Niagara Mohawk's Inservice Inspection (ISI) program, particularly their NDE methods, was reviewed in detail by NRC Region I during Inspection 50-220/88-81 and determined to be acceptable.

This inspection included independent wall thickness measurements performed by the NRC. Finally, as an alternative confirmation of the structural strength of the torus wall, Niagara Mohawk will be performing an Integrated Leak Rate Test (ILRT) of the containment prior to the restart of the plant to satisfy the requirements of 10 CFR 50 Appendix J.

In summary, based on this degree of oversight of the NDE surveillance on the torus wall, Niagara Mohawk's present NDE surveillance program on the torus is structured in accordance with all applicable NRC requirements and the NRC does not agree that reviews by additional parties are warranted.

4. Comment: Long-term management improvement plans should be implemented and goals achieved before restart of Unit 1.

Response: Niagara Mohawk has formulated a Nuclear Improvement Program (NIP) to improve their overall level of performance both prior to and following the restart of Nine Mile Point Unit 1. The NIP embodies a number of plans for long-term improvements in management and worker performance and goals to measure the success of that program. Niagara Mohawk has determined, and the NRC has agreed that these improvements are important but are not essential for the safe operation of the facility following restart, provided that the management issues in the RAP are resolved prior to restart. Thus, the issue is properly addressed in the NIP.

5. Comments: RAP Section 2, page II-9 - Part of the RAP long-term strategy for improving management performance is to identify training and development programs for manager, supervisor and employee intrapersonal and management skills, etc. Shouldn't these be implemented prior to restart?

Response: Above response to Comment No. 4 applies

6. Comment: NMPC should prove that the RAP works before restart is authorized.

Response: The NRC's approach toward approving restart of Nine Mile Point Unit 1 has been structured in the following manner:

i. Niagara Mohawk developed and implemented a Restart Action Plan (RAP).

ii. The NRC reviewed the RAP for approval.

iii. Niagara Mohawk conducted a self-assessment of their readiness for restart.

iv. NRC will review the quality and conclusions of the self-assessment.

v. NRC will conduct an Integrated Assessment Team Inspection of the Nine Mile Point 1 organization and its ability to safely operate the facility.

vi. A decision will be made by the Region I Regional Administrator regarding restart.

The NRC believes that the process, as outlined, is sufficient to determine the effectiveness of the RAP and the ability of Niagara Mohawk to safely operate the facility and is consistent with the intent of this public comment. This process has been followed successfully at other problem plants.

7. Comment: Nine Mile Point is unsafe based on the recent disclosures of the magnitude of the waste spill at Unit 1 plus the close to 50 percent failure rate of the operators at Unit 2.

Response: The waste spill in the radwaste storage building at NMP Unit 1 was investigated by an Augmented Inspection Team (AIT) from the NRC in August, 1989. The results of that team inspection indicate that the spill, while an example of poor operating practice, was never a hazard to the public health and safety and was properly surveyed and controlled by the NMPC health physics department to ensure that the spill did not pose a threat to plant workers. NMPC has also included the cleanup activity in the NIP. The specific management problems that led to this condition are addressed in RAP Items 1-5.

The high failure rate of the licensed reactor operators at Unit 2 during the recent requalification examination was attributed to weaknesses in the requalification training program. However, sufficient operators had successfully passed an NRC-administered requalification examination to allow the plant to continue to operate with shift crews augmented by extra personnel to compensate for the deficiencies noted until remediation could be completed. NMPC has also implemented a remedial training program to improve requalification training in the deficient areas noted. The examination results were indicative of significant

weaknesses in the requalification program which required correction but not of the scope or depth that would pose significant public health and safety concerns. The adequacy of the operator training program at Unit 1 will be thoroughly reviewed prior to restart to resolve RAP Specific Issue Nos. 2 and 3.

In summary, the NRC disagrees with this comment.

8. Comment: NRC reporting requirements/problems should be defined in the RAP.

Response: NMPC does not have a chronic history of failing to make NRC required notifications and reports. Based on past history, this issue is not required in the RAP as a specific technical issue requiring resolution. NRC reporting requirements are presently outlined in 10 CFR 50.72 and 50.73 as well as the NMP Unit 1 TS.

9. Comment: There should be a public hearing/evidentiary proceeding prior to restart.

Response: As specified in the Code of Federal Regulations Title 10 Part 2.206, "Any person may file a request to institute a proceeding pursuant to §2.02 to modify, suspend, or revoke a license, or for such other action as may be proper." Requests made in accordance with 10 CFR 2.206 will be reviewed and evaluated by the NRC. Typically, a proceeding would not be held prior to the restart of a unit such as Nine Mile Point Unit 1, because the evaluation of the restart of the unit is not a change to the operating license. Thus, the NRC disagrees with this comment.

10. Comment: Provide the details of the original problems leading to the shutdown, including the delays in that shutdown and all subsequent problems prolonging that shutdown to the LPDR. Also, ensure that the last several SALPs are in the Local Public Document Document Room (LPDR).

Response: The LPDR routinely receives copies of the publicly distributed NRC correspondence, including SALPs. The NRC will review the availability of the public documents relative to the shutdown in the LPDR and ensure the appropriate ones exist there.

11. Comment: The RAP should clearly state what verifications will be done in the presence of NRC inspectors.

Response: As a condition of the license for Nine Mile Point Unit 1, Niagara Mohawk is subject to unannounced NRC inspections to determine its compliance with the terms of the license and to assess its ability to safely operate the facility. However, the NRC has not found it necessary in this case that certain licensed activities be conducted in the presence of NRC inspectors. Although the NRC will not require NMPC to conduct any of the activities outlined in the RAP in the presence of NRC inspectors, we will review those actions deemed necessary to assure that the concerns addressed by the RAP are adequately resolved.

12. Comment: With reference to RAP Specific Issue No. 2, why didn't Niagara Mohawk establish responsibility and accountability for the maintenance of operator licenses 20 years ago?

Response: Establishing responsibility and accountability for the maintenance of a program is a fundamental principle of management that should have been implemented since the inception of the program. The reasons for NMPC's failure to implement these management principles in the past in this area is probably attributable to deficiencies in NMPC management in the Operations area, the extensive number of changes that have occurred in the operator training area since the TMI accident, and poor cooperation between the Operations and Training Departments. The RAP and CALs 88-13 and 88-17 were generated precisely because of questions like these by the NRC and to resolve these types of management deficiencies.

13. Comment: The spent fuel pool should be fixed prior to restart.

Response: The spent fuel pool has experienced minor leakage in recent months due to an apparent perforation in the stainless steel pool liner at a location yet to be identified. NMPC has proposed a plan of action to identify and resolve this problem subsequent to restart. Given the size of the leak which has been observed, as well as the design of the spent fuel pool, no threat to the public health and safety exists by delaying the repairs to the pool liner until after restart. Thus, the NRC disagrees with this comment and feels that the issue is properly addressed in the RAP (Specific Issue 15) as an issue which can be resolved after restart.

14. Comment: The radwaste spill should be cleaned up prior to restart.

Response: NMPC has in-place a plan for the decontamination and cleanup of the radwaste storage room spill at Unit 1. The plan involves the use of a robotic arm to remotely decontaminate the room. Given that the spill is confined to a small, abandoned area of the plant and that there is no indication that the radioactive contamination in the room is leaking to the environment, no threat to the public or worker health and safety exists. NMPC is scheduled to complete the decontamination of the room by March, 1990. Therefore, the NRC disagrees with this comment and does not consider the decontamination of the room a restart issue. The issue will be properly addressed in the NIP.

15. Comment: A determination should be made as to how much pressure the containment at Nine Mile Point Unit 1 can withstand.

Response: The Nine Mile Point 1 containment system consists of an upper section called the drywell and a lower portion called the suppression chamber. Per the Unit 1 Final Safety Analysis Report, the drywell is designed to withstand a peak 62 psig internal pressure. The suppression chamber is designed to withstand a peak 35 psig internal pressure. Prior to restart, the leak tightness and structural integrity of the containment will be tested during a containment integrated leak rate test. Any further analyses are not considered a restart issue.

16. Comment: Was NMP Unit 1 considered a safe plant by the NRC in December 1987 and why, all of the sudden, did the NRC ask Niagara Mohawk's management to come up the a plan to resolve their problems?

Response: Yes. Nine Mile Point was considered to be a plant which met the conditions of its license in December 1987. Had the NRC believed at anytime prior to December 1987 that the operation posed a threat to public health and safety, the Unit would have been immediately ordered to shutdown. Following the shutdown of the Unit in December 1987 due to technical problems, several other problems were identified in the areas of operator requalification training, control of commercial grade parts, fire barrier penetrations, and the operator's understanding and use of emergency operating procedures. As a result, on July 24, 1988, the NRC issued a Confirmatory Action Letter which documented Niagara Mohawk's commitment not to restart Unit 1 until corrective actions have been completed and the agreement of the NRC's Region I Regional Administrator was obtained. The actions required include a root cause assessment of why management has not been effective in recognizing and remedying problems, preparation of a restart action plan, and submission of a written report relative to readiness for restart.

17. Comment: Why did the NRC raise a safety concern regarding the scram discharge volume on June 24, 1983, but take until December 1987 to take action to resolve the issue?

Response: During a routine shutdown of Browns Ferry Unit No. 3 on June 28, 1980, 76 of 185 control rods failed to fully insert in response to a manual scram from approximately 30% power. All rods were subsequently inserted within 15 minutes and no reactor damage or hazard to the public occurred. Following an in-depth review of Boiling Water Reactor Control Systems, short and long-term corrective measures were identified. Short-term corrective measures were implemented by IE Bulletin 80-17 and an Order concerning these measures was issued to Nine Mile Point 1 on January 9, 1981 and modified March 31, 1981. A Confirmatory Order, dated June 24, 1983, was issued to Niagara Mohawk concerning long-term corrective measures. Due to inspection priorities, the NRC staff did not inspect the scram discharge volume design for Nine Mile Point Unit 1 to determine compliance with the June 24, 1983 Order until November 1987. As a result of the inspection, two areas of deviation from the Order and the Generic Safety Evaluation, dated December 1, 1980, for the scram discharge volume were identified. These deviations had been identified by Niagara Mohawk in a January 30, 1981 letter, but NMPC failed to obtain prior NRC approval for the deviations as required by the Order.

The NRC feels this issue is properly addressed in the RAP. By letter dated October 12, 1988, the NRC staff transmitted its safety evaluation with respect to this issue and concluded that operation with the system for another fuel cycle poses no undue risk to the public. The staff has evaluated Niagara Mohawk's proposed testing program for the scram discharge volume and determined it to be acceptable.

18. Comment: Niagara Monawk is not competent to operate atomic power plants based upon NRC findings/fines over the last six years.

Response: A purpose of the RAP and CAL 88-17 was to assure improvements in Niagara Mohawk's management of its nuclear facilities. While their performance in the last several years has raised NRC concerns, the NRC staff believes that Niagara Mohawk is capable of improving its operation and that a properly scoped management improvement plan that is effectively implemented is an appropriate means by which this can be accomplished. The NRC staff must conclude that the necessary improvements have been made prior to restart of Unit 1.

19. Comment: Safety concerns regarding cracking in the core spray spargers.

Response: In accordance with IE Bulletin 80-13, Niagara Mohawk visually examined the Unit 1 core spray spargers and associated piping during the 1981 refueling outage. As documented in their May 13, 1981 letter to the NRC, two cracks in one location were identified. The cracks were evaluated and determined to be insignificant. No corrective actions were required. NMPC has continued to examine the sparger each refueling outage since 1981. The most recent examination, completed during the current outage, indicated that the crack length remains within the tolerance of the original crack length. Therefore, no corrective actions were required. The licensee will continue to inspect the core spray spargers and associated internal piping during future refueling outages. Thus, this issue does not impact restart or the RAP.

20. Comment: Will failed plant equipment be replaced with like-in-kind parts or by components that are not part of the original design and may be untested?

Response: All safety-related components which NMPC replaces due to failure or as part of a preventive maintenance program are purchased either as a safety-related component (fabricated and tested under an approved quality assurance program) or as a commercial grade product later subject to a quality assurance program designed to qualify the component for safety-related applications. Given the age of the facility and the declining number of suppliers of safety-related components in this country, NMPC may not necessarily replace failed components with identical replacements specified by the original design. Instead, alternative components may be used which are of different design, but have been demonstrated to be suitable for safety-related applications and are capable of performing the function of the original part. This practice is commonplace in the nuclear industry and, if properly administered, is acceptable to the NRC. Thus, this issue is not a problem and is inappropriate for conclusion in the RAP.

21. Comment: NMP Unit 1 should not be restarted because it will add to the radwaste problem. What will happen to the radwaste generated?

Response: The low level radioactive waste that is generated by Nine Mile Point Unit 1 will be shipped to one of the three currently licensed burial sites in the United States. However, it is the ultimate responsibility of New York State to find and develop an alternative disposal site in the near future under the provisions of the Low Level Waste Policy Amendments Act of 1985. All

high level radioactive wastes will be stored onsite, either in the spent fuel pool or if later constructed, in a future onsite storage facility, until the federal government develops a high-level waste repository as required by the Nuclear Waste Policy Act of 1982.

At present, there is no prohibition against the operation of any nuclear power plant in this country due to questions regarding the ultimate resolution of the problem of disposing of high level or low level radioactive wastes. Therefore, the NRC disagrees that this issue should be included in the RAP and should be a basis for not restarting Unit 1.

22. Comment: The RAP does not require NMPC to conduct health studies to determine the long-term health effects of Nine Mile Point. These studies should be completed prior to restart of NMP Unit 1.

Response: As stipulated by the terms of its license, NMPC has been required to conduct environmental monitoring of the area around the plant site since prior to the licensing of Unit 1 to determine if there has been any accumulation of radioactive material in the environment or excessive radioactive emissions from the plant. There is no indication from the NMPC environmental monitoring program that any substantial potential danger to the public from normal plant emissions exists, particularly at levels that would suggest that a health study in the area is warranted. Based upon the results of the environmental monitoring program to date, the NRC does not consider this activity appropriate for the Restart Action Plan.

23. Comment: The permanent solution to the radwaste disposal problem should be included as an item in the RAP.

Response: The response to question 21 applies.

24. Comment: Cracks in the concrete walls in various parts of the plant should be fixed prior to restart. NMPC does not have a good handle on the wall cracking. Will the plant survive a seismic event due to all the building cracks?

Response: NMPC has observed cracks in the masonary walls of several buildings in the plant over time. The location of these cracks, as well as NMPC's plan of action to monitor and analyze the significance of these cracks, is documented under Specific Issue No. 15 of the RAP. The cause of the cracks in the walls appeared to be predominately due to either shrinkage during curing or tensile stress experienced due to temperature fluctuations

(concrete by nature has little structural strength in tension [its structural strength in tension is supplied by the reinforcing rods in the concrete] and thus has a tendency to crack in tension and during curing). NMPC has completed an extensive analysis and repair project on all masonary walls at Unit 1 correct all of the cracks noted in the plant walls. These repairs ensured that the masonary walls meet their original seismic design criteria. Therefore, the NRC does not consider this issue a safety problem and resolution of the issue is properly addressed in the RAP.

25. Comment: What is the status of the SPDS system at NMP Unit 1?

Response: The Safety Parameter Display System was declared fully operational at Nine Mile Point Unit 1 in June 1986 and is therefore not a restart issue.

26. Comment: New York State should police NMP Unit 1 activities.

Response: By the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974 as amended, the United State Congress gave the NRC statutory authority over the regulation of nuclear facilities. As such it is the NRC's responsibility to: 1) license the construction and operation of nuclear reactors and other nuclear facilities, 2) license the possession, use, processing, handling and disposal of nuclear material, 3) develop and implement rules and regulations that govern licensed nuclear activities, inspect licensed facilities and activities, investigate nuclear incidents and allegations concerning any matter regulated by the NRC, 4) conduct public hearings on matters of nuclear and radiological safety, environmental concern, common defense and security and antitrust laws and, 5) develop effective working relationships with the states regarding regulation of nuclear material. Further, to ensure adequate communication and cooperation between the NRC and the states, the State Liaison Officer program was established in 1976. If New York State so desires, the NRC will consider state proposals to enter into instruments of cooperation for state participation in NRC inspection activities if the state's program has provisions to ensure close cooperation with the NRC.

In summary, New York State is welcome to enter into an agreement with the NRC to provide an oversight role, but the NRC has preemptive federal authority in the licensing of nuclear plants.

27. Comment: The public has the perception that there isn't the slightest chance that the plant will stay closed.

Response: The Nine Mile Point Unit 1 facility was licensed under the terms of 10 CFR 50, which derives its statutory basis from the Atomic Energy Act of 1954, as amended. Since the facility remains licensed under Part 50, the emphasis of NMPC to date has been to correct the identified management deficiencies and to attempt to restore the plant to service. As a result, the NRC's actions to date have been oriented toward determining whether the facility can be safely returned to service and is based on NMPC having corrected the management and technical deficiencies identified. That is the purpose of our review of the Restart Action Plan. If NMPC can not correct the deficiencies noted, then the facility will remain closed.

28. Comment: Monthly public meetings should be held before restart to discuss the concerns of area citizens.

Response: The public meeting held on August 23, 1989, was an initiative on the part of the NRC to involve the public in the restart process at NMP Unit 1 and to gain their comments. Additional meetings of this type would result only if there are fundamental changes in the NMPC Restart Action Plan. The public is free to contact NRC directly to raise safety concerns.

29. Comment: Don't restart Unit 1.

Response: This comment was received from at least thirteen individuals during the public meeting. The response to question 27 applies.

Public Comments on Issues Not Related to the RAP

30. Comment: Spent fuel pool storage/acceptability of dry cask storage.

Response: The NRC is proposing to amend its regulations to provide, as directed by the Nuclear Waste Policy Act of 1982, for the storage of spent fuel at the sites of power reactors, to the maximum extent practicable, without the need for additional site-specific approvals. Holders of power reactor operating licenses would be permitted to store spent fuel in casks approved by NRC under a general license. The proposed rule contains criteria for obtaining an NRC Certificate of Compliance for spent fuel storage casks. The notice of proposed rule making, published in the Federal Register on May 5, 1989, solicited public comments by June 19, 1989. The NRC is currently evaluating the public comments. In any event, this issue does not impact the restart of the facility.

31. Comment: The public doesn't trust Niagara Mohawk.

Response: This comment was made by four members of the public at the public meeting and represents an issue for NMPC to address. This issue is not applicable to restart.

32. Comment: The public doesn't trust the NRC

Response: This comment was made by three members of the public at the public meeting and is not related to the Restart Action Plan.

33. Comment: Why is Unit 2 allowed to operate with the same management as Unit 1?

Response: While NMP Unit 2 has been categorized by NRC as a plant requiring close NRC scrutiny, the staff and management of Unit 2, which is separate for each unit below the site superintendent level, have performed better than the staff and management at Unit 1. Most notably, the operators at Unit 2 have clearly displayed a much more positive and responsive attitude than Unit 1 operators. In addition, management has displayed the ability to operate Unit 2 effectively in spite of the fact that it is significantly more complex than Unit 1 and has a staff with significantly less operational experience with the facility than Unit 1. Thus, while Nine Mile Point Unit 2 remains under close NRC scrutiny, NRC senior management has determined that NMPC can safely operate Unit 2 in spite of the noted deficiencies at Unit 1.

34. Comment: Does the radwaste building spill represent a threat to the public? Why wasn't the NRC notified of the spill? Why didn't the NRC find this problem?

Response: The results of the NRC Augmented Inspection Team (AIT) conducted at Nine Mile Point indicate that the radwaste spill does not pose a threat to the public health and safety.

The NRC inspectors failed to identify the spill previously because the problem occurred in the sub-basement of an abandoned portion of the radwaste building and was confined to one room. The room was properly designated and marked as a locked high radiation area. The NRC inspection program does not require all locked high radiation areas in the plant to be examined internally by the NRC inspectors due to the unnecessary radiation exposure which would occur to the inspector. Thus, the fact that the NRC did not know of the existence of this radioactivity contaminated room was attributable largely to its location and the fact that Niagara Mohawk did not report the incident versus a failure to conduct a required portion of the NRC inspection program. No RAP actions as a result of this incident are necessary.

35. Comment: It is alleged that radiation process monitors are turned off when radioactive discharges to the environment are made by NMPC.

Response: This allegation has been entered into the NRC allegation tracking system for follow-up and will be resolved prior to restart.

36. Comment: Who is responsible for ensuring that Nine Mile Point and FitzPatrick have adequate emergency plans?

Response: It is the responsibility of the NRC to ensure that the licensees of Nine Mile Point and FitzPatrick have adequate emergency plans for the staff and employees onsite and the proper notification of authorities offsite. Verification of the adequacy of offsite emergency planning is the responsibility of the Federal Emergency Management Agency (FEMA) acting as an agent for the NRC. This delegation of responsibility was established by Presidential Directive.

37. Comment: The background radiation levels in an area near the plant are elevated above normal.

Response: The NRC maintains a independent radiation monitoring program in the area around the plant. The results of this radiation monitoring program are published quarterly in NUREG - 0837. Review of the results of that monitoring program do not indicate that there are elevated background radiation levels in any area around the plant.

38. Comment: Does the NRC fee schedule pose a potential conflict of interest?

Response: The annual funding of the NRC is established and provided by Congress. The funds received through the fee schedule are deposited directly to the Federal Treasury. While the NRC fee schedule does recover a significant portion of its operating costs, the schedule is not related to the NRC's budget and does not present a conflict with the agency's role as a regulator of the utility.

39. Comment: Allowing radioactive waste to be categorized as being below a level of regulatory concern, and thus capable of being disposed of as normal trash, constitutes a public health and safety concern.

Response: The NRC was mandated by the Low Level Radioactive Waste Policy Amendments Act of 1985 to establish a regulatory limit for radioactive waste which does not pose a threat to the public health and safety and thus can be disposed of by normal methods. The NRC has yet to establish such a regulatory limit. When that radioactivity level is decided upon, it will be the subject of extensive review and rulemaking to determine that the limit does not pose a threat to the public health and safety.

40. Comments: What has been done regarding the concerns raised by Douglas Ellison? Why did the NRC pay Mr. Ellison $11,000 for information regarding Niagara Mohawk?

Response: The allegations initially raised by Douglas Ellison were reviewed by a special NRC team in inspection 50-220/86-17. As a result of new allegations, two additional inspections were recently conducted at Nine Mile Point to resolve the concerns raised. The results of those two inspections, one regarding the technical issues and one regarding allegations of potential employee harassment and intimidation, are documented in Inspection Reports 50-220/89-16 and 89-21, respectively. While some of the concerns were partially substantiated by the inspection team, no issues which affect the public health and safety were identified.

The matter of Mr. Ellison being paid by the NRC to provide information regarding Niagara Mohawk is the subject of an ongoing investigation and thus can not be discussed in detail at this time. Completion of this investigation does not impact the Restart Action Plan or restart of the facility.

41. Comment: Allegation concerning an unknown NRC employee/individual impersonating an NRC employee.

Response: The NRC can not take any further actions with regard to the individual described due to the lack of information provided. Although the individual stated that he was an NRC employee, it is not clear that the individual was in fact employed by the NRC. However, from the description of the actions of the individual from the person making this statement, the individual's actions would have been totally unacceptable behavior for an NRC employee.

42. Comment: I'm opposed to the operation of any nuclear power plant in this country.

Response: Congress has decreed through the Atomic Energy Act of 1954 that atomic energy is beneficial to this country relative to the inherent risks that the energy source poses. Thus, existing atomic power plants are permitted to operate in this country, and new power stations may be constructed and licensed to operate. It is the NRC's responsibility to ensure that the risks from this energy source are minimized.

43. Comment: (Received in Writing) - The audience at the public meeting was not representative of the public and is biased against the plant.

Response: The purpose of the public meeting was to receive comments on the RAP, both positive and negative. The meeting was not intended to provide a forum to determine the level of public support for or against the restart of Unit 1.

44. Comment: (Received in writing) - Do oil and coal-fired power plants that spew out the makings of acid in have to hold hearings before they start-up?

Response: The NRC is not responsible for the regulation of power plants other than those powered by nuclear energy. Those facilities are subject to regulation by other governmental agencies (i.e., Environmental Protection Agency, Occupational Safety and Health Administration, etc.) and may be subject to licensing actions, up to and including a public hearing, in the event that they do not comply with the agencies' regulations or if requested by another party in accordance with the procedural rules of the agency.

45. Comment: (Received in writing) - I have heard that there is a "collar" around the reactor at Nine Mile Point One that is so corroded that if the plant was turned on full power, it would blow up.

Response: After discussions between several members of the NRC staff to determine what the individual could have meant by the "collar" around the plant, it was determined that the person must have been referring to the torus. The written response provided to the individual stated that neither the torus nor the reactor can "blow up" regardless of the operating status of the plant. The NRC assured the individual that NMPC is taking actions to resolve the torus corrosion problem and that the NRC will closely follow their actions to determine that the problem does not pose a threat to public health and safety. Further, the individual was informed that Niagara Mohawk will be conducting a pressurization test of the torus prior to restart of the facility to ensure that it will withstand the maximum postulated pressure generated during a design basis reactor accident. Based upon these facts, the NRC feels that the individual's concerns are unsubstantiated.

46. Comment: (Received in writing) - What was the source(s) of the elevated cesium and strontium concentrations noted in local milk supplies during the late 1970's?

Response: The NRC conducted a review of the elevated Cs-137 and I-131 levels noted in milk samples in the area near the plant in 1981. The review was conducted in response to concerns regarding this issue which were raised by the Sierra Club. The results of this review indicated that the average levels of Cs-137 in milk near the site were not consistently higher than the rest of the State. The NRC's assessment at that time was that the source Cs-137 and I-131 concentrations in milk in the area could not be precisely determined. The source of the contamination could have been attributable to either reactor effluents or fallout from weapons tests, most particularly the Chinese atmospheric weapons tests in the late 1970's. Regardless of the source, the observed radiation levels constituted only a small fraction of the radiation dose received from natural background radiation. That small dose would also have been below regulatory limits even if the assumption was made that all the observed radioactivity came from effluents from Nine Mile Point and FitzPatrick.

47. Comment: (Received in writing) - Why is there a higher incidence of learning disabilities in area children born during the late 1970's?

Response: The NRC has no knowledge of any studies which show that the incidence of learning disabilities in area children is higher than normal at any time in history. Based on research studies into the effect of radiation on the incidence of mental retardation in children, no measurable effect on the incidence of learning disabilities in area children should be observable until radiation doses to the public from the operation nuclear facilities were at least three orders of magnitude above NRC limits.

48. Comment: (Received in writing) - A detailed study of the Lake Ontario bottom sediments or area wetlands should be conducted by NMPC to determine if sediments and biota were affected by the radwaste room flooding event.

Response: As mentioned in the response to earlier comments, NMPC is required to conduct an environmental sampling program as a requirement of their license. That sampling program involves collecting samples of sediments, biota, and fish from Lake Ontario as well as the surrounding land area. The results of the sampling program confirm that NMPC is operating the facility in accordance with the NRC's radioactive waste release limits as codified in 10 CFR 20.

49. Comment: (Unrelated to Nine Mile Point Unit 1 Restart) - Don't site a low level waste repository in the area.

Response: The siting of a low level waste repository is currently the responsibility of the State of New York.

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

ACTION

EDO Principal Correspondence Control

FROM:

DUE: ~~02/23/90~~ ~~02/29/90~~ 3/21/90

EDO CONTROL: 0005139
DOC DT: 02/06/90
FINAL REPLY:

Rep. Frank Horton

Revised Based upon revised response Guidance received from CA on 02/22/90.
R. Capra 2/22/90

TO:

Chairman Carr

FOR SIGNATURE OF: ** GRN **

CRC NO: 90-0128

Executive Director

DESC:

ENCLOSES LETTER FROM TOM WALSH, RETIRE NINE MILE ONE RE REFUELING OF THE NINE MILE ONE AND THE SAFETY OF THE MARK I CONTAINMENT DESIGN

ROUTING:

Russell, RI

DATE: 02/12/90

ASSIGNED TO: NRR
CONTACT: Murley

SPECIAL INSTRUCTIONS OR REMARKS:

NRR RECEIVED: FEB. 12, 1990
ACTION: DRPR:VARGA

NRR ROUTING: MURLEY/SNIEZEK
PARTLOW
MIRAGLIA
CRUTCHFIELD
GILLESPIE
MOSSBURG

ACTION
DUE TO NRR DIRECTOR'S OFFICE
BY 2/26/90

26

OFFICE OF THE SECRETARY
CORRESPONDENCE CONTROL TICKET

PAPER NUMBER: CRC-90-0128 LOGGING DATE: Feb 9 90

ACTION OFFICE: EDO

AUTHOR: Frank Horton--Const Ref
AFFILIATION: U.S. HOUSE OF REPRESENTATIVES

LETTER DATE: Feb 6 90 FILE CODE: ID&R-5 Nine Mile 1

SUBJECT: Refueling of the Nine Mile One and the safety of the Mark I containment design

ACTION: Direct Reply

DISTRIBUTION: OCA to Ack, DSB

SPECIAL HANDLING: None

NOTES: Tom Walsh

DATE DUE: Feb 27 90

SIGNATURE: DATE SIGNED:
AFFILIATION:

Rec'd Off. EDO
Date 2-12-90
Time 8 45 A

16-9MILE1

Retire Nine Mile One

A broad based coalition of citizens concerned about the safety of the Nine Mile One Nuclear Facility

January 8, 1989

Congressman Frank Horton
2108 Rayburn Building
Washington, D.C. 20515

Dear Congressman Horton:

RETIRE NINE MILE ONE has concerns about the safety of the Nine Mile One Nuclear Facilty that we hope you will give your immediate attention.

Nine Mile One was constructed with a Mark I containment designed by General Electric. Since 1975, General Electric has reported that there is a 90% chance the Mark I containment will fail in the event of a core accident. The Nuclear Regulatory Commission has described the Mark I containments as "virtually certain" to fail in the event of a core accident. Nine Mile One is also showing serious signs of deterioration due to age, particularly in the torrus, that makes its safe operation a considerable risk.

The deteriorated condition of the facility coupled with a containment that is "virtually certain" to fail make Nine Mile One vulnerable to a Chernobyl type of accident that would result in considerable loss of life and force the abandonment of hundreds of square miles in Central New York. When these consequences are considered, discussions of the probablity of such an event occurring seem moot.

We are asking that you respond in the following ways:

1. We are requesting that you contact the Nuclear Regulatory Commission and ask that the refueling of the plant be stopped. The NRC has not given Niagara Mohawk permission to restart Nine Mile One. Certainly the refueling of the plant prior to completion of the NRC's assessment of its worthiness to operate is premature.

P.O. Box 563 - University Station - Syracuse, New York 13210

2. We are requesting that you sponsor a public hearing in your district on this urgent matter. There is a need for the citizens of the district to be provided a forum in which to express their concerns about the facility. A hearing would also provide an opportunity to create a record of expert testimony on this matter.

3. We would like to request that you make yourself available for a briefing by the Union of Concerned Scientists who have monitored the Mark One facilities throughout the country and whose thorough and professional research brought many of us to the opinion we now hold about Nine Mile One.

We thank you for your attention to this urgent matter and look forward to your response.

Sincerely,

RETIRE NINE MILE ONE

by Tom Walsh, co-chair.
(315) 446-0435