50-219

State of New Jersey

Christine Todd Whitman
Governor

Department of Environmental Protection

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Rex - Action RJ

August 31, 1995

Mr. Phillip McKee, Director
Project Directorate I-4
Division of Reactor Projects
Office of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. McKee:

Subject: Oyster Creek Generating Station

Each year my staff reviews the status of resolution of nuclear industry safety issues as they relate to nuclear power plants in New Jersey. The NRC publishes this status annually in NUREG-1435. I've attached an internal memorandum which summarizes our understanding of the status of these issues as of August 30, 1995. I am requesting that your staff review this status and advise me as to its accuracy so that we have a mutual understanding of the implementation status of these issues for the Oyster Creek generating Station. By copy of this letter to GPU Nuclear I am requesting that they review this document and provide me with feedback in a similar manner.

Thank you for your attention to this matter. If you have any questions, please contact me.

Sincerely,

Kent Tosch / D.Z.

Kent W. Tosch, Manager
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M E M O R A N D U M

TO: Dennis J. Zannoni, Supervising Nuclear Engineer, Nuclear Engineering Section (NES)

FROM: Thomas J. Kolesnik, Nuclear Engineer, Nuclear Engineering Section (NES)

SUBJECT: Review of NUREG 1435, Supplement 4,"Status of Safety Issues at Licensed Power Plants", December 1994

INTRODUCTION

Each year the NES looks closely at open safety issues at nuclear power plants operating in New Jersey to determine the status of the safety issues, identify any problems, and determine an appropriate NES response, if warranted. In December 1994, the NRC issued NUREG-1435, Supplement 4,"Status of Safety Issues at Licensed Power Plants". Supplement 4 documents the implementation and verification status of four different categories of safety issues at 107 U.S. nuclear power plants. The four categories are (1) Three Mile Island (TMI)Action Plan Requirements, (2) Unresolved Safety Issues (USIs), (3) Generic Safety Issues (GSIs), and (4) Other Multiplant Actions (MPAs). The NES reviewed the NUREG Supplement to investigate the status of outstanding safety issues at Oyster Creek, Hope Creek, and Salem Units 1 and 2.

BACKGROUND

TMI Action Plan requirements, USIs, GSIs and other MPAs are all types of generic issues that originated from increased technical understanding of the safety of nuclear power plants. The NRC previously published three volumes of this NUREG series. Volume 1, published in March 1991, discussed the status of TMI Action Plan requirements. Volume 2, published in May 1991, identified the implementation and verification status of actions associated with USIs. Volume 3, published in June 1991, detailed the status of GSI actions. The first annual NUREG report, Supplement 1, combined these volumes into a single report and provided updated information as of September 30, 1991. Supplement 1 was published in December 1991. The second annual NUREG report, Supplement 2, provided updated information as of September 30, 1992. In addition, Supplement 2 also provided the status of licensee implementation and NRC verification of MPA issues not related to TMI Action Plan requirements, USIS, or GSIs. Supplement 3 provided status as of September 30, 1993. This fourth annual NUREG report, Supplement 4, provides updated information as of September 30, 1994, for all TMI, USI, GSI, and MPA issues. Subsequent volumes will continue to be published annually to document the progress of implementation and verification of these items.

DISCUSSION

1. Three Mile Island (TMI) Action Plan Requirements

TMI Action Plan requirements were instituted following the accident at TMI Unit 2 in 1979. The NRC staff developed the Action Plan, NUREG-0660, to provide a comprehensive and integrated plan to improve safety at nuclear power plants. More than 99 percent of the TMI Action Plan items have been implemented at 107 licensed plants. Of the 12,678 applicable items, 12,667 are complete and 11 remain open as of September 30, 1994, the closure date of NUREG 1435, Supplement 4. Of the 470 applicable items, none remain open at nuclear power plants operating in New Jersey.

The following table provides a numerical summary of applicable TMI Action Plan items for nuclear power plants operating in New Jersey.

TMI ACTION PLAN REQUIREMENTS

PLANT	APPLICABLE	COMPLETED	REMAINING
Oyster Creek	107	107	0
Salem I	116	116	0
Salem II	127	127	0
Hope Creek	120	120	0

2. Unresolved Safety Issues (USIs)

Unresolved Safety Issues are matters affecting a number of nuclear power plants that pose important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants affected as identified in NUREG-0510. Approximately 94 percent of the USI items have been implemented at the 107 licensed nuclear plants. Of the 1,751 applicable items, 1,646 are complete and 105 remain open as of September 30, 1994, the closure date of NUREG 1435, Supplement 4. Of the 68 applicable items, three currently remain open at nuclear power plants operating in New Jersey.

The following table provides a numerical summary of applicable USI items for nuclear power plants operating in New Jersey.

UNRESOLVED SAFETY ISSUES

PLANT	APPLICABLE	COMPLETED	REMAINING AS OF 9/30/94	CURRENT	USIs REMAINING AS OF 9/30/94
Oyster Creek	18	16	2	1	(1)Seismic Qualification of Equipment (2)Safety Implications of Control Systems
Salem I	16	15	1	1	(1)Seismic Qualification of Equipment
Salem II	17	16	1	1	(1)Seismic Qualification of Equipment
Hope Creek	17	17	0	0	

The following text provides background information on those USI items relevant to nuclear power plants operating in New Jersey which were not yet implemented as of September 30, 1994.

(1) USI (A-46) was issued in February, 1987 as Generic Letter 87-02, 'Verification of Seismic Adequacy of Mechanical and Electrical

Equipment in Operating Reactors'. As a result of the technical resolution of this USI, the NRC has concluded that the seismic adequacy of certain equipment in operating nuclear power plants must be reviewed against seismic criteria not in use when these plants were licensed. Direct application of current seismic criteria to older plants could require extensive, and probably impracticable, modification of these facilities. An alternative resolution of this problem makes use of earthquake experience data supplemented by test results to verify the seismic capability of equipment below specified earthquake motion bounds. In the NRC staff's judgment, this approach is the most reasonable and cost-effective means of ensuring that the purpose of General Design Criterion 2 (10 CFR Part 50 Appendix A) is met for these plants.

Because affected plants are being asked to carry out this evaluation against criteria not used to establish the design basis of the facility, this resolution is a backfit under 10 CFR 50.109. Seismic verification may be accomplished generically. Utilities participating in a generic program should identify the utility group and the schedule established for completion of the effort.

To address this issue, GPU Nuclear has completed walkdowns with a report to the NRC expected to be submitted by March 29, 1996.

PSE&G submitted a report to the NRC on May 22, 1995. NRC technical staff is scheduled to finish their review of the report by December 31, 1995 with a response letter to PSE&G to be issued by February 1996.

(2) USI(A-47) was issued in September 1989 as Generic Letter 89-19, 'Safety Implications of Control Systems in LWR Nuclear power Plants.' The primary focus of the resolution of this USI is to provide a mechanism to trip the main feedwater pumps when a high water level occurs in the reactor vessel or steam gemnerators. In 1990, the NRC staff reviewed the licensees' responses to Generic Letter 89-19 and determined that:

- o The Westinghouse pressurized-water reactor (PWR) utilities have implemented the letter recommendations in their designs. However, some facilities did not have the technical specifications for operability of instrumentation.

- o The boiling water reactor (BWR) utilities, except Oyster Creek and Big Rock Point, and the Combustion Engineering (CE) PWR utilities, except Palo Verde, concluded that the modifications recommended in the letter are not cost effective. The NRC staff agreed with the BWR Owners Group justification that no further modifications to the existing reactor vessel overfill protection system are necessary. The NRC staff continued its review of the CE Owners Group justification as it relates to assumptions on steam generator tube rupture probability.

- o NRC staff review of the Babcock & Wilcox (B&W) plants continued on a plant-specific basis because the B&W Owners Group had not taken a position on this issue.

GPU Nuclear implemented the requirements of USI (A47) during the 15R Outage, which commenced on September 28, 1994.

3. Generic Safety Issues (GSIs)

Generic Safety Issues are safety concerns that affect the design, construction, or operation of all, several, or a class of nuclear power plants and may have the potential for safety improvemnets at such plants. Approximately 98 percent of the items associated with GSIs have been implemented at licensed plants. Of the 2,579 GSI items, 2,516 are complete and 63 remain open as of September 30, 1994, the closure date of NUREG 1435, Supplement 4. Of the 96 applicable items, none currently remain open at nuclear power plants operating in New Jersey.

The following table provides a numerical summary of applicable GSI items for nuclear power plants operating in New Jersey.

GENERIC SAFETY ISSUES

PLANT	APPLICABLE	COMPLETED	REMAINING AS OF 9/30/94	CURRENT	GSIs REMAINING AS OF 9/30/94
Oyster Creek	22	21	1	0	1) Improved Accident Monitoring
Salem I	26	26	0	0	
Salem II	26	26	0	0	
Hope Creek	22	22	0	0	

The following text provides background information on those GSI items relevant to nuclear power plants operating in New Jersey which were not yet implemented as of September 30, 1994.

(1) GSI (67.3.3) addresses conformance with NRC Regulatory Guide 1.97, 'Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.' The NRC staff issued Generic Letter 82-33 in December 1982 to request licensees to submit schedules and details of their plans to implement the provisions of Regulatory Guide 1.97, Revision 2. The licensee responses to this letter prompted the NRC

staff to issue confirmatory orders in 1985. Because the industry has taken exception to and appealed some of the provisions of Regulatory Guide 1.97, Revision 2, implementation is incomplete at many plants. The issues of Category 1 neutron flux monitoring at BWRs and Category 2 instrumentation to monitor containment sump water temperature at PWRs have been resolved. The issue of Category 1 neutron flux monitoring at three PWRs is still under review.

GPU Nuclear implemented the requirements of GSI (67.3.3) during the 15R Outage, which commenced on September 28, 1994.

4. Other Multiplant Actions(MPAs)

A Multiplant Action item originates from industry experience, new regulations/requirements, or from resolution of generic issues resulting in the issuance of a generic communication requiring action by the licensees. TMI items, USIs and GSIs are all MPAs, however, there are also other MPAs that do not fit into one of these categories. These other MPAs may be either required or voluntary. For the purposes of this NUREG, only non-voluntary MPA issues are considered. Approximately 89 percent of the MPAs items have been implemented at licensed plants operating in the U.S. Of the 6,951 applicable items, 6,307 have been implemented and 644 remain open as of September 30, 1994, the closure date of NUREG 1435, Supplement 4. Of the 258 applicable items, 15 currently remain open at nuclear power plants operating in New Jersey.

The following table provides a numerical summary of applicable MPA items relevant to nuclear power plants operating in New Jersey.

OTHER MULTIPLANT ACTIONS

PLANT	APPLICABLE	COMPLETED	REMAINING AS OF 9/30/94	CURRENT	MPAs REMAINING AS OF 9/30/94
Oyster Creek	78	74	7	4	See A,B, C,E,G,H, and I.
Salem I	89	85	6	4	See D,E, F,I J, and K.
Salem II	54	50	6	3	See D,E, F,I J, and K.
Hope Creek	37	33	5	4	See D,E, G,I, and J.

The following text provides background information on those MPA items relevant to nuclear power plants operating in New Jersey which were not yet implemented as of September 30, 1994.

A. Thermo-Lag 330 Fire Barrier System (NRC Bulletin 92-01, NRC Generic Letter 92-08)

To help ensure that each licensee defines and implements acceptable solutions to the Thermo-Lag fire barrier issues, in December 1993, the NRC staff sent a request for additional information (RAI) in accordance with 10 CFR 50.54(f) to each licensee relying on the Nuclear Energy Institute (NEI, formerly NUMARC) program. Licensees' responses were received and evaluated February through April of 1994. In May 1994, the staff sent the Commission the results of the review in SECY-94-128. The number of operating units that have yet to resolve the Thermo-Lag issues has been reduced from 83 to 59. In addition, the staff presented a list of proposed options to resolve the Thermo-Lag issue, recommended a course of action, and asked the Commission for guidance during a briefing held on May 20, 1994. The Commission approved the option which specified compliance with existing NRC requirements and permits plantspecific exemptions where justified.

NEI issued an application guide on July 7, 1994, which provided the results of industry testing. In September, 1994, the NRC staff issued follow-up letters to the Thermo-Lag 50.54(f) letters sent to licensees in December 1993. The purpose of the follow-up letter is 1) to state the NRC's course of action to resolve the Thermo-Lag issue, 2) to provide exemption guidance, and 3) to request completion of licensees' responses to the original RAI (50.54(f)) for licensees that deferred answering until the completion of industry testing and the issuance of the new application guide.

GPU Nuclear is currently participating in the NEI sponsored chemical test program intended to establish similarity between the materials previously tested as part of the NEI fire barrier test program and the materials installed at OCNGS. GPU Nuclear has submitted Thermo-Lag samples for analysis consistent with NEI's instructions. The Oyster Creek Thermo-lag samples were taken from four locations for chemical analysis. Each sample and location was identified and individually packaged to prevent cross contamination. The results show that the chemical composition of the Oyster Creek samples are consistent with those materials tested as part of the NEI fire barrier test program based upon a preliminary evaluation of the results as reported to GPU Nuclear by NUCON. GPU Nuclear will submit a report addressing these results along with the industry-wide results being coordinated by NEI by November 1, 1995. This date is based upon NEI's expectation of releasing industry-wide sampling results by the end of July 1995. GPU Nuclear will submit a report to the NRC by December 31, 1995, which will document the Thermo-lag barrier evaluations currently being performed. This report will provide the detailed basis and methodology that has been applied for each barrier configuration to determine the important barrier parameters and to verify that the parameters if in-plant configurations are representative of the parameters of the NEI fire endurance test specimens.

Although the Salem and Hope Creek Stations do not have Thermo-lag as a fire barrier, PSE&G is monitoring the NEI research and any other documentation regarding fire barriers to prepare for the need to replace the current materials if deficiencies are identified.

This issue is very important and is being monitored closely. The BNE continues to have discussions with the utilities about interim fire protection measures while final corrective action is being pursued by the NRC, utilities and associated groups.

B. Debris Plugging of Emergency Core Cooling System (ECCS) Suction Strainers (NRC Bulletin 93-02)

On May 11, 1993, the NRC staff issued NRC Bulletin 93-02, 'Debris Plugging of Emergency Core Cooling System Suction Strainers.' The bulletin discussed several instances of ECCS suction blockage due to filtering action of fibrous material, and required certain compensatory actions by the licensees. Subsequently, a detailed study of a representative BWR 4 with a Mark I containment was contracted by the staff. The preliminary results of this study showed that insulation debris generated during a LOCA could potentially be transported to the ECCS suction strainers and clog them. This preliminary finding combined with Perry and Barseback events led the staff to issue NRC Bulletin 93-02, Supplement 1. The NRC issued the bulletin supplement on February 18, 1994. The proposes of the bulletin supplement were:

(1) To inform BWR licensees and PWR licensees about the vulnerability of ECCS suction strainers in BWRs and containment sumps in PWRs to clogging during the recirculation phase of a loss-of-coolant accident (LOCA).

(2) To request that BWR licensees take the appropriate actions to ensure reliability of the ECCS in view of the information discussed in this bulletin supplement regarding the vulnerability of the ECCS strainers to clogging.

(3) To require that BWR licensees report to the NRC whether and to what extent the requested actions will be taken and to notify the NRC when actions associated with this bulletin supplement are complete.

GPU Nuclear implemented the requirements of this MPA item in February 1995.

C. Reactor Vessel Water Level Instrumentation in BWRs (NRC Bulletin 93-03, NRC Generic Letter 92-04)

The NRC staff issued Generic Letter 92-04, 'Resolution on the

Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)" on August 19, 1992, to alert licensees of BWRS to errors related to instrumentation accuracy in water level instrumentation and to the results of the staff's review of the BWROG's generic analysis of these errors. The staff also requested addressees to (1) determine the impact of these errors on automatic safety system response, operator short- and long-term actions, and emergency operating procedures at their facilities; (2) take short- and long-term corrective actions; and (3) submit a report that includes the results of their determinations, a discussion of their short- and long-term actions, and the schedule for completion of their long-term programs.

All addressees responded by September 28, 1992. Most licensees requested deferral of the long-term corrective actions to allow BWROG to complete testing and analysis of the BWR water level instrumentation. The staff accepted delays in the implementation of long-term corrective actions pending BWROG development of plant and/or procedure modifications by July 1993.

Following an event at the WNP-2 plant in January 1993, additional analyses by the BWROG revealed additional safety concerns related to RPV water level instrumentation at low pressures following normal depressurizations. This led the staff to issue NRC Bulletin 93-03 on May 28, 1993, requesting additional actions by the addressees. These actions were to (1) provide additional procedures and training to address the new concerns and (2) implement, at the first cold shutdown after July 30, 1993, hardware modifications to ensure high functional reliability of the RPV water level instrumentation for long-term operation. Addressees have provided their responses.

All BWR licensees have completed the short-term compensatory actions requested in the bulletin. At this time, licensees for 35 of the 36 affected plants have either completed installation of hardware modifications or are currently shutdown and will install the hardware modifications prior to restart. The remaining licensee has committed to complete modifications during the next cold shutdown, and is currently scheduled to shutdown for refueling in January 1995. Based upon the licensees' safety analyses and the short-term compensatory measures provided in response to GL 92-04 and BL-93-03 that have been taken, the NRC staff considers these schedules to be acceptable.

GPU Nuclear implemented the requirements of this MPA item during the 15R Outage, which commenced on September 28, 1994.

D. Individual Plant Examination for Severe Accident Vulnerability (NRC Generic Letter 88-20)

The NRC staff issued Generic Letter 88-20, 'Individual Plant Examination for Severe Accident Vulnerability,' on November 23, 1988, to request addressees to perform an individual plant examination (IPE) of their plant-specific internal event severe accidents and report the results of their analysis.

The NRC staff issued Supplement 1 to Generic Letter 88-20, 'Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54," on August 29, 1989, requiring licensees to submit an IPE to identify plant-specific severe accident vulnerabilities using probabilistic risk analysis methodology.

Several licensees have delayed submittal of IPEs from 2 to 18 months. The staff has issued 18 evaluation reports documenting the results of the IPE review. As of June 30, 1994, all IPEs had been submitted. The NRC staff expects to complete all reviews by mid-1996.

PSE&G submitted its IPE for Salem 1 & 2 to the NRC on July 30, 1993. A Request for Additional Information was issued to the licensee on April 25, 1995 which included 38 NRC questions. PSE&G responded to the NRC's request on August 8, 1995. The NRC staff is currently in the process of reviewing the PSE&G response. PSE&G submitted its IPE for Hope Creek Station in the Fall of 1994 and the NRC staff is currently in the process of reviewing the submittal. PSE&G's IPE is based upon a Probabilistic Risk Assessment (PRA) approach (Level I and II). The PRA approach was chosen because it provides models which can be revised to incorporate design, operational, procedural and phenomenological updates.

E. Safety Related Motor-Operated Valve (MOV) Testing and Surveillance (NRC Generic Letter 89-10)

The NRC staff issued Generic Letter 89-10 to inform licensees of problems concerning the operability of safety-related motor-operated valves, and request addressees to (1) establish programs to demonstrate the operability of these valves and to ensure continued operability over the life of the plant, (2) provide a commitment to establish such a program and complete the demonstration of operability within the timeframe specified in GL 89-10, and (3) report completion of the demonstration phase of their programs. The subject matter of this generic letter is related to that of NRC Bulletin 85-03, 'Motor-Operated Valve Common Mode Failures During Plant Transient Due to Improper Switch Setting,' and its supplements.

Supplements 1 through 4 of Generic Letter 89-10 were addressed in Supplement 2 of NUREG-1435. Supplement 5,'Inaccuracy of Motor-Operated Valve Diagnostic Equipment,' was issued on June 28, 1993, to request licensees to reexamine their MOV programs in light of new information on MOV diagnostic equipment inaccuracies and to identify measures taken or planned to account for uncertainties in valve thrust. Licensees were also to determine the schedule necessary to satisfy the supplement.

All power reactor licensees have submitted responses to this supplement. Licensees indicated that many MOVs had the potential for underthrusting or overthrusting as a result of the higher than expected inaccuracy of MOV diagnostic equipment. Consequently, some licensees reported that MOVs have been retested, adjusted, or modified to resolve the concerns regarding the accuracy of MOV diagnostic equipment. Licensees have incorporated the margins necessary to account for diagnostic equipment inaccuracies in their MOV testing program and accounted for schedule adjustment.

Supplement 6, 'Information on Schedule and Grouping, and Staff Responses to Additional Public Questions,' was issued on March 8, 1994. This supplement includes discussions of schedule extensions, grouping of MOVS, and use of probabilistic risk assessment in the implementation of Generic Letter 89-10.

GPU Nuclear advised the NRC in letter dated December 28,1989, of the schedule to complete the GL actions. The stated Oyster Creek schedule was to complete the MOV program within three refueling outages of June 28, 1989 (Generic Letter 89-10, issued June 28, 1989, requested the recommended actions to be completed within five years or three refueling outages, whichever was later), and this was further discussed at NRC/GPU Nuclear management meetings held on February 21, 1992 and October 20, 1994. The third refueling outage after June 28, 1989 for Oyster Creek was the 15R Outage, which was completed on December 16, 1994. Modifications to Generic Letter 89-10 MOVs were completed during 15R Outage which significantly increased the available margins for Reactor Water Cleanup System valves and Isolation Condenser System valves, as described in the GPU Nuclear letter dated November 7, 1994. Modifications to four Core Spray System valves to eliminate susceptibility to pressure locking, and to two Containment Spray System valves to provide larger valve actuator motors, were also completed during 15R Outage. Remaining static and dynamic valve tests for Generic Letter 89-10 program valves were also completed during 15R Outage. As discussed at the October 20, 1994 GPU Nuclear/NRC meeting, the evaluation of best available industry test data to further support assumed valve factors was ongoing.

PSE&G claims to have met the requirements of Generic Letter 89-10 for both the Salem Units and Hope Creek Station, however, NRC followup inspections to confirm that claim are expected to be completed by December 1995.

F. Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies (NRC Generic Letter 93-04

The NRC issued Generic Letter 93-04 on June 21, 1993, (1) to notify addressees aboutsingle failure vulnerability within the Westinghouse solid-state rod control system that could cause an inadvertent withdrawal of control rods in a sequence resulting in a power distribution not considered in the design basis analyses and (2) to require that all action addressees provide the NRC with information describing their plant-specific findings related to this issue and actions taken. The letter was addressed to all holders of operating licenses or construction permits for Westinghouse-designed nuclear power reactors except Haddam Neck.

The letter requested licensees to assess within 45 days if their licensing basis is still satisfied with regard to single failure in the rod control system in light of the Salem event. If the licensing basis is not satisfied, the NRC requested the licensee to provide an assessment of the impact and describe any compensatory short term actions taken within 45 days and provide a plan and schedule for long term resolution within 90 days. On July 26, 1993, the NRC granted relief to the schedules to extend the licensing basis assessment portion of the 45-day response to the 90-day response, in response to a request from the Westinghouse Owners Group (WOG).

All licensees have provided their 45-day and 90-day responses. The responses generally indicated that they were following the WOG's efforts and would update their responses upon completion of the WOG's efforts. The long-term WOG efforts include a current timing order change to the rod control system and a new surveillance test. The timing change does not affect normal rod motion, but prevents asymmetric rod withdrawal should Salem-type failure occur. The modification was successfully tested at the Ginna Station on April 15, 1994.

The NRC issued a letter to PSE&G on February 9, 1995 accepting the licensees corrective actions as satisfactory which included surveillance testing each refueling outage. Salem Unit 1 will implement the requirement during the current outage while Salem Unit 2 implemented the requirement on January 27, 1995.

G. Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors (NRC Generic Letter 94-03)

The NRC staff issued Generic Letter 94-03, 'Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors,' on July 25, 1994. The purpose of this letter is to request each owner of a boiling-water reactor (BWR) with core shroud to: (1) inspect the core shroud no later than the next scheduled refueling outage, and perform an appropriate evaluation and/or repair based on the results of the inspection; and (2) perform a safety analysis supporting continued operation of the facility until inspections are conducted. Each of the BWR owners is required to respond with their schedule for inspection of the core shroud and a safety analysis supporting continued operation of the facility within 30 days from the date of the letter. In addition, no later than 3 months prior to performing the core shroud inspections, each licensee must submit an inspection plan and plans for evaluation and/or repair of the core shroud based on the inspection results.

GPU Nuclear implemented a core shroud modification during the 15R Outage, which commenced on September 28, 1994. The NRC staff accepted the core shroud modification design and issued its Safety

Evaluation on November 25, 1994. GPU Nuclear was required to submit, within six months of plant restart, the results of corrosion testing and proposed revisions to the inservice inspection program that were requested in support of implementation of the core shroud modification.

PSE&G will conduct an inspection of the Hope Creek core shroud , based on the guidance contained in the Boiling Water Reactor Vessel Internals Program (BWRVIP) Inspection Criteria, during the next refueling outage, currently scheduled to begin on November 11, 1995. PSE&G's plans for inspection and for evaluation and/or repair of the core shroud based on the inspection results, will be submitted no later than three months prior to performing the inspection. Inspection results will be submitted within 30 days of completion of the core shroud inspection.

H. Results of NRC Testing of MOVs (NRC Generic Letter 89-10,Supp.3)

On June 5, 1990, the NRC staff issued Information Notice 90-40, 'Results of NRC-Sponsored Testing of Motor-Operated Valves (MOVS).' The tests revealed that the valves required more thrust for opening and closing under various differential pressure and flow conditions than would have been predicted from standard industry calculations using typical friction factors. Therefore, the staff issued Supplement 3 to GL 89-10 on October 25, 1990, which described required actions for licensees of BWRS. Licensees were required to provide (1) criteria reflecting operating experience and the latest test data that were applied in determining whether the deficiencies exist in the subject MOVS, (2) a list of the MOVs found to have deficiencies, and (3) a schedule for the necessary corrective action.

GPU Nuclear has completed the detailed evaluation of best available industry data which confirms the appropriate design valve factors applied to Oyster Creek Generic Letter 89-10 Program MOVs not practicable to dynamically test, or tested at lower than design basis differential pressure. The completed evaluation is based on an extensive industry search for data on valves similar to Oyster Creek and provides assurance that the design valve factors are conservative. The Oyster Creek MOV Program design basis reviews, calculations and testing have been completed. The Oyster Creek MOV Program Description is being updated to reflect the current status of the overall program.

I. IPE External Events (NRC Generic Letter 88-20, Supp. 4)

The NRC staff issued Supplement 4 to Generic Letter 88-20 on June 28, 1991, to initiate the IPE process for external events. Five categories of external events were specified, and licensees were required to submit to a schedule and methodology by December 26, 1991. Licensees were requested to submit the results of the individual plant examination of external events (IPEEE) within 3 years of the issuance date of Supplement 4, or no later than June

28, 1994. A copy of NUREG-1407, 'Procedural and Submittal Guidance for the IPEEE for Severe Accident Vulnerabilities,' was sent to each licensee with Supplement 4.

All licensee schedules for submittal of their IPEEEs have been received and reviewed. A common difficulty with a large number of the responses was the linkage of IPEEE to implementation of the USI A-46 resolution required by Generic Letter 87-02 to verify seismic adequacy of mechanical and electrical equipment. Supplement 4 to Generic Letter 88-20 encouraged licensees to combine the walkdown that would be required for the seismic portion of the IPEEE with the walkdown required by Generic Letter 87-02.

Supplement 1 to Generic LetterL 87-02 was issued on May 22, 1992, approving the seismic qualification utility group generic implementational procedure for USI A-46 implementation and starting the clock for both USI A-46 and the IPEEE. Licensees were advised that the latest acceptable date for IPEEE submittal would be June 1995. A second round of licensee responses indicated that the best effort by the industry will have the IPEEEs for 72 plants submitted by the target date of June 1995 but for the remaining 37 plants, the submittal dates will range from September 1995 to July 1997. As of August 1, 1994, ten reports had been submitted.

There are a small number of plants that have unique problems requiring a more customized response (1) because the licensee proposed alternative methods or failed to provide any method at all for its IPEEE or (2) because the licensee's plant was one of the eight singled out by the Eastern United States Seismic Hazards Program as needing further NRC staff evaluation.

GPU Nuclear is expected to submit its IPEEE Report to the NRC for review by December 20, 1995, however, no schedule for this MPA implementation is currently available.

By letter dated May 11, 1995, PSE&G documented proposed revisions to the schedules for submittal of the Salem and Hope Creek Generating Station Individual Plant Examination of External Events (IPEEE) Reports. The original commitment dates of May 1995 and February 1996 (for SGS and HCGS respectively were established, by letters dated December 19, 1991, and July 30, 1993. As discussed in the following paragraphs, present schedules support submittal of the IPEEE Reports for Salem and Hope Creek by January 30, 1996 and July 31, 1997 respectively. The revised submittal dates are consistent with NRC SECY 93-118, issued May 1993, which supported IPEEE extensions from the proposed dates of June 1995 until July 1997. This change was discussed with the NRC Project Managers for Salem and Hope Creek Stations.

Regarding the Salem IPEEE Report, delays associated with the receipt of the vendor's final draft report have had a significant impact on the schedule. The delay in the receipt of deliverables led to a series of discussions and meetings with the vendor's senior management. The result of these meetings was a preliminary report delivered on October 15, 1994. Comment resolution on this report resulted in additional delay in the delivery of the remaining reports. The final Salem report was due on March 15, 1995, but had not been received as of the date of this letter. As such, a schedule extension was necessary to allow for adequate review and comment resolution by PSE&G.

The IPEEE Report for Hope Creek is currently due on February 2, 1996, with the final vendor draft report expected by the Fall of 1995. Project reviewers and Independent Review Team (IRT) members are committed to the Salem Station work until late 1995. While limited parallel Hope Creek activities can be accomplished during this time period, an extension to the Salem IPEEE preparation schedule will necessarily affect the Hope Creek IPEEE. Therefore, PSE&G is also proposing to extend the Hope Creek IPEEE to July 31, 1997.

J. Loss of Fill-Oil in Transmitters Manufactured by Rosemount (NRC Bulletin 90-01)

On April 21, 1989, the NRC staff issued Information Notice 89-42, 'Failure of Rosemount Models 1153 and 1154 Transmitters,' to alert the industry of the loss of oil-fill problem. On March 9, 1990, the NRC staff issued Bulletin 90-01, 'Loss of Fill-Oil in Transmitters Manufactured by Rosemount,' to request the licensees to promptly identify and to take appropriate corrective action for Model 1153-Series B, Model 1153-Series D, and Model 1154 transmitters that may have the potential for leaking fill-oil. From mid 1990 through 1992, the NRC staff reviewed information from the (1) licensee responses to Bulletin 90-01, (2) data from related licensee event reports, (3) visits to the sites, (4) NUMARC Report 91-02, 'Summary Report of NUMARC Activities to Address Oil Loss in Rosemount Transmitters,' and (5) meetings with the industry. The NRC staff found a relationship between operating pressure and time-in-service that can be trended for use in identifying transmitters that are most likely to fail. The NRC staff has concluded that (1) the requested actions in Bulletin 90-01 were insufficient in that they did not provide the desired high functional reliability and (2) a supplemental bulletin would be needed for ensuring appropriate licensee corrective action to the loss of fill-oil problem.

Subsequently, on December 22, 1992, the NRC staff issued Bulletin 90-01, Supplement 1, 'Loss of Fill-Oil in Transmitters Manufactured by Rosemount,' to request new action from the licensees. Specifically, licensees were to provide information on specified models of the Rosemount transmitters manufactured before July 11, 1989, that are in use or may be used in the future. The information should detail the use of the devices in either a safety-related system or a system governed by the NRC's ATWS (anticipated transient without scram) requirements where normal operating pressure is greater than 500 pounds per square inch. Requested corrective action includes the replacement of the suspect transmitter or the use of an enhanced surveillance monitoring program until the transmitter reaches the time-in-service pressure criterion recommended by the vendor.

Responses to the supplemental bulletin were received from all applicable licensees, and as of September 30, 1994, responses for 65 reactor units had been reviewed and found to have completed the reporting requirements with the remaining responses to have a scheduled completion date of December 31, 1994. Additionally, the NRC staff has developed an action plan to address implementation of the recommendations resulting from the further assessment of Rosemount transmitter problems performed by the Rosemount Transmitter Review Group (RTRG) as documented in its October 12, 1993 report.

PSE&G implemented the requirements of this MPA for both Salem Units 1 & 2 and Hope Creek Station on April 27, 1994. The NRC completed its review of PSE&G's implementation on December 2, 1994.

K. Thermal Stress in Piping Connected to RCS (NRC Bulletin 88-08)

Following a circumferential crack in an unisolable section of emergency core cooling piping at Farley 2, the NRC issued Bulletin 88-08, 'Thermal Stresses in Piping Connected to Reactor Coolant Systems,' dated June 22, 1988. The Bulletin requested all licensees and applicants to take the following three actions: (1) Review their reactor coolant systems (RCSs) to identify any connected unisolable piping that could be subjected to temperature distributions that could result in unacceptable thermal stresses, (2) Examine unisolable piping sections for existing flaws, and (3) Implement a program to provide continuing assurance that unisolable sections will not be subject to stresses that could cause fatigue failure.

In summary, NRC Bulletin 88-08 was closed for those BWRs and PWRs whose responses to action item 3 above were consistent with the stated modification or monitoring alternatives. However, some plants replied that assurance for certain lines would be provided by inspection alone, when conducted as part of their inservice inspection program. The licensee responses for these plants were unacceptable without further justification, because inservice inspection was not identified by the bulletin as an acceptable alternative. The basis for this position is that the fundamental precept of the actions of the bulletin is to prevent the initiation of cracks in piping. Inservice inspection is not a technique that prevents the initiation of cracks. Rather, inservice inspection identifies cracks after they appear, and then a safety significance determination is made and corrective action is proposed. The NRC staff is reviewing the supplemental responses of licensees whose initial submittals contained insufficient information.

The NRC staff completed its review of PSE&G's submittal on August 15, 1995.The corrective actions included no hardware modifications just piping leakage testing on an ongoing basis.

CONCLUSION

After analyzing the information presented in NUREG-1435, Supplement 4, the NES has determined that significant progress in resolving the remaining safety issues has been made, especially during the last year. Since the September 30, 1994 closure date of this NUREG 1435 supplement, 9 MPA items have been resolved at nuclear power plants operating in New Jersey. Currently, there are 18 safety issue items remaining to be resolved at nuclear power plants operating in New Jersey (Oyster Creek - 5, Salem Unit 1 - 5, Salem Unit 2 - 4, Hope Creek - 4). Though concrete progressive action has been made on these open safety issues at the nuclear power plants operating in New Jersey, these issues, and the corresponding resolution of them, remain a high priority.

RECOMMENDATION

Submit this report to the respective utilities and the Nuclear Regulatory Commission for feedback on the accuracy of the information contained in the report before deciding on any BNE course of action.

c: Manager Tosch