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Mr. Eugene E. Fitzpatrick
Vice President/Director
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Dear Mr. Fitzpatrick:

Enclosed is a draft copy of NUREG-0822, Supplement No. 1, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Oyster Creek Nuclear Generating Station," which we intend to issue in May. Because of our recent reorganization, we feel it appropriate that you review this draft document before issuance to assure material continuity. Therefore, please review this draft for historical accuracy rather than technical concurrence.

We request that you provide comments, if any, by May 27, 1988.

*ORIGINAL SIGNED BY
JOHN F. STOLZ
John F. Stolz, Director
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Enclosure: Draft NUREG-0822, Suppl No. 1

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## ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) has prepared Supplement 1 to the final Integrated Plant Safety Assessment Report (IPSAR) (NUREG-0822), under the scope of the Systematic Evaluation Program (SEP), for the Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey and operated by GPU Nuclear Corporation and Jersey Central Power and Light Company (colicensees). The SEP was initiated by the NRC to review the design of older operating nuclear power plants to reconfirm and document their safety. This report documents the review completed under SEP for those issues that required refined engineering evaluations or the continuation of ongoing evaluations subsequent to issuing the Final IPSAR for the Oyster Creek Plant.

The review has provided for (1) an assessment of the significance of differences between current technical positions on selected safety issues and those that existed when the Oyster Creek Plant was licensed, (2) a basis for deciding how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety. The final IPSAR and its supplement will form part of the bases for considering the conversion of the existing provisional operating license to a full-term operating license.

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## ACRONYMS AND INITIALISMS

| | |
|---|---|
| ABT | automatic bus transfer |
| API | American Petroleum Institute |
| APRM | average power range monitor |
| ASME | American Society of Mechanical Engineers |
| BTP | Branch Technical Position |
| BWR | boiling-water reactor |
| CCW | closed cooling water |
| CFR | Code of Federal Regulations |
| cfs | cubic feet per second |
| CRD | control rod drive |
| DBE | design-basis event |
| DER | design electrical rating |
| ECCS | emergency core cooling system |
| EPFY | effective full-power year |
| ESF | engineered safety feature |
| FMEA | failure mode and effects analysis |
| FTOL | full-term operating license |
| GDC | General Design Criterion(a) |
| GE | General Electric |
| gpm | gallons per minute |
| HEPB | high energy pipe break |
| HVAC | heating, ventilating, and air conditioning |
| IE | Office of Inspection and Enforcement |
| IEEE | Institute of Electrical and Electronics Engineers |
| IPSAR | Integrated Plant Safety Assessment Report |
| IREP | Integrated Reliability Evaluation [illegible] |
| IRM | intermediate range monitor |
| ISI | inservice inspection |
| JCP&L | Jersey Central Power and Light Company |
| LER | licensee event report |
| LOCA | loss-of-coolant accident |
| LSSS | limiting safety system setpoint |
| LWR | light-water reactor |
| MCC | motor control center |
| MCPR | minimum critical power ratio |
| MDC | maximum dependable capacity |
| MOV | motor-operated valve |
| mph | miles per hour |
| MSIV | main steam line isolation valve |
| MSL | mean sea level |
| MWe | megawatt electric |
| MWt | megawatt thermal |
| NDT | nil ductility temperature |
| NRC | U.S. Nuclear Regulatory Commission |
| ORNL | Oak Ridge National Laboratory |
| PMH | probable maximum hurricane |
| PMP | probable maximum precipitation |
| POL | provisional operating license |
| ppm | parts per million |
| PRA | probabilistic risk assessment |

| | |
|---|---|
| psi | pounds per square inch |
| psig | pounds per square inch gage |
| PWR | pressurized-water reactor |
| RCPB | reactor coolant pressure boundary |
| RG | Regulatory Guide |
| RPS | reactor protection system |
| RTS | reactor trip system |
| RWCU | reactor water cleanup |
| SALP | systematic assessment of licensee performance |
| SAR | safety analysis report |
| SCS | shutdown cooling system |
| SEP | Systematic Evaluation Program |
| SER | safety evaluation report |
| SGTS | standby gas treatment system |
| SRP | Standard Review Plan |
| SSE | safe shutdown earthquake |
| STS | Standard Technical Specification |
| TMI | Three Mile Island |
| UHS | ultimate heat sink |
| USI | unresolved safety issue |
| VAR | voltage-ampere reactive |

INTEGRATED PLANT SAFETY ASSESSMENT REPORT
SUPPLEMENT NO. 1
SYSTEMATIC EVALUATION PROGRAM
OYSTER CREEK NUCLEAR GENERATING STATION

# 1 INTRODUCTION

The Systematic Evaluation Program (SEP) was initiated by the U.S. Nuclear Regulatory Commission to review the designs of older operating nuclear power plants to reconfirm and document their safety. The review provides (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

The results of the SEP review of the Oyster Creek plant were published in NUREG-0822, the Final Integrated Plant Safety Assessment Report (IPSAR), dated January 1983. The review compared the as-built plant design with current review criteria in 137 different areas defined as "topics." During the review, 54 of the topics were deleted from consideration in the SEP because a review was being conducted under other programs (unresolved safety issues or Three Mile Island Action Plan tasks), the topic was not applicable to the Oyster Creek Plant, or the items to be reviewed under that topic did not exist at the site.

Of the original 137 topics, 83 were, therefore, reviewed for Oyster Creek; of these 43 met current criteria or were acceptable on another defined basis. From the review of the 40 remaining topics, certain aspects of plant design were found to differ from current criteria. These 40 topics were considered in the integrated assessment of the plant, which consisted of evaluating the safety significance and other factors of the identified differences from current design to arrive at decisions on whether modification was necessary from an overall plant safety viewpoint. To arrive at these decisions, engineering judgment was used as well as the results of a limited probabilistic risk assessment study.

In general, the staff's positions in the integrated assessment fell into one or more of the following categories: (1) equipment modification or addition, (2) procedure development or Technical Specification changes, (3) refined engineering analysis or continuation of ongoing evaluation, and (4) no modification necessary. Table 4.1 of the IPSAR summarizes the staff's integrated assessment positions and documents the licensee's agreement with these positions.

For those positions classified as either Category (1) or (2), the IPSAR lists the scheduled completion dates agreed upon by the staff and the licensee. Region I has verified or is verifying the implementation of these positions.

For those positions classified as Category (3), the licensee has provided the results of the ongoing evaluation to the staff for review. The purpose of this supplement to the IPSAR is to provide the staff's evaluation of the Category (3) issues and to summarize the status of all actions to be implemented as result of the SEP review.

The Oyster Creek plant is presently one of the four SEP plants that has not received a full-term operating license (FTOL). Therefore, a safety evaluation report (SER) to support the conversion of the provisional operating license (POL) to an FTOL will be prepared. The SER will consist of the IPSAR, the IPSAR supplement, a consideration of major plant modifications that have been made and substantive regulations adopted since the POL was issued, and the unresolved safety issues and Three Mile Island Action Plan issues.

# 2 TOPICS THAT REQUIRED REFINED ENGINEERING ANALYSIS OR CONTINUATION OF ONGOING EVALUATION

The licensee has submitted an evaluation for each of the issues that required refined engineering analysis or further evaluation. The staff reviewed these submittals and concluded that either the licensee met current criteria, the evaluation was acceptable on another defined basis, or corrective action will be required, or further analysis will be required. Factors considered in reaching this conclusion include the perceived safety significance of the difference from current licensing criteria, a qualitative assessment of the financial and exposure costs to make a modification, and, to a lesser extent, implementation impact and schedule. The evaluation of these issues also considered any applicable risk perspectives, developed for the integrated assessment and described in the IPSAR, and related corrective actions proposed by the licensee as part of the integrated assessment or as a result of the follow-on evaluations.

A brief discussion of each of the outstanding issues is presented below. Each evaluation references the more detailed license evaluation and staff topic evaluation. References for correspondence pertaining to safety evaluation reports for each section appear in Appendix A. Appendix B is a listing of the staff contributors.

The status of each of these issues is summarized in Table 4.1 along with the status of all SEP issues for the Oyster Creek Nuclear Generating Station.

## 2.1 Topic II-3.B, Flooding Potential and Protection Requirements; Topic II-3.B.1, Capability of Operating Plants To Cope With Design-Basis Flooding Conditions; Topic II-3.C, Safety-Related Water Supply (Ultimate Heat Sink (UHS)) (NUREG-0822, Section 4.1)

10 CFR 50 (GDC 2) as implemented by SRP Sections 2.4.2, 2.4.5, 2.4.10, and 2.4.11 and Regulatory Guides 1.59 and 1.27, require that structures, systems and components important to safety be designed to withstand the effects of

natural phenomena such as flooding. The safety objective of these topics (II-3.B, II-3.B.1, and II-3.C) is to verify adequate operating procedures and/or system design provided to cope with the design-basis flood.

The site grade elevation is 23 ft mean sea level (MSL). During the staff's review of the hydrology-related topics, the following flooding elevations were identified by current licensing criteria:

- probable maximum hurricane (PMH) - 22 ft MSL
- probable maximum precipitation (PMP) - 23.5 ft MSL

As a result of these flooding levels, the staff identified in the IPSAR nine issues.

These nine issues are the following: (1) condensate transfer pumps, (2) plant operating limits on canal water level in the Oyster Creek Technical Specifications (TS), (3) canal water level instrumentation, (4) makeup isolation condenser water sources, (5) plant operating limits in the TS on water level at the service water intake, (6) procedures for a flood, (7) protection during internal flooding, (8) hydrostatic loads on buildings and (9) reactor and turbine building parapets and scuppers. Issues (2), (4), (6) and (9) were resolved by commitments made by the licensee for specific plant modifications or plant procedure changes. These are discussed below in Section 4.0. Issue 8 is discussed in Section 2.4. Issue (5) is discussed in Section 3.1. Items (1), (3), and (7) are discussed below in subsections 2.1.1 to 2.1.3, respectively.

2.1.1 Condensate Water Pumps (NUREG-0822, Section 4.1(1))

In Section 4.1(1) of the IPSAR, the staff concluded that two condensate transfer pumps are essential to charge the emergency condenser with cooling water during a hurricane induced flood. Because both of these pump motors are powered from the same engineered safety features bus, a single failure of the power bus would disable both condensate transfer pumps.

The staff further concluded that in conjunction with the resolution of Topic III-4A (see Section 4.6.4 of the IPSAR), the licensee has committed to provide a portable pump to provide cooling water in the event of a loss of cooling resulting from tornado-missile damage. The staff concludes that this diverse means of cooling is sufficient to alleviate the need for redundant power for the condensate transfer pumps. Therefore, backfitting is not recommended.

In a letter dated July 3, 1985 the licensee proposed to utilize a main core spray pump to supply the Isolation Condenser. This would be accomplished by connecting a temporary hose to one of the core spray system loops and routing the hose to the Isolation Condenser. Both the water supply (suppression chamber and the components) would be protected by potential tornado missiles and external flooding. Subsequently the licensee in a letter dated August 14, 1987 stated that through a detail field walk down and line loss analysis of an existing system interconnection between Core Spray and Condensate and Demineralized Water Transfer Systems, it was determined that the existing plant configuration is capable of supplying make-up water to the Isolation Condenser. The staff has not completed its review regarding this matter. Upon completion of our review, we will document the results of our review in a supplement to the IPSAR.

2.1.2 Canal Water Level Instrumentation (NUREG-0822, Section 4.1(3))

On the basis of its review, the staff concluded in IPSAR Section 4.1(3) that water level instrumentation in the intake canal is inadequate and there is no water level measurement in the discharge canal. Accordingly, the staff recommended that automatic water level instrumentation be provided in both canals, with measurement indication in the control room, so that the operator would be able to implement emergency shutdown procedures when the specified flooding levels occur. Because these instruments are not intended for postaccident monitoring, they need not necessarily be safety grade.

We also stated in Section 4.1(3) of the IPSAR, that the licensee committed to install an automatic water level gage, with a remote readout, in the intake canal. Another water level gage in the discharge canal is not necessary because flooding conditions can be identified from the intake canal measurement. This modification will be coordinated with other modifications being considered by the licensee for canal monitoring, including upgrading the existing visual gages, and the installation will be completed by the end of the Cycle XI refueling outage.

In its letter dated April 21, 1986, the licensee requested to cancel its commitment to install an automatic water level gage in the intake canal with a remote readout in the control room. The licensee proposed revising station procedure 2000-ABN-3200.31, High Winds, to require a plant shutdown when the water level at the intake structure cannot be verified to be less than elevation 4.5 ft MSL. This is acceptable to the staff and was documented in the staff's Safety Evaluation (SE) dated November 28, 1986. The former Project Manager (Jack Donehew) also verified that this shutdown requirement had been added to Procedure 2000-ABN-3200.31. This closes out this SEP issue. A discussion of low water level in the intake structure is presented in Section 3.1.2 of this supplement.

2.1.3 Protection During Internal Flooding (NUREG-0822 Section 4.1(7))

In IPSAR Section 4.1(7), the staff stated that protection against internal flooding of structures caused by local PMP should be provided to a flood level of 23.5 ft MSL. The licensee should verify that all entrance levels are above this level. The southwest door of the offgas building may flood even though the sill is at 23.5 ft MSL because of the configuration of contours near the door.

We further stated that the licensee proposed to evaluate the consequences of flooding in the offgas building and will confirm that no other entrance level is below 23.5 ft MSL.

By letter dated June 6, 1983, the licensee stated that all sill and entry flood elevations are at or above 23'-6" MSL for the reactor building, turbine building and new and old rad waste buildings and, thus, modifications are not required. However, the licensee's review indicated that the diesel generator building has two entrances at elevation 23'-0" MSL which could potentially expose the enclosed switchgear cabinets to flooding. The licensee proposed to construct a six inch high asphalt dike at the two above entrances to provide protection from surface water entry during the next operating cycle. Further, the licensee stated that a review of contour maps of the site has shown no indication of contours which might impound water at the southwest door of the offgas building and, therefore, modifications are not required.

In a letter dated June 23, 1983, the staff found the corrective actions proposed to be acceptable and considers it sufficient to resolve this SEP issue.

Region I will verify that modifications discussed above have been completed.

## 2.2 Topic III-1, Classification of Structures, Components, and Systems (Seismic and Quality) (NUREG-0822, Section 4.2)

10 CFR 50 (GDC1), as implemented by Regulatory Guide 1.26, requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of safety functions to be performed. The codes used for the design, fabrication, erection, and testing of the Oyster Creek plant were compared with current codes.

In IPSAR Section 4.2, the staff stated that the review of this topic identified several systems and components for which the licensee was unable to provide imformation to justify a conclusion that the quality standards imposed during plant construction meet quality standards required for new facilities. The staff did not identify any inadequate components. However, because of the limited information on the components involved the staff was unable to conclude that for code and standard changes deemed important to safety, the Oyster Creek plant met current requirements.

The staff further stated that the licensee agreed to complete the evaluations described in Sections 4.2 of NUREG-0822 and incorporate the results in the Final Safety Analysis Report update, which must be submitted within 2 years after completion of the SEP review [10 CFR 50.71 (e)(3)(ii)]. If the results of the licensee's evaluations indicate that the facility modifications are required those actions will be reported in a licensee event report.

The licensee has indicated that he will provide this information. Upon receipt of this information the staff will evaluate it and present its finding in a supplement to the IPSAR.

## 2.3 Topic III-2, Wind and Tornado Loadings (NUREG-0822, Section 4.3)

10 CFR 50 (GDC2), as implemented by SRP 3.31 and 3.32 and Regulatory Guides 1.76 and 1.117, requires that the plant be designed to withstand the effects of natural phenomena such as wind and tornadoes.

In Section 4.3 of the IPSAR, the staff identified several structures important to safety as not meeting current licensing criteria regarding their ability to resist tornadoes.

### 2.3.1 Reactor Building Steel Structure Above the Operating Floor (NUREG-0822, Section 4.3.1)

In IPSAR Section 4.3.1, the staff concluded that the capacities calculated by the staff were lower (differential pressure induced by a 61-mph windspeed) than those required by the site-specific tornado-imposed loads. The staff also indicated that the licensee is analyzing these structures to determine capacities and will provide the results and identify proposed corrective actions to the NRC upon completion.

In a letter dated February 2, 1983, the licensee provided supporting calculations to justify their conclusions which were originally presented in their letter of May 7, 1981.

The staff, with assistance from Franklin Research Center, reviewed the supporting calculations and did not agree with the limiting windspeed presented by the licensee. In an SER dated March 8, 1986, the staff concluded that the licensee should: (1) determine the capability of the structure with appropriate considerations as presented by the staff in Section IIIA of the SER and (2) evaluate potential modifications which would increase the plant's capability to withstand severe wind and tornado loads.

Meetings at the Oyster Creek Station on Monday, February 2, 1987 to Friday, February 6, 1987, the licensee stated that they will provide the information concerning this matter. Upon receipt of this information, the staff will evaluate it and report its finding in a supplement to the IPSAR.

### 2.3.2 Ventilation Stack (NUREG-0822, Section 4.3.2)

In IPSAR Section 4.3.2, the staff concluded that the stack capacities calculated by the staff are lower (164-mph windspeed) than those required by the site-specific tornado-imposed loads. Failure of the stack could affect the integrity of seismic Category I structures because the stack is in close proximity to these structures. The licensee is performing an analysis of the stack and a probabilistic evaluation of tornado-(or high-wind) induced stack failure and its consequences. The licensee had agreed to perform the analysis, identify any necessary corrective actions, and submit the results to the staff.

By letter dated February 2, 1983, the licensee submitted the results of their analysis and concluded that the stack is capable of withstanding a 180 mph wind load. The licensee concluded that 180 mph wind load corresponds to an exceedance probability of $1x10^{-6}$/year which is sufficiently low to make the installation of modifications unwarranted. In an SER dated March 8, 1986, the staff noted that 180 mph corresponds to a probability of exceedance of approximately $5x10^{-6}$/year using the NRR estimate of tornado hazard at Oyster Creek. The staff also concluded that considering the various means of plant shutdown available the staff considers that the conditional probability of core damage given stack failure is acceptably small. The stack is capable of withstanding 180 mph ($5x10^{-6}$/year) if resonance does not occur. Therefore, the staff concludes that no further evaluation of the stack is warranted. The staff also concluded that the issue of tornado loads in conjunction with wind loads for the stack is considered resolved. This closes out this SEP issue.

### 2.3.3 Effects of Failure of Nonseismic Category I Issues (NUREG-0822, Section 4.3.3)

In IPSAR Section 4.3.3, the staff stated that the licensee will evaluate the turbine building capacity and the effect of its failure on other structures (e.g., the control room). The licensee has agreed to perform the analysis, identify any necessary corrective actions, and submit the results to the staff.

The licensee provided the results of its evaluation in a submittal dated March 13, 1984. The analysis presented by the licensee modeled the turbine building as two and three dimensional frames and analyzed them using a computer code.

As a result of the analysis, the licensee concluded that for load combinations involving loads such as dead load and snow in combination with wind, the turbine building will remain stable for load conditions involving an 80 mph wind loading. The licensee has also concluded that failure of the roof purlins by the roof deck/purlin connections will not cause turbine building failure.

In an SE dated March 8, 1986, the staff concluded that overall, the structural system of the turbine building is capable of resisting reasonably high levels of loading. It further concluded that no further evaluation of the turbine building is warranted. This closes out this SEP issue.

The issue of wind load combinations is addressed in Section 3.12 of this supplement.

2.3.4 Roof Decks (NUREG-0822, Section 4.3.6)

In IPSAR Section 4.3.6, the staff stated that the licensee had indicated that the roof deck of the reactor building can withstand a 280-mph wind and a 0.68-psi differential pressure. In addition, the licensee had stated that the roof of the diesel generator building can withstand a 300-mph wind and a 2-psi differential pressure. The licensee will evaluate the capacity of the roof deck of the turbine building.

In a letter dated March 13, 1984, the licensee provided an evaluation of the roof decks of these structures in their analysis regarding capacities of the reactor building above the operating floor and the turbine building.

In our Safety Evaluation dated March 8, 1986, the staff addressed the roof decks as a part of their evaluation of the reactor building above the operating floor and the turbine building. As discussed in Section 2.3.1 of this supplement the evaluation of the reactor building above the operating floor is not complete and therefore this issue is considered open. With regard to the roof deck of the turbine building, we concluded in Section 2.3.3 of this supplement that the overall structural system of the turbine building is capable of resisting high levels of loading and considered the turbine building issue resolved.

### 2.3.5 Intake Structure, Oil Tanks and Diesel Generator Building (NUREG-0822, Section 4.3.7)

In Section 4.3.7 of the IPSAR, we stated that sufficient information was not available for the staff to conclude that these structures had enough capacity for the postulated wind and tornado loadings and that the licensee will submit an analysis of these structures.

In the safety analysis report provided by the licensee dated May 7, 1981, the licensee concluded that the intake structure and the diesel generator and oil tank vaults are capable of withstanding a 300 mph wind and a 2.0 psi depressurization load. By letters dated February 2, 1983 and October 25, 1983, the licensee provided support calculations for the values given in the report.

In our Safety Evaluation of March 8, 1986, we concluded that the windspeed ratings of these structures as determined by the licensee are valid. However, no evaluations were presented to evaluate the effects of tornado missile loads in load combinations involving tornado wind loads for the diesel generator and oil tanks.

We further stated that the major portions of the diesel generator and oil tank vaults are substantial reinforced concrete structures with a roof thickness of 1'-0" and wall thicknesses of 1'-6". Although no evaluation of missile loads in combination with wind loads has been performed, the thicknesses of the structure's roof and walls are such that it is expected that they will afford a substantial amount of protection. The capacity to resist missile and wind loads simultaneously would be less than 300 mph as reported for wind acting alone; however, even if the resistance reduces to a windspeed such as 120 mph, the probability of exceedance is approximately $5 \times 10^{-5}$/year which is low.

In our Safety Evaluation of March 8, 1986, we also stated that in response to the tornado missile issue, the licensee has committed to provide a portable pump in a protected area and hose connections to a protected water supply to be used in conjunction with the isolation condenser to achieve hot shutdown. Thus, even if damage to the oil tank and diesel generators snould occur, safe shutdown still can be achieved. The staff concluded from this that analyses of the oil tank and diesel generator vaults to withstand tornado missile loads in combination involving tornado wind loads is not warranted.

In a letter dated July 3, 1985, the licensee proposed to utilize a main core spray pump to supply the Isolation Condenser. This would be accomplished by connecting a temporary hose to one of the core spray system loops and routing the hose to the Isolation Condenser. Both the water supply (suppression chamber and the components) would be protected by potential tornado missiles and external flooding. Subsequently, the licensee in a letter dated August 14, 1987 stated that through a detailed field walkdown and line loss analysis of an existing system interconnection between Core Spray and Condensate and Demineralized Water Transfer Systems, it was determined that the existing plant configuration is capable of supplying makeup water to the Isolation Condenser. The staff has not completed its review regarding this matter. Upon completion of our review, we will document the results of our review in a supplement to the IPSAR.

2.3.6 Load Combination (NUREG-0822, Section 4.3.8)

In Section 4.3.8 of the IPSAR, we stated that as a result of the topic review, the staff was unable to determine if straight wind loads (not tornado loads) were combined with other loads (i.e., snow loads, operating pipe reaction loads, and thermal loads). The staff also stated that the licensee stated that recent analyses have included these loads. These analyses will be submitted to the staff in conjunction with Topic III-7.B (see Section 4.12).

The staffs' evaluation of this matter is discussed in Section 2.10 of this supplement.

2.3.7 Control Room (NUREG-0822, Section 4.3)

The staff Safety Evaluation Report dated September 1, 1982, concluded that the licensee should provide a description of the methods and sample calculations used to qualify the control building. By letter dated February 2, 1983, the licensee provided supporting calculations for control room capacities. The licensee concluded that the control room north wall is capable of resisting a 160 mph tornado wind and a .53 psi depressurization load. The balance of the control room is capable of resisting a 300 mph tornado wind and a 2.0 psi pressure differential. The 160 mph wind and a .53 psi pressure drop corresponds to a probability of exceedance of approximately $1 \times 10^{-5}$ year.

In its Safety Evaluation dated March 8, 1986, the staff concludes that the wind-speed ratings developed by the licensee are valid. The licensee's February 1, 1983 submittal also notes that the control room cannot withstand the tornado missile load in conjunction with the tornado wind load. No assessment of the effects of failure of the wall has been provided. The staff concludes that the licensee should demonstrate that failure of the wall will not prevent safe plant shutdown or should propose corrective actions.

The licensee is presently performing an evaluation of this matter and will submit the results of sits evaluation to the staff. Upon receipt of the licensee's evaluation, the staff will review it and document the results in a Supplement to the IPSAR.

2.3.8 Architectural Components (NUREG-0822, Section 4.3)

In the staff's Safety Evaluation of March 8, 1986, the staff stated that the licensee should verify that architectural components, such as rollup doors, are not located such that damage to required equipment could occur.

The licensee is presently evaluating this matter and will submit the result of its evaluation to the staff. Upon receipt of the licensee's evaluation, the staff will review it and document the result of its review in a Supplement to the IPSAR.

## 2.4 Topic III-3.A, Effects of High Water Level on Structure, (NUREG-0822, Section 4.4)

10 CFR 50 (GDC 2), as implemented by SRP Section 3.4 and Regulatory Guide 1.59, requires that plant structures be designed to withstand the effects of flooding.

In IPSAR Section 4.4 (2), the staff concluded that the licensee should demonstrate that safety-related structures remain functional for a short-term hydrostatic loading and can resist flotation for water levels up to 22 ft msl.

By letter dated July 1, 1983, the licensee provided results of their analyses for the reactor building, the turbine building, the diesel generator building, and the new radwaste building. Based on the results, the licensee concluded that the structures are adequate to resist the loadings.

The staff evaluation issued on February 23, 1984, concluded that based on the factors of safety obtained against flotation, the adequacy of the subgrade walls and the adequacy of bearing capacity, the Oyster Creek facility can adequately withstand a groundwater level of elevation 23 ft msl. This closes out this SEP issue.

## 2.5 Topic III-4A, Tornado Missiles (NUREG-0822, Section 4.6)

10 CFR (GDC 2) as implemented by Regulatory Guide 1.117 prescribes structures, systems, and components that should be designed to withstand the effects of a tornado, including tornado missiles, without loss of capability to perform their safety functions.

In Section 4.6 of the IPSAR, the staff identified several structures and components as being vulnerable to tornado missiles.

### 2.5.1 Emergency Diesel Generators and Fuel Oil Day Tank (NUREG-0822, Section 4.6.1)

In Section 4.6.1 of the IPSAR we stated that the licensee concluded that the diesel generators are not necessary for safe shutdown because makeup water could be provided to the isolation condenser by diesel-driven fire water pumps and by dc power to the main steam relief valves. Section 4.6.1 of IPSAR also indicates that the licensee has agreed to evaluate the potential for and consequences of tornado-missile damage to the diesel generator building. The status of this issue is discussed in Section 2.3.5 of this supplement.

2.5.2 Mechanical Equipment Access Area (NUREG-0822, Section 4.6.2)

In Section 4.6.2 of the IPSAR, the staff identified several components (e.g., motor control centers (MCC-DC-1 and MCC-1AB 21B), control rod drive hydraulic filter, isolation fill piping, and containment spray valve) in the vicinity of the mechanical equipment access opening of the reactor building that are potential targets for missiles penetrating the access doors, which had not been considered in the staff's original evaluation.

We further stated that the licensee has agreed to evaluate the potential for and consequences of tornado-missile impact on components in this area and provide protection, if necessary.

By letter dated September 16, 1983, the licensee provided an analysis of tornado missile risk for Oyster Creek. This analysis included development of an annual tornado windspeed exceedance curve.

In a letter dated December 27, 1983, we stated that the staff independently calculated a probability distribution for high winds and tornadoes for the Oyster Creek site and found that non-tornado wind frequency is higher than the licensee's values. Therefore, we requested the licensee to evaluate the consequences of wind-generated missiles (from windspeeds less than 125 mph) to determine whether such missiles contribute significantly to target damage.

In letters dated October 15, 1984, and June 7, 1985, the licensee provided the requested information.

We have not completed our review of the licensee's information. Upon completion of our review, we will document the results in a supplement to the IPSAR.

2.5.3 Condensate Storage Tank, Torus Water Storage Tank, and Service Water and Emergency Service Water Pumps (NUREG-0822, Section 4.6.4)

In Section 4.6.4 of the IPSAR we stated that the licensee's position is that the condensate storage tank and torus water storage tank are not required to accomplish safe shutdown because the plant can be safely shut down using one of the two service water pumps or any of the four emergency service waterpumps. It is also the licensee's position based on redundancy of these pumps that backfitting is not required.

We further indicated that redundancy is not acceptable protection for tornado missiles. Therefore, it is the staff's position that the licensee provide protection for sufficient systems and components to ensure a safe shutdown in the event of damage from tornado missiles.

In IPSAR Section 4.6.4, we also indicated that the licensee has agreed to provide a portable pump in a protected area and hose connections to a protected water supply. Further, the licensee will provide procedures that specify the conditions for and use of this equipment. The staff found this action acceptable. However, as discussed in Section 2.1.1 of this report, the licensee now proposes to use an existing system interconnection between Core Spray and Condensate and Demineralized Water Transfers Systems to achieve safe shutdown of the plant. As discussed in Section 2.1.1 of this report, the staff had not completed its review of this matter. Upon completion of our review, we will document the results of our review in a supplement to the IPSAR.

## 2.6 Topic III-4.B, Turbine Missiles (NUREG-0822, Section 4.7)

10 CFR 50 (GDC 4) as implemented by R.G. 1.115 and SRP Section 3.5.1.3, requires that structures, systems and components important to safety be appropriately protected against dynamic effects, which include missiles.

One means of providing adequate protection is assurance of a low probability of failure of the turbine at design or destructive overspeed. This assurance arises in part from inspection of the turbine discs and testing and inspection of stop and control valves at regular intervals.

In the IPSAR, Section 4.7, the staff concluded that the licensee should do the following:

(1) Perform a volumetric inspection of the turbine during the cycle 10 outage, and based on results of that inspection propose an inspection frequency.

(2) Describe the monitoring program for main steam control valves and reheat control valves to justify why cycling these valves individually to a fully closed position on a weekly basis should not be done.

In a letter dated May 17, 1984, the licensee described inspections performed in April and June 1983 by General Electric (GE) of the shrunk-on wheels from the low pressure rotors LPA, LPB and LPC. Visual and ultrasonic examinations were performed. Indications on the wheel bores and keys were found by ultrasonic examination. General Electric and GPU have concluded that the indications do not affect the structural integrity of the wheels and keyways and, as a result, General Electric has recommended that another ultrasonic inspection be performed after approximately 6 years of additional operation. The licensee has committed to conduct the inspection within six years of operation, which is the schedule typically recommended by the vendor for its turbines.

In a letter dated December 8, 1983, the licensee described the valve monitoring program in effect. The four individual turbine stop valves are currently fully closed on a daily basis. The six reheat stop valves and six intercept valves are individually brought to full closure once a week.

In the staff's SER dated August 21, 1986, the staff concludes that the licensee has proposed a turbine inspection schedule based on a previous inspection and on vendor recommendations. The testing meets the intent of staff criteria, that is, to verify the ability of the stop and control valves to close to prevent turbine overspeed, even though full closure testing of the control valves is not practical. Therefore, the staff concludes that the licensee's response to IPSAR Section 4.7 is acceptable. This closes out this SEP issue.

## 2.7 Topic III-4.D, Site-Proximity Missiles (Including Aircraft) (NUREG-0822, Section 4.8)

10 CFR 50 (GDC 4), as implemented by SRP Sections 3.5.1.5, 3.5.1.6, and 2.2.3, requires that structures, systems and components important to safety be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.

In Section 4.8.2 of the IPSAR, the staff concluded that because there are several airports near the site, the licensee should addresss the potential for a consequence of aircraft impact.

The licensee submitted its analysis of the probability of an aircraft strike on the plant in a letter dated March 4, 1983. The staff reviewed the licensee's submittal and issued its evaluation dated May 3, 1983. In that evaluation, the staff concluded that the licensee's analysis was performed in accordance with current criteria and that because the aircraft strike probabilities are extremely low, aircraft traffic does not pose a significant threat to the Oyster Creek Plant. Therefore, this issue is considered resolved.

## 2.8 Topic III-5.B, Pipe Break Outside Containment (NUREG-0822, Section 4.10)

10 CFR 50 (GDC 4), as implemented by SRP Sections 3.6.1, and 3.6.2 and Branch Technical Position (BTP) MEB 3-1 and ASB 3-1, requires, in part, that structures, systems, and components important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures. The safety objective for this topic review is to ensure that if a pipe should break outside the containment, the plant can be safely shut down without a loss of containment integrity.

### 2.8.1 Emergency Condenser Steam Lines (NUREG-0822, Section 4.10.2)

In Section 4.10.2 of the IPSAR, the staff stated that the emergency condenser steam lines have two automatic isolation valves outside and adjacent to the drywell. A break between these valves with a failure of the first isolation valve or a pipe break between the second valve and the condenser resulting in pipe whip such that the isolation valves would not close would both result in a LOCA outside containment. The physical arrangement and space availability preclude installation of restraints. In addition, it is not practical to install an isolation valve inside the drywell.

We further stated that "the licensee has submitted a fracture mechanics analysis that demonstrates that through-wall cracks in the emergency condenser steam pipe would open up, yet remain stable, under severe pipe pressure loading and rotation stresses. No instantaneous pipe break would occur. The estimated pipe leakage for these through-wall cracks would be less than 1 gpm."

"The licensee inservice inspection of these lines is in accordance with Section XI of the ASME Code. The licensee has committed to submit a reanalysis of the emergency condenser piping along with an evaluation of leakage detection and a schedule for any necessary corrective actions. This evaluation is scheduled to be submitted in February 1983. The staff finds this action acceptable."

In a letter dated October 16, 1984 the licensee provided the report entitled "Crack Growth and Leak Rate Assessment of the Oyster Creek Emergency Condenser System Piping Outside Containment Below the 95 Foot Elevation." The results of the analysis indicate that the leak rates from postulated cracks are sufficiently high to use visual monitoring as an acceptable method of leak detection. The licensee further stated that sufficient time exists to take appropriate actions (i.e., shut down or isolate the affected condenser) between the time of leak detection and the time that a crack would grow to an unstable length.

Subsequently, in January 1986, Niagara Mower Power Corporation (NMPC) notified NRC of a condition involving a failure mode for the Nine Mile Point/drywell penetrations. Loads calculated for the penetrations due to a postulated High Energy Line Break (HELB) in the process piping within the penetrations were determined to exceed those for which the penetrations were designed. These higher loads resulted from use of a more accurate analysis model which included both pressure and momentum effects. The licensee became aware of the NMPC analysis and in February 1986, voluntarily initiated an investigation of the Oyster Creek drywell penetrations. The results of this investigation were discussed with the staff in the meeting of August 22, 1986. The NRC meeting minutes are dated October 1, 1986. At that time, the four piping penetrations for the Isolation Condensers (IC) piping were identified to be below the updated Final Safety Analysis Report design criteria and would fail if a guillotine rupture of the pipe occurred within the penetration. This was also reported to the staff in License Event Report No.86-024, dated October 17, 1986.

As a result of several telephone discussions with the staff, the licensee in letters of September 17, and November 25, 1986 provided additional information. In the latter letter the licensee committed (1) to coordinate the final resolution of four piping penetrations with the resolution of NUREG-0313, Revision 2, requirements on welds inside these penetrations and with the staff's Systematic Evaluation Program (SEP) Topic III-5.B on the two containment isolation valves outside containment on the IC steam lines and (2) to operate Oyster Creek with additional limiting conditions for operation on the reactor coolant leakage within the drywell.

In a letter dated December 24, 1986 the staff stated that they revirwed the licensee's letter and data on the HELB within the IC penetrations through the drywell and as discussed in the Safety Evaluation dated December 24, 1986, the staff concludes that operations of Oyster Creek for operating Cycle 11 is acceptable. The modifications to the four piping penetrations will be completed in the Cycle 12R outage which starts in September 1988.

In our letter of December 24, 1986 we also stated that as the license explained in its letter dated November 25, 1986, completing the modification in Cycle 12R outage is contingent on finalizing the design of the penetrations and completing the engineering for modifications in time for the outage. The staff will be involved in this activity with the licensee because this design will involve NUREG-0313 Revision 2 and SEP Topic III-5-B.

Upon receipt of the licensee's proposed resolution of this matter, we will review the information and report the results of our review in a supplement to the IPSAR.

## 2.9 Topic III-6, Seismic Design Considerations (NUREG-0822, Section 4.11)

10 CFR 50 (GDC 2) as implemented by SRP Sections 2.5, 3.7, 3.8, 3.9, and 3.10 and SEP review criteria (NUREG/CR-0098), requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes.

In Section 4.11 of the IPSAR, the following areas were identified for further evaluation.

### 2.9.1 Piping Systems (NUREG-0822, Section 4.11.1)

In Section 4.11.1 of the IPSAR, we stated that it was the staff's position that the licensee perform analyses on a sampling basis (e.g., two randomly selected piping analyses) on piping systems 2 1/2 to 10 in. in diameter as well as reanalysis of the control rod drive system to the site-specific spectra, including information on the building model and floor respond spectra. In addition, the licensee should verify the design adequacy of piping supports for the main steam and feedwater lines.

The licensee provided several submittals responding to our requirements as specified in Section 4.11.1 of the IPSAR. The staff reviewed this information and by letter dated Janu y 9, 1986, provided a draft Technical Evaluation Report which identified the areas where additional information is needed. This matter was discussed at a meeting on April 24, 1986 (meeting summary dated May 19, 1986) and the licensee in a letter dated June 24, 1986 provided the additional information required.

The staff is presently reviewing this information. Upon completion of our review we will document the results of our review in supplement to the IPSAR.

### 2.9.2 Mechanical Equipment (NUREG-0822, Section 4.11.2)

In Section 4.11.2 of the IPSAR, the staff required the licensee to demonstrate that the control rod drive hydraulic units and associated tubing supports as well as the reactor vessel internals have sufficient capacity to maintain integrity following the safe shutdown earthquake (SSE).

In a letter date January 20, 1983, the licensee submitted a report entitled "Reanalysis of the Control Rod Drive Return System Piping Considering Axial U-Bolt Restraint and Site Specific Spectra." The licensee stated that the reanalysis demonstrates that the Control Rod Drive return system piping stresses are within Code Allowables for the Oyster Creek Site specific seismic spectra.

In a letter dated January 24, 1983, the licensee submitted a report entitled "Oyster Creek Seismic Analysis of Reactor Vessel Internals. The licensee stated that the analysis was performed to address questions raised by the NRC staff during the review of SEP seismic considerations.

The staff is presently reviewing this information. Upon completion of its review, the staff will document the results of its review in a Supplement to the IPSAR.

2.9.3 Electrical Equipment (NUREG-0822, Section 4.11.3)

In Section 4.11.3 of the IPSAR, the staff stated that it is concerned that the structural integrity of the panels (load path from an internally mounted element to anchorage support systems) has not been demonstrated. The licensee has proposed to perform an evaluation of the load path for at least two typical cabinets.

In a letter dated March 13, 1984, the licensee submitted its evaluation and results of a seismic analysis of two types of safety-related equipment at the Oyster Creek Nuclear Generating Station; 4160 volt switchgear and 460 volt unit substation cabinets. Anchor adequacy and internal load path evaluations for both types of the cabinets were conducted. The staff reviewed this information and in a letter dated January 8, 1986, provided a draft Technical Evaluations Report which identified the areas where additional information is required. This matter was discussed at a meeting on April 1986 (meeting summary dated May 19, 1986) and the licensee in a letter dated June 24, 1986 provided additional information. The staff is presently reviewing this information. Upon completion of our review we will document the result of any review in a supplement to the IPSAR.

2.9.4 Qualification of Cable Trays (NUREG-0822, Section 4.11.5)

The staff is concerned that safety-related cable trays may not be able to withstand the postulated seismic loads. The SEP Owners Group has conducted tests on typical cable tray configurations found in nuclear power plants. One report summarizing the test results was submitted to the staff in April 1983; a second report containing cable tray evaluations criteria and guidelines developed from the tests was submitted in August 1983.

On February 19, 1987, the NRC issued Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors Unresolved Safety Issue (USl) A-46. The Generic Letter provided guidance for resolution of USl on A-46 and requested that licensees submit a schedule for final resolution of the issue using that guidance.

In a letter dated October 9, 1987, the Seismic Qualification Utility Group (SQUG) of which GPU is a member stated that it is currently developing the "Generic Implementation Procedure for Verification of Seismic Adequacy of Nuclear Plant Equipment" (G.I.P) for use by its members. This procedure will be in three parts. Part 1 provides SQUG's positions relative to Generic Letter 87-02. Part 2 is a detailed technical document containing criteria and associated guidance for USl A-46 resolution. Part 3 will consist of a series of training seminars to be sponsored by SQUG.

In a letter dated November 19, 1987, we advised SQUG that its letter of October 9, 1987 is acceptable to meet the December 1, 1987 reporting deadline as set forth in the April 28, 1987 letter. Therefore, licensees participating in the SQUG program do not need to respond and provide separate responses to GL-87-02 until the staff issues its Safety Evaluation Report (SER). All licensees participating in the SQUG implementation program should provide their schedules for plant specific implementation no later than sixty days after receiving the generic SER.

2.10 Topic III 7.B. Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria (NUREG-0822, Section 4.12)

10 CFR 50 GDC 1, 2, and 4, as implemented by SRP Section 3.8, requires that structures, systems, and components be designed for the loading that will be imposed on them and that they conform to applicable codes and standards.

In Section 4.12 of the IPSAR, the staff concluded that design code changes potentially applicable to the Oyster Creek Plant, where the current code requires substantially greater safety margins than the earlier version of the code or where no original code provision existed, should be evaluated to ensure adequate margins of safety. The licensee committed, in the integrated assessment, to (1) review the NRC evaluation to determine applicability of the structural elements identified and (2) perform, on a sampling basis an evaluation of the code, load and load combination changes on existing as-built structures to assess adequacy of the design.

By letter dated June 4, 1984, the licensee submitted an evaluation of design codes, design criteria, and load combination changes for the Oyster Creek Station as requested in Section 4.12 of the IPSAR.

In the staff's Safety Evaluation dated October 29, 1986, the staff concluded that based on their review, its consultant, Franklin Research Center, the loads and load combination issues are satisfactorily resolved. With respect to the design codes and criteria changes, twenty of the twenty-three issues are fully resolved. For two of the design code changes, related to reinforcement of openings, further information is requested. For the remaining issue, concrete, subject to high temperatures and thermal transients, the licensee stated that further investigation of drywell thermal conditions are necessary.

Upon receipt of the information provided by the licensee, the staff will review the information and report the results of their review in a future supplement to the IPSAR.

## 2.11 Topic III-10.A, Thermal-Overload Protection for Motors of Motor-Operated Valves (NUREG-0822, Section 4.14)

10 CFR 50.55a(h), as implemented by Institute of Electrical and Electronics Engineers (IEEE) Std. 279-1971 and 10 CFR 50 (GDC 13, 21, 22, 23 and 29), require that protective actions be reliable and precise and that they satisfy the single-failure criterion using quality components.

In IPSAR Section 4.14(1) the staff concluded that the adequacy of the setpoints for unbypassed thermal overloads on some safety-related valves had not been demonstrated. The licensee agreed to evaluate the setpoints and propose any necessary corrective actions.

The licensee provided the methodology for establishing setpoints in a latter dated July 30, 1983. In the staff's SER dated August 20, 1984, it was concluded that the licensee has developed a coherent methodology for establishing thermal-overload trip setpoints with all uncertainties resolved in favor of completing the safety-related valve action. The staff further concluded that the program, methods and schedule proposed in the licensee's letter of July 30, 1984, provide an acceptable resolution to IPSAR Section 4.14. This closes out this SEP issue.

## 2.12 Topic V-5, Reactor Coolant Pressure Boundary (RCPB) Leakage Detection (NUREG-0822, Section 4.16)

10 CFR 50 (GDC 30), as implemented by Regulatory Guide 1.45 and SRP Section 5.2.5, prescribes the type and sensitivity of systems and their seismic, indication, and testability criteria necessary to detect leakage of primary reactor coolant to the containment or to other interconnected systems. Regulatory Guide 1.45 recommends that at least three separate leak detection systems be installed in a nuclear power plant to detect unidentified leakage from the RCPB to the primary containment of 1 gpm within 1 hour. Leakage from identified sources must be isolated so that flow of this leakage may be monitored separately from

unidentified leakage. The detection systems should be capable of performing their functions after certain seismic events and of being checked in the control room. Of the three separate detection methods recommended, two of the methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be either monitoring the condensate flow rate from air coolers or monitoring airborne gaseous radioactivity. Other detection methods -- such as monitoring humidity, temperature, or pressure -- should be considered to be indirect indications of leakage to the containment. In addition, provisions should be made to monitor systems that interface with the RCPB for signs for intersystem leakage through methods such as monitoring radioactivity and water levels or flow.

2.12.1 Leakage Detection System (NUREG-0822, Section 4.16.1)

In IPSAR Section 4.16.1, the staff stated that Oyster Creek had only one of the detection systems (sump level monitoring) recommended in Regulatory Guide 1.45. The staff further stated that the plant had an airborne particulate and gaseous radiation monitoring system (APGRMS) installed in the drywell. This latter system is also recommended in the Regulatory Guide; however, the system had never been placed in operation at Oyster Creek because of problems. The APGRMS would be used to detect RCPB leakage indirectly by measuring the radioactivity in the drywell atmosphere which had come from the reactor coolant water leakage into the drywell.

The licensee committed to (1) identify the system modifications necessary to make the airborne particulate and gaseous radioactivity monitors operational, (2) evaluate the reliability and sensitivity of the existing leakage detection systems, and (3) propose a schedule for any necessary system modifications or procedural changes. The staff concluded in the IPSAR Section 4.16.1 that the licensee's proposed action was acceptable.

In its letter dated July 29, 1985, requesting deferment of the APGRMS to the Cycle 12R outage, the licensee stated that its evaluation of the APGRMS had revealed numerous problem areas requiring extensive redesign, modification or replacement of the system. The licensee was assessing various alternatives in order to arrive at a working system and stated that considering the extent of the remaining design work and projected delivery times for equipment it anticipated installation and testing of the APGRMS in the Cycle 12R outage.

Based on this, the licensee requested deferment of the installation and testing of an operating APGRMS to detect RCPB leakage to the Cycle 12R outage. The staff granted this deferment in its letter dated October 6, 1986.

In its letter dated July 8, 1986, the licensee described the adequacy of its sump monitoring system to detect RCPB leakage. The licensee concluded that this system's sensitivity is sufficient to allow safe shutdown before a crack would grow to an unstable length. Limiting conditions for operation and surveillance requirements on this system were incorporated in the Technical Specifications (TS) in Amendment No. 97 to the license dated January 6, 1986. Therefore, the licensee has evaluated the reliability and sensitivity of the existing sump detection system requested in IPSAR Section 4.16.1.

The licensee also stated in its letter dated July 8, 1986, that a new APGRMS would have to be designed, installed, and tested for Oyster Creek. It concludes, however, that the APGRMS would be of little use in quantifying leakage rates to meet TS leakage limits. The APGRMS would measure the leakage indirectly through released radioactivity and could only be used as a trending indication of the leakage which must be confirmed and quantified by other means. Therefore, the licensee concludes the APGRMS is of limited value and there are other data available as drywell pressure, humidity and temperature which can provide the information needed concerning RCPB leakage.

The licensee has identified the system modifications needed to make the APGRMS operational and has committed to install the system in the Cycle 12 outage. This completes the information requested from the licensee in IPSAR Section 4.16.1. Its request in its letter dated July 8, 1986, to cancel this commitment has been reviewed by the staff. In its letter of March 12, 1987, the staff concludes that the licensee has not provided sufficient justification to cancel its commitment to install the APGRMS. The licensee has not provided in detail the lack of sensitivity of the APGRMS, the cost of making the APGRMS operational and the sensitivity of other data as drywell pressure, humidity and temperature to provide data concerning RCPB leakage. Regulatory Guide 1.45 recommends at least three separate detection methods should be used including sump detection and an APGRMS. Therefore, the staff concludes that the APGRMS should be installed in the Cycle 12R outage. Region I will verify that the APGRMS is installed during the Cycle 12R outage.

## 2.13 Topic V-11.A, Requirements for Isolation of High- and Low-Pressure Systems (NUREG-0822, Section 4.19)

10 CFR 50.55a, as implemented by SPR Section 7.6 and BTP ICSB 3, requires that interlock systems important to safety be adequately designed to ensure their availability in the event of an accident. This includes those systems with direct interface with the reactor coolant system that have design pressure ratings lower than the reactor coolant system design pressure.

In IPSAR Section 4.19, the staff concluded that the pressure interlocks on the reactor water cleanup system do not satisfy current licensing criteria because they are not independent.

By letter dated August 4, 1983, the licensee submitted further information on the design features of the reactor water cleanup system that would prevent low pressure piping from being exposed to high pressure reactor coolant. In an evaluation issued by letter dated September 20, 1983, the staff concluded that the interlock logic, the diversity of signals and the relief valves provided reasonable assurance that the piping with design pressure ratings lower than the reactor coolant system design pressure will not be overpressurized. Therefore this issue is considered resolved.

## 2.14 Topic VI-4, Containment Isolation System (NUREG-0822, Section 4.22)

10 CFR 50 (GDC 54 through 57) require isolation provisions for lines penetrating reactor containment to maintain an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment.

In Section 4.22.2 of the IPSAR, the staff requested the licensee to evaluate the leakage detection provisions and operating stations of the remote manual valves in the core spray and containment spray systems. The subject valves are those listed in Section 4.22.2 of the IPSAR and V-21-15.

By letter dated August 27, 1985, the licensee provided a response to our request. In accordance with Section 4.22 of the IPSAR, the licensee performed an evaluation of the isolation provisions for the core spray and containment spray suction lines. The licensee's evaluation concluded the operating station for all valves in question is location in the 460V switchgear room which is accessible during post-accident conditions. In addition, should a failure occur in any of these lines outside containment which would require the system to be isolated, there are alarms and indications to alert the operator. These indications include pressure and flow indications, and sump pump operation alarms.

In the staff's Safety Evaluation dated August 20, 1986, the staff stated that they considered that the means to detect the need to isolate these lines are adequate and that the licensee has committed to revise plant procedures to include operator actions for line isolation before restart from the Cycle 11 refueling outage.

On this basis, the staff concluded that the proposed procedural revisions will ensure that the core spray and containment spray systems can be isolated when the need arises so as to provide containment integrity. Implementation will be verified as part of routine inspections.

Subsequent to the issuance of the staff's SE, Region I has advised that they have verified that the procedures have been implemented and this matter has been discussed in Region I Inspection Report 50-219/87-22. This issue is considered fully resolved.

## 2.15 Topic VII-1.A, Isolation of Reactor Protection System from Nonsafety Systems, Including Qualifications of Isolation Devices (NUREG-0822, Section 4.27)

10 CFR 50.55a(h) through IEEE Std. 279-1971 requires that safety signals be isolated from non-safety signals.

In Section 4.27(1) of the IPSAR, the staff concluded that insufficient isolation capability has been demonstrated between the nuclear flux monitoring system (intermediate range monitors [IRM] and average power range monitors [APRM]) and non-safety devices (process recorders and plant computer). The licensee agreed to perform a failure mode and effects analysis to evaluate the potential for common-mode electrical fault propagation. This analysis was submitted on August 3, 1984.

In a letter dated October 23, 1984 to the licensee, the staff stated that they reviewed the licensee's submittal and concluded that there is insufficient information to support the licensee's conclusion that the lack of qualified isolation devices would not compromise the integrity of the Reactor Protection System (RPS). Specifically, the following information or justification was not included in the licensee's submittal:

1. The evaluation did not address the resistor isolation buffer circuitry between the RPS and the process computer.

2. The evaluation concludes that the probability of maximum recorder input voltage being applied across the recorder input signal terminals (or R-18) is negligible. However, no justification is presented to support this conclusion.

3. The evaluation does not describe any periodic testing for stray voltages and system capability to withstand maximum credible voltages, as required by IEEE 279-1971 and IEEE 379-1977. In the absence of such testing, redundancy does not provide sufficient protection.

In letters dated July 8, 1985 and April 4, 1986, the licensee addressed the outstanding issues. The staff reviewed this information and in a letter dated November 10, 1987, the staff advised the licensee that we required additional information regarding (1) the isolation amplifier between the nuclear instrumentation analog signals and the multiplexer cabinet for the process computer and (2) the R105 IRM/APRM process recorder.

Upon receipt of the requested information from the licensee, the staff will review the information and will document the results of its review in a Supplement to the IPSAR.

## 2.16 Topic VII-1.B, Trip Uncertainty and Setpoint Analysis Review of Operating Data Base (NUREG-0822, Section 4.28)

10 CFR 50.36c.1.ii(A) requires that where limiting safety-system settings are specified for a variable on which a safety limit has been based, the setting should be chosen so that the automatic corrective action will correct the most severe abnormal event anticipated before a safety limit is exceeded.

In Section 4.28 of the IPSAR, the staff stated that, "the safety objective of this review was to ensure that margins between the allowable trip parameters and the actual RPS setpoints are adequate and properly identified."

"Sensors RE23A, B, C, and D (close main steam isolation valve on low steam pressure) have shown unacceptably high drift rates because of the large span (20-1,400 psig) compared with the actual limiting safety system setting (LSSS) setpoint of 825 psig. This setpoint should be changed to eliminate licensee event reports that result from 'drift'."

"Sensors RE18A, B, C, and D (autodepressurization on low low level) and Sensors RE02A, B, C, and D (core spray and isolation on low low reactor water level) have setpoints at the extreme low end of their ranges. These setpoints should be increased to a point where the margin to extreme range is at least equal to the instrument accuracy, or the sensors should be replaced with those having different ranges more suitable for the LSSS."

"A limited PRA to address the importance of setpoint drift and failure to function because of setpoints near the extreme range of instrument accuracy was performed. Risk assessments have shown that the issue of setpoint drift alone is of low importance to risk. This is because with monthly checking of the setpoints, drift might cause a trip outside the limiting safety system setting; however, drift is not large enough to cause a failure of the required function. When setpoints are set near the extreme range of instrument accuracy, as is the case for the Oyster Creek water level sensors (RE02 and RE18), it is possible that the drift could be large enough to prevent the function. The unavailabilities of the systems to meet specific demands for the two sensors in question were evaluated. There was a negligible increase in unavailability because of setpoint drift; rather, unavailability was controlled by other instrument failures. Also, for Oyster Creek the two sensors of concern do not contribute to the unavailability of any of the affected systems. For these reasons it is concluded that setpoint drift is of low importance to risk."

"However, the licensee has already committed for this and other reasons to install the General Electric (GE) analog trip system, which has been previously reviewed and approved by the staff in conjunction with the review of the GE Topical Report NEDO-21617, during the Cycle XI outage."

In the Region I Inspection Report No. 50-219/87-08, the staff states that the licensee has installed analog trip systems in place of sensors RE02A, B, C, and D. Because of concerns regarding Static O-ring (SOR) switches, to be installed (reference IE Bulletin 86-02), the replacement of other sensors with analog trip systems is being evaluated by the licensee.

Upon receipt of the licensee's evaluation, we will review it and report the results in a supplement to the IPSAR.

### 2.17 Topic VIII-3.B, DC Power System Bus Voltage Monitoring and Annunciation (NUREG-0822, Section 4.32)

10 CFR 50.55a(h), through IEEE Std. 279-1971 and 10 CFR 50 (GDC 2, 4, 5, 17, 18 and 19) as implemented by SRP 8.3.2, RG 1.6, 1.29, 1.32, 1.47, 1.75, 1.118, and

BTP ICSB-21, require that the control room operator be given timely indication of the status of batteries and their availability under accident conditions.

In Section 4.32 of the IPSAR, the staff stated that the licensee has committed to install alarms for the B and C battery breaker open, C battery charger open, and C battery ground in the control room. The staff concluded in the IPSAR that these alarms were acceptable and that with other battery indications listed above, the plant dc power system bus voltage monitoring and annunciation will meet current criteria. The licensee was to provide a schedule to complete these modifications.

By the licensee's letters dated November 16 and 29, 1982, it was stated that the necessary modifications would be completed by the end of the current Cycle 11 Refueling (Cycle 11R) outage and that, for an interim measure, there would be periodic inspections of the battery systems after the Cycle 10R outage. In its SE dated June 22, 1983, the staff, however, was concerned with the ability of the licensee to monitor the battery charging current with sufficient accuracy to assure that the battery has a low resistance connection to the bus. The staff noted that a current shunt that would provide for easy charging current measurement may be too large for full load operation. Therefore, the staff requested a description of how the battery connection integrity will be monitored by the instrumentation that will be part of the final modifications.

The licensee's responses to the staff's concern in the letter dated June 22, 1983, were its letters dated June 7, 1985 and April 4, 1986, and the meeting at the site on June 16 and 17, 1986, on the status of licensing actions. The meeting summary is dated August 1, 1986.

In the staff's SER dated December 16, 1986, the staff indicated that they had reviewed the information provided by the licensee and based on IPSAR Section 4.3.2 and the staff's SE dated June 1983, they concluded that the addition of the battery status alarms to be installed in the Cycle 11R outage are sufficient to have dc power system bus voltage monitoring and annunciation meet current criteria. They also indicated that based on the procedures provided by the licensee and its proposed check of the resistance through the breakers, the staff concluded that its concern is resolved.

In Region I Inspection Report No. 50-219/87-08, the staff indicated that with respect to the installation alarms for B and C battery breaker open, C battery charger open, and C battery ground, the inspector verified the functions identified are alarmed in the control room. The alarm annunciators do not always have the same designation as the function, however, a review of the alarm response procedures verified that the functions are in fact included in the alarm. (Also see Section 4.12.)

Based on the above, the staff considers this SEP issue closed.

## 2.18 Topic IX-5, Ventilation Systems (NUREG-0822, Section 4.34)

10 CFR 50 (GDC 4, 60 and 61), as implemented by SRP Sections 9.4.1, 9.4.2, 9.4.3, 9.4.4, and 9.4.5, requires that the ventilation systems shall have the capability to provide a safety environment for plant personnel and for engineered safety features.

### 2.18.1 Loss of Reactor Building Ventilation (NUREG-0822, Section 4.34(3))

The core spray and containment spray pumps are located in two corner rooms within the reactor building. These rooms do not have specific area ventilation systems. Therefore, the staff concluded in Section 4.34(3) of the IPSAR that the licensee should demonstrate that these pump motors are qualified for the conditions that could be expected in these rooms following a LOCA or make the appropriate plant modifications.

The licensee in a letter dated September 1, 1983, stated that the core spray and containment spray pump motors are designed to function in environments with temperatures of up to 185°F and 203°F, respectively. The licensee previously calculated the maximum post-LOCA temperature expected in the corner rooms without ventilation to be approximately 173°F in Amendment 42 to the Oyster Creek Unit No. 1 facility description and Safety Analysis Report.

The staff in a Safety Evaluation dated April 26, 1984, concluded that based on the thermal capability of the core spray and containment spray pump motors, compared to maximum calculated room, provisions for ventilation are not necessary. Therefore, this SEP issue is resolved.

2.18.2 Loss of Ventilation for Battery, Motor Generator Room and Switchgear Room (NUREG-0822, Section 4.34(4))

In Section 4.34(4) of the IPSAR, the staff found that both the B battery and motor generator room and the switchgear room ventilation systems are manually actuated from the control room by energizing a single relay. Transfer of this single control relay (Relay K) applies power to both the supply and exhaust fans in each room. Thus, a failure of that relay to transfer or loss of power to that relay would preclude electrical power to the fans of each room. The licensee agreed to evaluate the ventilation system design for the B battery and motor generator room and the consequences of a loss of ventilation in the switchgear room.

By letter dated August 21, 1984, the licensee provided the results of their evaluation of the B battery and motor generator room and switchgear room ventilation systems. The licensee also committed to install redundant relays to ensure adequate ventilation in these areas in the Cycle 11 refueling outage.

In the staff's Safety Evaluation dated July 1, 1985, the staff concluded that based on its review of the licensee's evaluation and the resulting commitment to install redundant relays for these ventilation systems, this SEP issue is considered resolved.

Region I will verify that the redundant relays have been installed.

## 3 TOPICS RESOLVED BY CHANGES TO PLANT TECHNICAL SPECIFICATIONS OR PROCEDURES

During the integrated assessment for Oyster Creek, a number of issues were resolved by commitments from the licensee to perform evaluations in order to determine whether modifications to plant Technical Specifications are warranted.

This section describes the actions taken regarding resolution of IPSAR issues involving Technical Specifications or procedural changes.

### 3.1 Topic II-3.B, Flooding Potential and Protection Requirements; Topic II-3.B.1, Capability of Operating Plants To Cope With Design-Basis Flooding Conditions; Topic II-3.C, Safety-Related Water Supply (Ultimate Heat Sink (UHS)) (NUREG-0822, Section 4.1)

10 CFR 50 (GDC 2), as implemented by SRP Sections 2.4.2, 2.4.5, 2.4.10, and 2.4.11 and Regulatory Guides 1.59 and 1.27, requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as flooding. The safety objective of these topics (II-3.B, II-3.B.1, and II-3.C) is to verify adequate operating procedures and/or system design provided to cope with the design-basis flood.

The site grade elevation is 23 ft mean sea level (MSL). During the staff's review of the hydrology-related topics, the following flooding elevations were identified, as defined by current licensing criteria:

probable maximum hurricane (PMH) - 22 ft MSL
probable maximum precipitation (PMP) - 23.5 ft MSL

As a result of these flooding levels, the staff identified the following issue:

#### 3.1.1 Makeup Isolation Condenser Water Sources (NUREG-0822, Section 4.1(4))

Because the makeup sources for the isolation condensers are susceptible to a single failure of flooding, the plant does not have a reliable means for maintaining a safe shutdown. In addition, the makeup water sources to the isolation condenser should be identified so that the best quality of water is available.

The licensee will evaluate the need for redundancy for the condensate transfer pump power supply. In addition, the plant Emergency Procedure for flooding will be revised to include the fire water storage tank as a redundant source of water supply to the emergency condenser. (The elevation of the fire water storage tank is above the PMH flood level.) The licensee has agreed to demonstrate that the minimum quantity of water maintained in the condensate storage tank and in the fire water storage tank is sufficient for long-term cooling, using either tank. The fire water storage tank is a backup to the condensate storage tank.

The licensee proposed to include minimum inventory of the water to be maintained in the condensate storage tank in the operating procedures and designate this as the primary source.

In its letters dated July 26, 1985, and April 21, 1986, the licensee stated that makeup to the isolation condensers is provided by the condensate storage tank and the Fire Water Storage Tank. These tanks can provide a volume of water of nearly 1 million gallons which should be sufficient to maintain the reactor in hot shutdown using the isolation condensers for 10 days. This time is sufficient to take corrective actions to restore submerged components. The tanks and the pumps are above the probable maximum hurricane flooding level (PMHFL) of 22 ft msl.

The licensee explained in the meeting of June 16 and 17, 1986, that its procedures require a minimum of 20 feet or 250,000 gallons in the condensate storage tank (CST). The intake tour sheet requires a minimum of 350,000 gallons in the Fire Water Storage Tank (FWST). The high wind conditions for emergency procedure 2000-ABN-3200.31 are the following: (1) tornado watch or warning, (2) hurricane watch or warning, (3) tornado funnel cloud in the area and (4) sustained wind speeds greater than 74 mph. This procedure requires the CST to be filled to 43 feet or 537,500 gallons and the isolation condensers to be filled (50,000 gallons). The licensee stated that it could, if needed, bring in a fire truck and pump water into the isolation condensers using an alternate connection from the fire water main. This is in the staff's SE dated November 28, 1986. The meeting summary is dated August 1, 1986.

This completes the staff review of this SEP Topic issue. The evaluation of the fact that the condensate transfer pumps are located outside plant and susceptible to damage to tornado missiles is discussed in Section 2. The procedures will be verified by Region I as discussed in Section 4.1.1.

3.1.2 Low Water Level in the Intake Structure (NUREG-0822, Section 4.1(5)

Section 4.1(5) of the IPSAR stated that a Technical Specification change was under review that would allow the operator to stop operation of the dilution pump when level is low and thus raise the intake canal water level. The dilution pumps were required to continue running under certain conditions to maintain canal water temperature within limits. This Technical Specification change along with a water level gage (with remote readout) (Item 3) in the intake canal, will enable the operator to respond in a timely manner to the low water level in the canal.

In Amendment 66, issued on March 24, 1983, the Technical Specifications related to water quality were removed from the Oyster Creek license; therefore the intent of the above change is met. However, in a letter dated April 21, 1986, the licensee requested to cancel its commitment to install an automatic water level gage in the intake canal with a remote readout in the control room. As stated in Section 2.1 of this supplement, the licensee proposed revising station procedure 2000-ABN-3200.31, High Winds, to require a plant shutdown when the water level at the intake structure cannot be verified to be less than elevation 4.5 ft msl.

In a letter dated November 28, 1986, we advised the licensee that its submittal of April 21, 1986, did not address the use of this instrumentation for measuring the water level at the intake structure to determine if it is near or below the service water pump suction elevation. We also stated that this was discussed in Section 4.1(5) of the IPSAR and requested the licensee to provide a discussion on the administrative controls to monitor the canal water for low water level near or below the service water pump suction elevation and the actions to be taken by the control room operators.

The licensee in a letter dated November 6, 1987, provided the requested information. The staff is presently evaluating this information. Upon completion of our review, we will report the results of our review in a supplement to the IPSAR.

3.2 Topic V-5, Reactor Coolant Pressure Boundary (RCPB) Leakage Detection Operability Requirements (NUREG-0822, Section 4.16.2)

In IPSAR Section 4.16.2 we stated that the Oyster Creek Technical Specifications do not contain limiting conditions for operation or surveillance requirements regarding the leakage detection systems, as recommended by Regulatory Guide 1.45 and the BWR Standard Technical Specifications (NUREG-0123).

We also stated that in conjunction with the procedural changes described in Section 4.16.1, the licensee also committed to provide the appropriate action requirements in the Technical Specifications for inoperable leakage detection systems (i.e., an inability to measure leakage) and any necessary procedural changes to provide surveillance and testing commensurate with the required sensitivity.

By letter dated August 23, 1985, which superseded its letter dated October 22, 1984, the licensee requested an amendment to the Oyster Creek Technical Specifications. This amendment would revise the limiting conditions for operation and add surveillance requirements for the reactor coolant system leakage.

In its letter of January 6, 1986, the staff issued an Amendment to the Oyster Creek Provisional Operating License which authorized changes to the Technical Specifications, to revise the limiting conditions for operation and to add surveillance requirements for reactor coolant system leakage. In its letter the staff found that the proposed specifications regarding the coolant leakage and leakage detection systems to be more restrictive than the current specifications; and with one exception, consistent with BWR Standard Technical Specifications. The exception is that the proposed specifications do not include a Limiting Condition for Operation (LCO) on pressure boundary as in the BWR Standard Technical Specifications. The staff recommended that the licensee submit an additional license amendment application which includes an LCO on pressure boundary leakage in TS3.3.D. The staff also advised the licensee that its letter closes out the staff's actions on Section 4.16.2 of the integrated Plant Safety Assessment Report, NUREG-0822 dated January 1983, for Oyster Creek.

Office of Inspection and Enforcement Inspection Report No. 50-219/87-08 identified inadequate procedural implementation of the technical specification. Resolution of this aspect of the item is discussed in Section 4.3.1.

On March 17, 1987, the licensee submitted a Technical Specification Change Request which would limit the unidentified leakage for the reactor coolant system to a maximum leak rate increase of 2 gpm within any 24 hour period while operating at steady state power. The staff is presently reviewing the licensee's request independently of the SEP.

3.3 Topic V-6, Reactor Vessel Integrity (NUREG-0822, Section 4.17)

Appendices G and H to 10 CFR 50 and 10 CFR 50-55(g) as implemented through R.G. 1.99 require that reactor vessel integrity be ensured by review of aspects such as fracture toughness, surveillance programs, and neutron irradiation.

In Section 4.17 of the IPSAR, the staff determined that the licensee should submit a plan for the capsule exposure schedule and how the test results will be used to modify operation (e.g., setting nil ductility temperature (NDT) limits.)

By letter dated March 10, 1983, the licensee provided information regarding the reactor vessel material surveillance program at Oyster Creek. That letter indicated that the No. 2 material capsule would be removed during the cycle X refueling outage and testing and analysis results would be provided to the NRC. In addition the licensee provided the schedule for removal of the next capsule.

In a letter dated April 28, 1983, the staff determined that the licensee's letter of March 10, 1983, constitutes an acceptable response to the issue identified in Section 4.17 of the Oyster Creek Integrated Plant Safety Assessment Report (NUREG-0822). The staff also concluded that staff review of the capsule test results will be conducted as a routine operating reactor action independently of the SEP. The licensee response is considered sufficient to close out this IPSAR section.

3.4 Topic V-12.A, Water Purity of BWR Primary Coolant (NUREG-0822, Section 4.20)

10 CFR 50 (GDC14), as implemented by R.G. 1.56 requires that the reactor coolant boundary (RCPB) have minimal probability of propogationing failure. This includes corrosion-induced failures from impurities in the reactor coolant system.

In Section 4.20 of the IPSAR, the staff concluded that the licensee should provide Technical Specification changes to incorporate pH level conductivity and chloride limits and implement a time-related conductivity limit in operating procedures. The licensee agreed to complete these actions before startup from the cycle 10 outage.

As a result, the licensee in a letter dated September 18, 1984, proposed to revise the Technical Specification for chlorides and conductivity to be consistent with R.G. 1.56

The staff concluded in its Safety Evaluation dated November 21, 1985 that the proposed TS changes regarding reactor water conductivity and chloride concentration limits meet the limits and appropriate corrective actions in R.G. 1.56 and are therefore acceptable. Based on this, the staff issued

Amendment No. 93 to Provisional Operating License No. OPR-16 for Oyster Creek Nuclear Generating Station on November 21, 1985. This amendment authorizes changes to the Oyster Creek A Technical Specifications (TS) to incorporate the additional restrictions on conductivity and chloride limits in Section 3.3.E Reactor Coolant Quality and revise its Basis. However in its Safety Evaluation the staff also concluded that the application to amend the TS did not address the guideline in footnote "a" of Table 1 of R.G. 1.56 that states the total time for all incidents exceeding the acceptable reactor water chemistry limits in Table 1 should not exceed 2 weeks per year. The staff considers this restriction on plant operation to be a necessary part of method, described in R.G. 1.56 and acceptable to the staff, for implementing the criteria in General Design Criterion 14 with regard to minimizing the probability of corrosion induced failure of the reactor coolant boundary in boiling water reactors (BWRs). This restriction is in the Standard Technical Specification for BWR's (NUREG-0123). On this basis the staff requested the licensee to propose appropriate TS to incorporate such a restriction in the TS, or provide a justification that such a TS is not needed.

At a meeting on February 6, 1987, the licensee stated that the controls on the reactor coolant quality in Specification No. 1302-28-001, Revision 2, provided the additional requirement requested by the staff in its letter issuing Amendment 93, dated November 21, 1985. This specification restricts Oyster Creek Operation when the limit is exceeded 2 weeks in any consecutive 12 month period.

In Region 1 Inspection Report 50-219/87-08 it is stated that the inspector verified that the time-related conductivity limit, the chloride concentration limit, and the pH limit for reactor coolant specified in Specification No. SP-1302-28-001 have been incorporated into Station Procedure 827.1, Primary System analysis; Reactor Water. Also license Amendment No. 93 incorporates into the Technical Specifications the chloride and conductivity limit established in Regulatory Guide 1.56. The implementation of this Technical Specification has been verified by IE inspection (Inspection Report No. 50-219/87-08) as discussed in Section 4.5. Therefore, this SEP issue is fully resolved.

3.5 Topic VI-7A.3, Emergency Cooling System Actuation (NUREG-0822, Section 4.21)

10 CFR 50.55a(h), as implemented by IEEE-Std. 279-1971, and 10 CFR 50 (GDC 37) as implemented by R.G. 1.22 requires that equipment important to safety be tested periodically at power.

In Section 4.23 of the IPSAR the staff concluded that testing of the logic trains and associated components of the emergency condenser should be included in the facility Technical Specifications.

The licensee by letter datec June 4, 1984, provided the results of its review of the Oyster Creek Technical Specifications (TS). The licensee has concluded that modifications of the existing TS concerning the testing of the emergency condenser logic trains is not warranted.

The plant parameters that actuate the emergency condenser are reactor high pressure and low-low reactor water level; the testing and calibration of these instrument channels is included in TS Table 4.1.1 Similarly, the instrument channels that detect and isolate an emergency condenser line break (high-flow-differential pressure) are also included in TS Table 4.1.1.

Section 4.8 and Table 4.1.2 of the TS require a test of the emergency condenser actuation and isolation trip system at each refueling outage. This frequency is consistent with that required for other protective instrumentation and trip systems at Oyster Creek.

In addition the licensee is implementing a Reference Index which will cross-reference the TS surveillance requirements.

The above is acceptable to the staff and was documented in the staff's Safety Evaluation dated July 1, 1985. This closes out this SEP issue.

3.6 Topic VI-10A, Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing (NUREG-0822, Section 4.26)

10 CFR 50 (GDC-2) requires that the reactor protection system be designed to permit periodic testing of its functioning, including a capability to test channels independently.

In Section 4.26.2 of the IPSAR, the staff found that the reactor mode switch and some instruments are not specifically identified for testing in the Technical Specification (TS). The licensee stated that plant procedures require testing of all redundant instrumentation required for safety. However, the licensee agreed to review plant surveillance procedures to ensure that all safety logic channels tied to the reactor mode switch are surveyed.

In addition, Technical Specifications would be made to incorporate the required testing prior to startup from the Cycle 10 outage.

By letter dated May 31, 1984, the licensee provided the results of its review of protective instrumentation testing required by either plant procedure or Technical Specifications.

The licensee noted that although an explicit test of the reactor mode switch is not specified in the TS, the switch is tested in various positions when other logic channels are tested. Attachment 1 of of the May 31, 1984, submitted shows for each contact in the switch, tests that are performed, associated TS requirements and which position (Run, Shutdown, etc) the switch is in. In the staffs Safety Evaluation dated July 15, 1985, and Inspection Report No.50/219/87-08 related to this matter, the staff concluded that the testing being performed is sufficient to test the functioning of the mode switch.

The staff also stated that the licensee has developed a cross-reference indexing system between the Technical Specifications and Plant surveillance procedures. This index will show the correspondence between the testing of instrumentation and logic channels and trip systems by the surveillance procedures and the TS requirements. On this basis the staff concluded that the existing TS requirements, as supplemented by the cross-reference indexing system between the TS and plant procedures developed by GPU Nuclear, should be adequate to assure needed testing is performed. However, the staff requested GPU Nuclear to submit the index to the staff.

On October 22, 1985, the cross-reference index and surveillance procedure were submitted.

In a letter dated January 17, 1986, the staff advised the licensee that they reviewed the index and noted that it contains tables listing the TS requirements for surveillance, the actual frequency as specified in the procedure, the procedure number and which schedule the test is on. In its letter, the staff stated that based on review of the October 22, 1985 submittal, the staff concludes that the indexing system is sufficiently detailed and complete to satisfy the issue raised by the staff in the IPSAR review. Therefore, the staff's review of IPSAR Section 4.26.2 is complete. This closes out this SEP item.

### 3.7 Topic IX-5, Ventilation Systems (NUREG-0822, Section 4.34)

10 CFR 50 (GDC 4, 60, and 61), as implemented by SRP Sections 9.4.1, 9.4.2, 9.4.3, 9.4.4, and 9.4.5, requires that the ventilation systems shall have the capability to provide a safe environment for plant personnel and for engineered safety features.

#### 3.7.1 Restoration of Ventilation (NUREG-0882, Section 4.34(1))

The licensee will provide a submittal in March, 1988 as discussed in Section 4.13.

### 3.8 Topic XV-16, Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment (NUREG-0822, Section 4.36)

10 CFR 100, as implemented by SRP 15.6.2 requires that the radiological consequences of failure of small lines carrying primary coolant outside containment be limited to small fractions of the exposure guidelines of 10 CFR 100.

In Section 4.36 of the IPSAR, the staff concluded that the Technical Specifications should be modified to include the BWR Standard Technical Specification (NUREG-0123) reactor coolant activity limits, sampling frequencies and action requirements.

By letter dated October 22, 1984, the licensee proposed a revised Technical Specification for Oyster Creek for primary coolant activity. We reviewed the license submitted and requested additional information concerning (1) the definition for Dose Equivalent Iodine-131, (2) limits for non-iodine radioactivity in the reactor coolant as shown in Standard Technical Specifications for General Electric Boiling Water Reactors (NUREG-0123), (3) Limiting Condition for operation as shown in NUREG-0123 and (4) annual reporting requirement for radioiodine spiking as shown in NRC Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes."

In a letter dated October 23, 1986, the licensee provided the requested information and proposed a Technical Specification change. The staff reviewed this information and in a meeting of June 30, 1987, we advised the licensee to revise its proposed Technical Specification in regard to wording of definitions and sampling frequencies.

The licensee is presently revising the proposed Technical Specification in accordance with our requirements. Upon receipt of the licensee's submittal, the staff will review the information and will document the results of its review in a Supplement to the IPSAR.

## 3.9 Topic XV-19, Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Pipe Breaks Within the Reactor Coolant Pressure Boundary (NUREG-0822, Section 4.38)

10 CFR 100, as implemented by SRP 15.6.5, requires that the radiological consequences of a design basis loss-of-coolant accident be limited to the exposure guidelines for both the 0- to 2-hr exclusion area boundary and the 30-day low population zone (LPZ) boundary.

In Section 4.38 of the IPSAR, the staff found that the major contribution to the 30-day LPZ exposure was from main steam live isolation valve (MSIV) leakage. Therefore, the staff required that the licensee develop and implement a preventive maintenance program to limit MSIV leakage or to justify the existing program based on the results of testing experience for the valves.

The licensee proposed to review existing maintenance practices and those of other BWR facilities, identify any necessary corrective actions and upgrade the maintenance program, if necessary, before startup from the Cycle XI refueling outage. The licensee has responded to IPSAR Section 4.38 in its submittals dated May 18, 1984 and September 12, 1985. The staff has evaluated the information provided by the licensee, and our evaluation is presented in the staff's SE dated May 22, 1986. During the NRC Region I Inspection 50-219/86-04, the Oyster Creek MSIV leakage test results and maintenance history were reviewed by the Region to determine the extent of leakage experienced at Oyster Creek and the effectiveness of the licensee's maintenance program. During this inspection The Region determined that MSIV leakage has not been excessive - generally less than 100 scfh. Available test data showed that on only one occasion, in 1982, did one valve leak in excess of 100 scfh. Leakage data from five outages was reviewed for the period 1977 through 1983-84 outage and shows that on only two occasions did two valves in series not meet this acceptance criteria. This was in 1978 and 1982. Here the maximum leakage through any single penetration would have been 14.13 and 22.9 scfh.

Also, based on the review of MSIV maintenance data, the Region determined that the preventive and corrective maintenance being performed has been effective in maintaining the valves' performance. The licensee's maintenance program for these valves includes input from both General Electric and the valve manufacturers, Atwood and Morrill. The licensee's routine preventive maintenance program for MSIVs calls for two valves to be rebuilt each refueling outage and the other two valves to be repacked. The licensee is continuing discussions with General Electric and the valve manufacturers to ensure that repair methods are kept up-to-date.

In the staffs SE dated May 22, 1986, we conclude that the licensee has developed, implemented, and is keeping up-to-date a maintenance program adequate to maintain the MSVIs in an acceptable condition. Therefore, the staff concludes that the issue in IPSAR Section 4.38 is satisfactorily resolved.

## 4 IPSAR TOPIC RESOLUTIONS CONFIRMED BY NRC REGION I OFFICE

During the integrated assessment for Oyster Creek, a number of issues were resolved by commitments made by the licensee for specific plant modifications or procedural changes. After IPSAR for Oyster Creek was issued, the Region I office was asked through Task Interface Agreement 83 to verify that plant modifications had been implemented and to review changes to plant operating procedures made by the licensee. Table 4.1 provides a list of IPSAR actions for which confirmation by the Region I office was requested.

Region I personnel conducted onsite inspections for each item identified in Table 4.1. The inspections consisted of examinations of installed equipment as well as a review of supporting procedures and other documentation. The Region I office concluded that the licensee had met the commitments documented in the IPSAR for the items in Table 4.1. Inspection findings with the results of the review are documented in inspection reports as noted in the following sections.

### 4.1 Topics II-3.B, Flooding Potential and Protection Requirements; II-3.B.1, Capability of Operating Plants to Cope With Design-Basis Flooding Conditions; and II-3.C, Safety-Related Water Supply (Ultimate Heat Sink (UHS) (NUREG-0822, Section 4.1)

#### 4.1.1 Isolation Condenser Flooding (NUREG-0822, Section 4.1(4))

IPSAR Section 4.1(4) requires procedural revisions to include the fire water storage tank as a redundant source of water supply to the emergency condenser, and include in operating procedures a minimum inventory of water to be maintained in the condensate storage tank.

The licensee has several procedures which specify actions associated with emergency condenser water supplies. These are identified as follows:

-- Procedure 307, Isolation Condenser System. This procedure states, in relation to filling the isolation condenser, "In emergency situations fire protection shall be used if condensate transfer is not available." Also, the procedure provides instruction for makeup to the isolatior . ndenser from the fire protection system.

-- Procedure 316. Condensate System, specifies maintaining 20' (250,000 gallons) of water in the condensate storage system.

-- Procedure 333, Plant Fire Protection System, specifies maintaining equal to or greater than 310,000 gallons in the fire water storage tank.

-- Procedure 7000-ABN-3200.31, High Winds, specifies certain actions to be taken at specific sea water levels. Among these actions are filling the isolation condenser to the high level alarm (7.7') and filling the condensate storage tank to the high level alarm (43').

The above is reported in Inspection Report No. 50-219/86-38.

These provisions are acceptable, however full resolution of this issue is dependent on the resolution of related IPSAR item 4.1(1) (Section 2.1.1). See related discussions in Sections 2.1.1 and 2.5.3.

4.1.2 Hurricane Flooding of Pumps (NUREG-0822, Section 4.1(6))

IPSAR Section 4.1(6) indicates that the licensee proposed to update Emergency Procedures and to identify the alternate water sources and flow paths if the intake structure becomes flooded, and to identify the priority of water sources and flow paths to be used to ensure a safe shutdown.

The licensee has provided procedural instructions in Station Procedure 2000-ABN-[illegible]0.31, High Winds, for the actions to be taken in the event of high water [illegible] the intake structure. The instructions include actions to be [illegible] ing down the circulating water pumps and the service water pumps. [illegible] cation Procedure 307, Isolation Condenser System, provides instructions for providing makeup to the isolation condenser using the fire protection system should the preferred condensate transfer system not be available.

The above is reported in Inspection Report No. 50-219/87-04.

This issue is considered to be fully resolved.

4.1.3 Roof Flooding (NUREG-0822, Section 4.1(9))

IPSAR Section 4.1(9) states that the licensee is to drill holes in the parapets and install scuppers to preclude the potential for buildup of rain-water on the roof of either the reactor building or turbine building.

The resident inspector verified that holes have been provided and scuppers installed in the reactor and turbine building parapets.

This is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

4.2 Topic III-3.C, Inservice Inspection of Water Control Structures (NUREG-0822, Section 4.5)

4.2.1 Intake Structure Trash Racks and Intake Screens (NUREG-0822, Section 4.5.2)

IPSAR Section 4.5.2 states that the licensee agreed to formalize as part of shift turnover procedures the shift inspection of the intake structure, and to modify the screen wash system to prevent buildup of sea lettuce.

The licensee has in place an intake area tour sheet which is required to be completed each shift. Also, SDD-OC-533 Div II (Budget Activity #402188) describes the modifications performed on the screen wash system during the Cycle 10 refueling outage to prevent buildup of sea lettuce.

The above is reported in Inspection Report No. 50-219/87-04.

This issue is considered fully resolved.

4.2.2 Inspection Program (NUREG-0822, Section 4.5.4)

IPSAR Section 4.5.4 states that the licensee to provide an inspection program which includes review by qualified engineering personnel of water control structures. Also, to establish inspection and documentation of water control structures following extreme events.

The licensee has in place four procedures which deal with the inspection of water control structures. Procedures 9410-SUR-4512.09, OCNGS Non-Radiological Environmental Surveillance, provides for a monthly or following severe storms inspection of intake and discharge canal banks. Procedure 9410-SUR-4570.01, Oyster Creek/Forked River Hydrographic Surveying, provides for an annual hydrographic survey of the Oyster Creek and Forked River waterways which serve as discharge and intake waterways for the plant. Procedure 9430-SUR-4550.01, Oyster Creek/Forked River Environmental Engineering Survey, provides for an annual environmental engineering surveillance of the Oyster Creek intake and discharge waterways east of Route 9. Also, Procedure 2000-ABN-3200.31, High Winds, provides for the inspection of the intake structure following the existence of high wind conditions. These procedures appear to satisfy the water control structure inspection requirements.

The above is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

## 4.3 Topic V-5, Reactor Coolant Pressure Boundary (RCPB) Leakage Detection (NUREG-0822, Section 4.16)

### 4.3.1 Operability Requirements (NUREG-0822, Section 4.16.2)

IPSAR Section 4.16.2 requires the licensee to provide the appropriate action requirements in the Technical Specification (TS) for inoperable leakage detection systems and any necessary procedural changes to provide surveillance and testing commensurate with the required sensitivity.

The licensee submitted the necessary Technical Specification Change Request which resulted in the issuance of License Amendment No. 97 on January 1986. This amendment provided the limiting conditions for operation and added surveillance requirements for reactor coolant leakage detection system.

The licensee normally provides instructions for performing TS required surveillance tests of this type in 600-series procedures. These procedures satisfy the requirements of the TS and Regulatory Guide 1.33, which require implementing procedures for each surveillance test listed in the TS. During the review of this item, it was determined that no 600-series procedure had been prepared to perform the surveillance listed in TS 4.3.H for channel calibration of the primary containment sump flow integrator and the primary containment equipment drain tank flow integrator. This failure to provide a surveillance test implementing procedure is contrary to the requirments of TS 6.8.1 and Regulatory Guide 1.33, which require implementing procedures for each surveillance calibration listed in the TS, and is considered to be a violation (219/87-08-01).

Records show calibrations had been performed on these instruments in July 1985 and July 1986. These calibrations were performed in accordance with a Technical Specification Supporting Installed Instrumentation List (TSSIIL) procedure. Calibrations performed in accordance with this TSSIIL do not have a detailed implementing procedure nor do they have the same documentation and review requirements as do 600-series surveillance procedures. For the 1985 and 1986 tests, TSSIIL calibration data sheets were available. However, without the benefit of an implementing procedure, it took plant engineers several hours, including talking to the technician who performed the test, to determine how the calibration was conducted.

During the July 1986 calibration, the drywell sump leak rate counter was found to be defective. A Maintenance and Construction Short Form was initiated and the counter was replaced and tested satisfactorily. Following the identification of the failed counter on July 3, 1986, a Deviation Report (86-287) was prepared on the same day. A Plant Engineering Work Request was initiated on July 17, 1986 to review the effect of the drywell sump flow counter error on leak rate calculations. A Plant Engineering Task Assignment (PETA) 86-141 was prepared on July 29, 1986 to perform this evaluation. A Responsible Technical Review of the Deviation Report was performed on August 1, 1986 and the PETA was completed on October 9, 1986. This review determined the TS limit of $\leq$ 5.0 gpm unidentified leak rate was not exceeded due to the as-found counter error.

The actions associated with the failure to prepare the required surveillance test procedure were also reviewed. This review determined that following the preparation of the Technical Specification Change Request associated with the leak rate instrumentation a Licensing Action Item (LAI) was written on January 7, 1985. This LAI 84179.01 assigned Plant Engineering with the responsibility of preparing the necessary administrative controls, surveillance, etc. In response to this LAI, Plant Engineering identified the actions which had been taken. These actions included the assignment of a PETA (85-244) to the I & C group to write a calibration procedure for the drywell equipment drain tank flow integrator and for the drywell sump flow integrator. This PETA was written on January 23, 1985 and specifically identified the task scope as writing a 600-series procedure for calibration of the drywell sump flow integrator and drywell equipment drain tank flow integrator. Another LAI, 84179.03, was written on January 22, 1986 following the issuance of TS Amendment 97 to ensure procedural compliance with the amendment.

This PETA (85-244), two years after its preparation, is still open. The failure to complete this PETA in a timely manner was discussed in detail with the licensee. The inspector was of the opinion that the failure to provide a required 600-series surveillance procedure resulted from an improper prioritization of the PETA. Licensee representatives stated that at the time the PETA was prioritized, some personnel considered the performance of the surveillance testing in accordance with the TSSIIL program to be satisfactory to meet the TS requirements. They felt the failure to prepare a required surveillance procedure was not due to improper prioritization but was the fault of the TSSIIL procedure which did not clearly establish that the TSSIIL program is not to be used for the performance of surveillance tests listed in the TS.

The licensee further agreed that proper prioritization of PETAs is important and that following the identification of this incident, a review of PETA prioritizations had been undertaken.

The above is reported in Inspection Report No. 50-219/87-08.

In a letter dated June 1, 1987, responding to Inspection Report No. 50-219/87-08 and the deficiency identified therein related to this topic, the licensee committed to issue the needed procedure prior to the next scheduled surveillance. Region I will verify the implementation to resolve this item.

## 4.4 Topic V-10.B, Residual Heat Removal System Reliability (NUREG-0822, Section 4.18)

IPSAR Section 4.18 indicated that the licensee agreed to implement generic guidelines for emergency procedures.

The licensee has replaced the previously existing emergency procedures with General Emergency Operating Procedures developed in conjunction with the BWR Owners Group and the TMI Action Plan requirements.

The above is reported in Inspection Report No. 50-219/87-04.

This provision is considered to be acceptable; however specifics of the procedural resolution could be affected by the resolution of NUREG-0822 items 4.1(1), 4.1(4), 4.6.4 and 4.30 (Sections 3.1.1, 4.1.1, 3.4.3 and 4.10 respectively).

## 4.5 Topic V-12.A, Water Purity of BWR Primary Coolant (NUREG-0822, Section 4.20)

IPSAR Section 4.20 indicates that the licensee is to incorporate into plant procedures the time-related conductivity limit, the chloride concentration limit, and the pH limit for reactor coolant referenced in "BWR Water Quality Specification" (Specification No. SP-1302-28-001). Also, the licensee is to incorporate the conductivity and chloride limits in Regulatory Guide 1.56 into the facility Technical Specifications.

The inspector verified that the time-related conductivity limit, the chloride concentration limit, and the pH limit for reactor coolant specified in Specification No. SP-1302-28-001 have been incorporated into Station Procedure 827.1, Primary System Analysis; Reactor Water. Also, License Amendment No. 93 incorporated into the Technical Specifications the chloride and conductivity limits established in Regulatory Guide 1.56.

The above is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

## 4.6 Topic VI-1, Organic Materials and Postaccident Chemisty (NUREG-0822 Section 4.21)

### 4.6.1 Organic Materials (NUREG-0822, Section 4.21.1)

IPSAR Section 4.21.1 stated that the licensee is to ascertain the chemical composition of the existing drywell coatings. If these coatings are found to contain hydrocarbons, they should be removed or the licensee should submit an evaluation to justify the continued use of these coatings.

By letter dated February 10, 1984, the licensee provided a description and results of the drywell inspection conducted during the Cycle 10 refueling outage. The licensee concluded that the chemical composition is satisfactory. The torus interior was coated during the Cycle 10 refueling outage.

The above is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

## 4.7 Topic VI-4, Containment Isolation System (NUREG-0822, Section 4.22)

### 4.7.1 Locked-Closed Valves (NUREG 0822, Section 4.22.1)

IPSAR Section 4.22.1 identified 31 valves which are either test, vent, drain, or sample line manual isolation valves that connect to piping penetrating the containment. The licensee was to provide administrative procedures to ensure these valves are locked closed. Two of the valves on the list, V-14-21 and V-14-39, should have been V-14-20 and V-14-40, respectively. Also, V-17-51 has been replaced with V-17-51 due to the PASS system addition.

The inspector verified that all valves listed have been included in the Containment System Integrity Valve Check Off List of Procedure No. 312, Reactor Containment Integrity and Atmosphere Control, or on the valve check off list for Procedure No. 305, Shutdown Cooling System Operation. All valves identified in Item 4.22.1 are required by these lists to be locked closed. The inspector verified the Containment System Integrity Valve Check Off List had been last performed in November 1986 prior to the startup following the last refueling.

The above is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

## 4.8 Topic VI-7.C.1, Appendix K-Electrical Instrumentation and Control (NUREG-0822, Section 4.25)

10 CFR 50 (GDC-17) as implemented by R.G. 1.6 an IEEE Std 308-1974, requires that onsite electrical power supplies and their onsite distribution systems shall have sufficient independence to perform this safety function assuming a single failure.

In Section 4.25 of the IPSAR, the staff noted that there are seven automatic transfers of load or load groups between redundant sources in the ac system.

The licensee agreed to perform a coordinated load and circuit breaker analysis to establish any corrective actions necessary to preclude automatic transfer of faults between redundant power sources.

By letter dated September 1, 1983, the licensee submitted an analysis of the coordination of protective devices for fault current interruption. In some cases, a fault could be transferred to a redundant safety bus. Therefore, by letter dated July 30, 1984, the licensee proposed to replace the timing devices in these breakers so that the fault interrupt currents are so coordinated that the fault cannot be transferred.

In the staff's Safety Evaluation dated November 16, 1984, the staff concluded that the relay coordination (i.e., time response characteristics) provides sufficient independence between redundant electrical divisions to proclude the automatic transfer of faulted loads between those devisions. Therefore, the staff also found that the licenses's proposal to replace the two relays with overlapping time response characteristics acceptable. The staff also agreed with the licensees amended schedule as discussed in their letter of October 25, 1984 to provide for installation of the new relays at the first outage of five or more days after reciept of the necessary parts. Region I will verify that the proper relays have been installed. Upon verification of this installation of the relays, this SER issue is considered resolved.

4.9 Topic VII-1.A, Isolation of Reactor Protection System From Nonsafety Systems, Including Qualifications of Isolation Devices (NUREG-0822, Section 4.27)

4.9.1 Flux Monitoring Isolation (NUREG-0822, Section 4.27.1)

IPSAR Section 4.27.1 indicated that the licensee is to install Class 1E protection at the interface between the reactor protection system power supply and the reactor protection system.

A completed licensing action item documents the installation of six electrical protection assemblies, qualified to 1E requirements between the reactor protection system motor-generator sets 1-1, 1-2, and auxiliary transformer and protection system panel No. 1 and Panel No. 2. This modification was completed during Cycle 11R under B/A 402032.

The above is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

4.10 Topic VII-3, Systems Required for Safe Shutdown (NUREG-0822, Section 4.30)

This item references NUREG-0822 items 4.1(4), 4.1(6), and 4.18. Items 4.1(6) and 4.18 are resolved, as discussed above. Full resolutions of this issue is dependent upon the resolution of item 4.1(1) which is indirectly referenced through item 4.1(4). See related discussions of NUREG-0822 items 4.1(1), 4.1(4), and 4.6.4 in Sections 3.1.1, 4.1.1, and 4.10.

4.11 Topic VIII-2, Onsite Emergency Power Systems (Diesel Generator) (NUREG-0822, Section 4.31)

4.11.1 Diesel Generator Annunciators (NUREG-0822, Section 4.31(1))

IPSAR Section 4.31(1) stated that the licensee agreed to make certain modifications to the diesel generator annunciators.

Licensee documentation shows that certain modifications to the diesel generator annunciators were performed under modication E.T. 312-78 to satisfy the commitments made to the NRC. These modifications were:

1. Removing existing non-disabling alarms from the present diesel generator trouble alarm.

2. Providing a new annunciator for the manual mode switch not in auto.

3. Re-designating the working of the annunciator windows to reflect the conditions more clearly.

4. Providing a low battery voltage sensor with an alarm function indicating diesel generator DC failure.

The above is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

4.11.2 Diesel Generator Trip Bypass (NUREG-0822, Section 4.31(2))

By Letter dated November 16, 1982, the licensee committed to modify diesel generator trips (bypass of lead voltage-ampere reactive (VAR) and reverse power trips) by the end of the cycle 11 outage. The staff concluded that the modifications would be acceptable when completed and verified. Region I will verify this implementation to resolve this item.

4.12 Topic VIII-3.B, DC Power System Bus Voltage Monitoring and Annunciation (NUREG-0822, Section 4.32)

IPSAR Section 4.32 stated that the licensee committed to install alarms for B and C battery breaker open, C battery charger open, and C battery ground.

The inspector verified the functions identified are alarmed in the control room. The alarm annunciators do not always have the same designation as the function, however, a review of the alarm response procedures verified that the functions are in fact included in the alarm.

The above is reported in Inspection Report No. 50-219/87-08.

This issue is considered to be fully resolved.

4.13 Topic IX-5, Ventilation Systems (NUREG-0822, Section 4.34)

4.13.1 Restoration of Ventilation (NUREG-0822, Section 4.34(1))

IPSAR Section 4.34(1) states that the licensee is to review and modify as required the loss of offsite power procedure to ensure that operation of ventilation systems is adequately addressed and will not overload the diesel generators. The result of this evaluation was to be submitted by March 1983.

The licensee does not have a loss of offsite power procedure, but provides the necessary instructions for restoring emergency busses to service if lost in Station Procedure No. 341, Emergency Diesel Generator Operation. This procedure provides guidance on diesel generator load limitations and load sequencing. Also, Region I Inspection Report 50-219/86-37 documents a review conducted to ascertain that the present configuration of the plant's offsite and onsite electric power systems is capable of sustaining and/or switching loads as required to support the safe operation of the plant. Based on discussions with licensee representatives, the submittal of the required evaluation has been discussed with NRR and a revised submittal date of March 1988 has been agreed upon.

Except as noted above, no violations were identified.

Table 4.1 Integrated assessment summary

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| II-3.B, II-3.B.1, II-3.C | 4.1(1) | Condensate Transfer Pump Power | See IPSAR Item 4.6.4 | 2.1.1 | Under Review |
| | 4.1(2) | Flooding Level Procedures | None | -- | -- |
| | 4.1(3) | Canal Water Level Instrumentation | Install water level instrumentation in intake canal. | 2.1.2 | See IPSAR Item 4.1(5) |
| | 4.1(4) | Isolation Condenser Flooding | Demonstrate minimum quantity of water maintained in condensate storage tank sufficient for long-term cooling and include minimum inventory in plant procedures. | 3.1.1<br>4.1.1 | See IPSAR Item 4.1(1)<br>See IPSAR Item 4.1(1) |
| | 4.1(5) | Low Water Level Shutdown | None | 3.1.2 | Under Review |
| | 4.1(6) | Hurricane Flooding of Pumps | Revise emergency procedures to identify alternate water sources and flow paths should low elevation pumps be flooded. | 4.1.2 | -- |
| | 4.1(7) | Flooding Elevation | Evaluate consequences of offgas building flooding and confirm all other entrance levels above 23.5 ft. | 2.1.3 | Region I to verify |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| II-3.B, II-3.B.1, II-3.C | 4.1(8) | Groundwater Elevation | See IPSAR Item 4.4(2) | -- | -- |
| | 4.1(9) | Roof Drains | Install scuppers in the reactor building and turbine building parapets. | 4.1.3 | -- |
| III-1 | 4.2 | Classification of Structures, Components, and Systems | Evaluate design of specified components on a sampling basis, upgrade if necessary, and document classification in FSAR update. | 2.2 | Submit Information for Staff Review |
| III-2 | 4.3.1 | Reactor Building Steel Structure Above the Operating Floor | Analyze and identify any needed upgrading of reactor building upper steel structure for wind loads. | 2.3.1 | Submit Information for Staff Review |
| | 4.3.2 | Ventilation Stack | Analyze and identify any needed upgrading of ventilation stack for wind loads. | 2.3.2 | -- |
| | 4.3.3 | Effects of Failure of Nonseismic Category I Structures | Analyze turbine building capacity for wind loads, evaluate consequences of failure and identify any needed upgrading. | 2.3.3 | -- |
| | 4.3.4 | Components Not Enclosed in Qualified Structures | None | -- | -- |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| | 4.3.5 | Exterior Masonry Walls | None | -- | -- |
| | 4.3.6 | Roof Decks | Provide analysis of reactor building roof. | 2.3.4 | See IPSAR Item 4.3.1 |
| | | | Analyze capacity of turbine building roof to withstand wind loads. | -- | -- |
| | 4.3.7 | Intake Structure, Oil Tanks, and Diesel Generator Building | Analyze capacity to withstand wind and tornado loads and upgrade, if necessary. | 2.3.5 | Under Review |
| | 4.3.8 | Load Combinations | See IPSAR Item 4.12 | 2.3.6 | See IPSAR Item 4.12 |
| | 4.3.9 | Soil and Foundation Capacities | None | -- | -- |
| | 4.3 | Control Room/ Architectural Structures | None | 2.3.7/ 2.3.8 | Submit Evaluation/ Submit Evaluation |
| III-3.A | 4.4(1) | Hydrostatic Loads (Combination) | None | -- | -- |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| III.3.A | 4.4(2) | Hydrostatic Loads (Short-Duration) | Evaluate short-duration hydrostatic loads on and flotation potential of structures essential to safe shutdown in conjunction with flooding emergency procedures (IPSAR Item 4.1(6)). | 2.4 | -- |
| | 4.4(3) | Below-Grade Penetration Flooding | None | -- | -- |
| III-3.C | 4.5.1 | Intake and Discharge Canals | None | -- | -- |
| | 4.5.2 | Intake Structure Trash Racks and Intake Screens | Formalize existing inspection practice as part of shift turnover or inservice inspection (ISI) procedures until water level modification is complete (IPSAR Item 4.1(3)) | 4.2.1 | -- |
| | 4.5.3 | Roof Drains | See IPSAR Item 4.1(2) | -- | -- |
| | 4.5.4 | Inspection Program | Develop and implement a formal inspection program for water control structures. | 4.2.2 | -- |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| III-4.A | 4.6.1 | Emergency Diesel Generators and Fuel Oil Day Tank | Analyze potential for and consequences of tornado-missile damage of the diesel generator building. | 2.5.1 | See IPSAR Item 4.3.7 |
| | 4.6.2 | Mechanical Equipment Access Area | Evaluate the potential for and consequences of tornado-missile impact in the reactor building access door region and identify any necessary corrective actions. | 2.5.2 | Under Review |
| | 4.6.3 | Control Room, Reactor Building, and Turbine Building Heating, Ventilating, and Air Conditioning (HVAC) Systems | None | -- | -- |
| | 4.6.4 | Condensate Storage Tank, Torus Water Storage Tank, and Service Water and Emergency Service Water Pumps | Provide protection for sufficient systems and components to ensure a safe shutdown in the event of damage from tornado missiles. | 2.5.3 | See IPSAR Item 4.1(1) |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| III-4.B | 4.7 | Turbine Missiles | Inspect turbine and propose inspection frequency based on results. | 2.6 | -- |
| | | | Justify monitoring program for main steam and reheat control valves. | -- | -- |
| III-4D | 4.8.1 | Truck Explosion | None | -- | -- |
| | 4.8.2 | Aircraft Hazards | Evaluate potential for or consequences of aircraft impact. | 2.8 | -- |
| III-5.A | 4.9(1) | Cascading Pipe Breaks | See IPSAR Item 4.16 | -- | See IPSAR Item 4.16 |
| | 4.9(2) | Jet Impingement Effects | None | -- | -- |
| | 4.9(3) | Drywell Penetration | None | -- | -- |
| III-5.B | 4.10(1) | LOCA Outside Containment | None | -- | -- |
| | 4.10(2) | Emergency Condenser Isolation | Evaluate and identify any necessary modifications to provide leakage detection to ensure that flaws would be detected before pipe break occurs. | 2.8.1 | Submit Information for Staff Review |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| III-6 | 4.11(1) | Piping Systems | Analyze on a sampling basis and verify adequacy of support designs for the seismic resistance of specified piping systems. | 2.9.1 | Under Review |
| | 4.11(2) | Mechanical Equipment | Demonstrate that the control rod drive system and vessel internals have sufficient capacity to resist a safe shutdown earthquake or take corrective action. | 2.9.2 | Under Review |
| | 4.11(3) | Electrical Equipment | Reevaluate 4160-V switchgear panel anchorage and demonstrate, on a sampling basis, adequacy of electrical panel supports. | 2.9.3 | Under Review |
| | 4.11(4) | Ability of Safety-Related Electrical Equipment to Function | None | -- | -- |
| | 4.11(5) | Qualification of Cable Trays | Provide plan to implement results of SEP Owners Group Program on a plant-specific basis. | 2.9.4 | Submit Information for Staff Review |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| III-7.B | 4.12 | Design Codes, Design Criteria, Load Combinations and Reactor Cavity Design Criteria | Evaluate adequacy of original design criteria on a sampling basis for specified structural elements. | 2.10 | Submit Information for Staff Review |
| III-8.A | 4.13 | Loose-Parts Monitoring and Core Barrel Vibration Monitoring | None | -- | -- |
| III-10.A | 4.14(1) | Thermal Overload Bypass | Evaluate thermal-overload bypasses for engineered safety features (ESF) valves. | 2.11 | -- |
| | 4.14(2) | Magnetic Trip Breakers | None | -- | -- |
| IV-2 | 4.15 | Reactivity Control Systems, Including Functional Design and Protection Against Single Failures | None | -- | -- |
| V-5 | 4.16.1 | Leakage Detection Systems | Evaluate reliability of leakage detection systems and evaluate sensitivity in conjunction with Topic III-5.A analysis. | 2.12.1 | Install APGRMS, Region I to verify |
| | 4.16.2 | Operability Requirements | Identify action for loss of leakage detection in Technical Specifications and include testing in procedures. | 3.2<br>4.3.1 | --<br>Region I to verify |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| V-5 | 4.16.3 | Intersystem Leakage | None | -- | -- |
| | 4.16.4 | Reactor Coolant Inventory Balances | None | -- | -- |
| V-6 | 4.17 | Reactor Vessel Integrity | Submit a plan for the material surveillance capsules. | 3.3 | -- |
| V-10.B | 4.18 | Residual Heat Removal System Reliability | Review and upgrade, if necessary, shutdown procedures to specify alternate sources of water for primary and secondary makeup, with particular attention to external events. | 4.4 | See IPSAR Items 4.1(4), 4.6.4, 4.30 |
| V-II.A | 4.19 | Requirements for Isolation of High- and Low-Pressure Systems | Demonstrate relief capacity and acceptable consequences, or identify corrective action to protect RWCU system. | 2.1.3 | -- |
| V-12.A | 4.20 | Water Purity of BWR Primary Coolant | Implement proposed procedure and modify Technical Specifications to be consistent. | 3.4<br>4.5 | --<br>-- |
| VI-1 | 4.21.1 | Organic Materials | Inspect and repair, if necessary, drywell coatings and recoat the torus. | 4.6.1 | -- |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| VI-1 | 4.21.2 | Postaccident Chemistry | None | -- | |
| VI-4 | 4.22.1 | Locked-Closed Valves | Provide physical locking devices to ensure valves are not inadvertently opened. | 2.14<br>4.7.1 | --<br>-- |
| | 4.22.2 | Remote Manual Valves | Evaluate leakage detection provisions and, if necessary, relocate the operating station for isolation valves in the containment spray and core spray systems. | 2.14 | Region I to verify |
| | 4.22.3 | Valve Location | None | -- | -- |
| | 4.22.4 | Instrument Lines | None | -- | -- |
| | 4.22.5 | Valve Location and Type | None | -- | -- |
| | 4.22.6 | Administrative Controls | None | -- | -- |
| VI-7.A.3 | 4.23 | Emergency Core Cooling system Actuation System | Include emergency condenser logic testing in the Technical Specifications. | 3.5 | -- |
| VI-7.A.4 | 4.24 | Core Spray Nozzle Effectiveness | None | -- | -- |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| VI-7.C.1 | 4.25(1) | AC Automatic Bus Transfers | Evaluate the existing automatic bus transfers and identify corrective actions to ensure faulted loads would not be transferred. | 4.8 | Region I to verify |
| | 4.25(2) | DC Automatic Bus Transfers | None | -- | -- |
| VI-10.A | 4.26.1 | Response-Time Testing | None | -- | -- |
| | 4.26.2 | Instrumentation for Reactor Trip System (RTS) Testing | Verify all safety logic channels tied to the reactor mode switch are tested by procedure. | 3.6 | -- |
| | | | Include logic channel testing in Technical Specifications. | 3.6 | -- |
| | 4.26.3 | Dual-Channel Testing | None | -- | -- |
| VII-1.A | 4.27(1) | Flux Monitoring Isolation | Perform failure mode and effects analysis to determine whether isolation devices are required and identify any needed upgrading. | 2.15 | Submit Information for Staff Review |
| | 4.27(2) | Reactor Protection System (RPS) Protective Trip | Install Class 1E protection at the RPS power supply and RPS interface. | 4.9.1 | -- |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| VII-1.B | 4.28 | Trip Uncertainty and Setpoint Analysis Review of Operating Data Base | Install analog trip system. | 2.16 | Submit Information for Staff Review |
| VII-2 | 4.29 | Engineered Safety Features System Control Logic and Design | See IPSAR Item 4.14(1) | -- | -- |
| VII-3 | 4.30 | Systems Required for Safe Shutdown | Provide minimum inventory for condensate storage tank as a water source for flooding events (IPSAR Item 4.1(4)) and identify non-ESF equipment in cooldown procedures (IPSAR Item 4.18). | 4.10 | See IPSAR Items 4.1(4), 4.1(6), 4.18 |
| VIII-2 | 4.31 | Onsite Emergency Power Systems (Diesel Generator) | Modify annunciators to conform to IEEE Std. 279-1971. | 4.11.1 | -- |
| | | | Evaluate bypass of two trips (voltage-ampere reactive and reverse power) during accident conditions. | 4.11.2 | Region I to verify |
| VIII-3.B | 4.32 | DC Power System Bus Voltage Monitoring and Annunciation | Schedule installation of specified battery status alarms. | 2.17<br>4.12 | --<br>-- |
| VIII-4 | 4.33 | Electrical Penetrations of Reactor Containment | None | -- | -- |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| IX-5 | 4.34(1) | Restoration of Ventilation | Evaluate and revise, if necessary, the loss-of-offsite-power procedures to ensure that restoration of ventilation systems will not overload the diesels. | 3.7.1<br>4.13.1 | Submit Information for Staff Review<br>Submit Information for Staff Review |
| | 4.34(2) | Reactor Building Ventilation | None | -- | -- |
| | 4.34(3) | Core Spray and Containment Spray Pump Ventilation | Demonstrate subject pumps can operate with a loss of ventilation, or identify corrective action, as necessary. | 2.18.1 | -- |
| | 4.34(4) | Battery, Motor-Generator, and Switchgear Ventilation | Evaluate effects of loss of ventilation to the subject rooms and identify any needed upgrading. | 2.18.2 | Region I to Verify |
| XV-1 | 4.35 | Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve | None | -- | -- |
| XV-16 | 4.36 | Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment | Implement BWR Standard Technical Specification limits for primary coolant activity. | 3.8 | Submit Information for Staff Review |

Table 4.1 (Continued)

| SEP Topic No. | IPSAR Section No. | Title | IPSAR Requirements | Supplement Section No. | Supplement Requirements/ Status |
|---|---|---|---|---|---|
| XV-18 | 4.37 | Radiological Consequences of a Main Steam Line Failure Outside Containment | See IPSAR Item 4.36 | -- | See IPSAR Item 4.36 |
| XV-19 | 4.38 | Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Pipe Breaks Within the Reactor Coolant Pressure Boundary | Develop and implement a preventive maintenance program for the main steam isolation valves, or justify existing maintenance based on operating experience. | 3.9 | -- |
| | | | Submit results of evaluation including testing experience. | 3.9 | -- |

# APPENDIX A

# REFERENCES

A. Regulatory Guidance and Industry Codes and Standards

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Generic Letter 87-02, from H. L. Denton (NRC) to All Holders of Operating Licenses Not Reviewed to Current Licensing Criteria on Seismic Qualification of Equipment, Subject: Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46, February 19, 1987.

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## B. Licensee Letters and Submittals


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---, January 20, 1983, from P. B. Fiedler (GPU) to D. M. Crutchfield (NRC), Subject: Oyster Creek Nuclear Generating Station, Docket No. 219, SEP Topic III-6, Seismic Design Considerations.

---, January 24, 1983, from Yosh Nagai (GPU) to G. Cwalina (NRC), Subject: Oyster Creek Nuclear Generating Station, Docket No. 50-219, SEP Topic No. III-6, Seismic Considerations.

---, February 2, 1983, from P. B. Fiedler (GPU) to D. M. Crutchfield (NRC), Subject: Oyster Creek Nuclear Generating Station, Docket No. 50-219, SEP Topic No. III-2, Wind and Tornado Loadings.

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---, March 10, 1983, from P. B. Fiedler (GPU) to D. M. Crutchfield (NRC), Subject: Oyster Creek Nuclear Generating Station, Docket No. 50-219, SEP Topic No. V-6, Reactor Vessel Integrity.

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C. NRC Letters and Safety Evaluations

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---, March 24, 1983, from D. M. Crutchfield (NRC) to P. B. Fiedler (GPU), Subject: Deletion of Water Quality Technical Specifications (Provisional Operating License No. DPR-16, Amendment No. 66 enclosed).

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---, August 1, 1986, from J. N. Donohew, Jr. (NRC) to GPU Nuclear Corporation, Subject: April and May 1986 Progress Review Meeting on Licensing Actions with GPU Nuclear Plant Site Personnel and Corporate Management. (Summary, including June 16 and 17, 1987, meeting.)

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## Appendix B

## NRC Staff Contributors and Consultants

This supplement is a product of the NRC Staff and its consultants. The principal contributors to this report include:

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