picture, and television cameras during this meeting will be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Chief, Nuclear Waste Branch, prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should notify Mr. Major as to their particular needs.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Richard K. Major, Chief, Nuclear Waste Branch (telephone 301/415–7366), between 8:00 A.M. and 5:00 P.M. EST.

ACNW meeting agenda, meeting transcripts, and letter reports are available for downloading or reviewing on the internet at http://www.nrc.gov/ACRSACNW.

The ACNW meeting dates for Calendar Year 1998 are provided below:

| ACNW meeting No. | 1998 ACNW meeting date |
|---|---|
| ................ | No Meeting in January. |
| 98 ........... | February 24–26, 1998. |
| 99 ........... | March 24–26, 1998. |
| 100 ......... | April 21–23, 1998. |
| ................ | No Meeting in May |
| 101 ......... | June 10–12, 1998. |
| 102 ......... | July 21–23, 1998. |
| ................ | No Meeting in August. |
| 103 ......... | September 22–24, 1998 (Las Vegas, NV). |
| 104 ......... | October 20–22, 1998. |
| ................ | No Meeting in November. |
| 105 ......... | December 15–17, 1998. |

Dated: November 26, 1997.

**Andrew L. Bates,**

*Advisory Committee Management Officer.*

[FR Doc. 97–31730 Filed 12–2–97; 8:45 am]

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## NUCLEAR REGULATORY COMMISSION

## Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

### I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 7, 1997, through November 20, 1997. The last biweekly notice was published on November 19, 1997 (62 FR 61836).

### Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 2, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, *Attention:* Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

*Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina*

*Date of amendments request:* November 6, 1997.

*Description of amendments request:* The proposed amendments change the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP) Units 1 and 2 to allow three 18-month diesel generator (DG) surveillance requirements (SR) to be performed during both plant operation (Operational Conditions 1 and 2) and shutdown (Operational Conditions 3, 4, and 5) rather than, as currently required, only during shutdown. The first SR is an inspection of the DG involving a partial disassembly. The second ensures that non-critical DG protective functions are bypassed on an Emergency Core Cooling system actuation signal. The third verifies that the DG operates for greater than or equal to 60 minutes while loaded to at least 3500 kw, which bounds the maximum expected post-accident diesel generator loading. The proposed amendments additionally remove an expired footnote from the BSEP Unit 2 DG TS.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

10 CFR 50.92 provides standards for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed these proposed license amendment requests and has concluded that their adoption would not involve a significant hazards consideration. The basis for this determination follows.

1. The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed license amendments add a footnote to SR 4.8.1.1.2.d to allow performance of SR 4.8.1.1.2.d.1, SR 4.8.1.1.2.d.4, and SR 4.8.1.1.2.d.5 in OPERATIONAL CONDITION 1, 2, 3, 4, or 5 rather than only during shutdown. The footnote requires the unit to be in OPERATIONAL CONDITION 3, 4, or 5 when performing SR 4.8.1.1.2.d.2, SR 4.8.1.1.2.d.3, SR 4.8.1.1.2.d.6, and SR 4.8.1.1.2.d.7 for its associated diesel generators. No such limitation is placed on SR 4.8.1.1.2.d.1, SR 4.8.1.1.2.d.4, or SR 4.8.1.1.2.d.5.

There is no relaxation of any limiting condition for operation (LCO) and no decrease in surveillance requirements as a result of the proposed amendments. As such, the proposed license amendments will not affect the ability of the diesel generators to perform their intended safety function. Performance of SR 4.8.1.1.2.d.1, SR 4.8.1.1.2.d.4, and

SR 4.8.1.1.2.d.5, during power operations, will not adversely affect overall nuclear safety. Diesel generator capacity is such that any three of the four diesel generators can supply the required loads for the safe shutdown of one unit and a design basis accident on the other unit without relying on offsite power. The diesel generator is not tied to the emergency bus (E bus) during performance of SR 4.8.1.1.2.d.1 or SR 4.8.1.1.2.d.4. Therefore, performance of SR 4.8.1.1.2.d.1 and SR 4.8.1.1.2.d.4, during power operation, will not affect the operability of any other safety-related systems nor will it create any perturbations of the electrical distribution system that could challenge plant operation.

Performance of SR 4.8.1.1.2.d.5, during power operation, will not adversely affect overall nuclear safety. SR 4.8.1.1.2.d.5 is performed in a similar manner to SR 4.8.1.1.2.a.5, which requires that, at least once per 31 days on a staggered test basis, a diesel generator be synchronized to the E bus and loaded to 1750 kw for 15 minutes. The critical portions of these surveillances are when the diesel generators are being synchronized to the E bus or disconnected from the E Bus. As such, performance of SR 4.8.1.1.2.d.5 during power operation does not create an additional opportunity of a perturbation of the electrical distribution system that could challenge plant operation than currently exists as a result of the performance of SR 4.8.1.1.2.a.5. The existing design of the electrical distribution system ensures that a grid problem will not result in failure of a diesel generator when it is synchronized to the E bus. The E buses are normally supplied by offsite power, via a 4160 V balance of plant (BOP) bus, through a master/slave breaker combination. When performing SR 4.8.1.1.2.d.5, the diesel generator is started in manual mode and synchronized to the E bus. With a diesel generator synchronized to the E bus, the diesel generator is protected from a potential overload condition. Class 1E protective relaying, at the E bus, is aligned to the trip circuit of the slave breaker to protect the diesel from an overload condition should the normal source of power be lost. These relays sense E bus voltage, E bus frequency, and directional power from the E bus to the BOP bus. Actuation of any of these relays, with the diesel in manual, will trip the slave and master breakers to separate the diesel generator from the BOP bus. This separates the diesel generator from the potential overload condition. In addition, either a loss of offsite power or loss of coolant accident results in the diesel generator output breaker opening, E bus loads stripping, and the diesel generator reverting to automatic mode. This allows the diesel generator to tie back to the E bus and carry the E bus loads.

The proposed license amendments reflect the clarification, previously made to Bases Section 3/4.8, ''Electrical Power Sources,'' in SR 4.8.1.1.2.d itself. Accordingly, SR 4.8.1.1.2.d.2, SR 4.8.1.1.2.d.3, SR 4.8.1.1.2.d.6, and SR 4.8.1.1.2.d.7 are performed for diesel generator 1 or 2 with BSEP, Unit No. 1 in OPERATIONAL CONDITION 3, 4, or 5 and for diesel generator 3 or 4 with BSEP, Unit No. 2 in OPERATIONAL CONDITION 3, 4, or 5. Defining the term ''during shutdown'' as ''OPERATIONAL CONDITION 3, 4, or 5'' is consistent with the current TS requirements of SR 4.8.1.1.2.d. TS Table 1.2, ''OPERATIONAL CONDITIONS,'' defines five OPERATIONAL CONDITIONS for the BSEP. There are two OPERATIONAL CONDITIONS applicable to power operation with the unit critical (i.e., POWER OPERATION and STARTUP) and three OPERATIONAL CONDITIONS applicable to a subcritical, shutdown unit (i.e., HOT SHUTDOWN, COLD SHUTDOWN, and REFUELING). Therefore, ''during shutdown'' and ''in OPERATIONAL CONDITION 3, 4, or 5'' have equivalent meaning.

Eliminating the expired BSEP, Unit No. 2 footnote to SR 4.8.1.1.2.d.1 is an administrative change and, therefore, cannot increase the probability or consequences of an accident previously evaluated.

Based on the above, the proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed license amendments to allow performance of SR 4.8.1.1.2.d.1, SR 4.8.1.1.2.d.4, and SR 4.8.1.1.2.d.5 in OPERATIONAL CONDITION 1, 2, 3, 4, or 5, rather than only during shutdown, do not affect the operation or response of any plant equipment, including the diesel generators, or introduce any new failure mechanism. Plant systems and equipment will continue to respond in accordance with design and as analyzed. There will not be a malfunction of a new or different type introduced by the proposed license amendments.

The proposed license amendments reflect the clarification, previously made to Bases Section 3/4.8, in SR 4.8.1.1.2.d itself. Accordingly, SR 4.8.1.1.2.d.2, SR 4.8.1.1.2.d.3, SR 4.8.1.1.2.d.6, and SR 4.8.1.1.2.d.7 are performed for diesel generator 1 or 2 with BSEP, Unit No. 1 in OPERATIONAL CONDITION 3, 4, or 5 and for diesel generator 3 or 4 with BSEP, Unit No. 2 in OPERATIONAL CONDITION 3, 4, or 5. Defining the term ''during shutdown'' as ''OPERATIONAL CONDITION 3, 4, or 5'' is consistent with the current TS requirements of SR 4.8.1.1.2.d. TS Table 1.2, ''OPERATIONAL CONDITIONS,'' defines five OPERATIONAL CONDITIONS for the BSEP. There are two OPERATIONAL CONDITIONS applicable to power operation with the unit critical (i.e., POWER OPERATION and STARTUP) and three OPERATIONAL CONDITIONS applicable to a subcritical, shutdown unit (i.e., HOT SHUTDOWN, COLD SHUTDOWN, and REFUELING). Therefore, ''during shutdown'' and ''in OPERATIONAL CONDITION 3, 4, or 5'' have equivalent meaning.

Eliminating the expired BSEP, Unit No. 2 footnote to SR 4.8.1.1.2.d.1 is an administrative change and, therefore, cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

Based on the above, the proposed license amendments do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety.

Bases Section 3/4.8, ''Electrical Power Systems,'' states that the operability of the alternating current (ac) and direct current power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for the safe shutdown of the facility and the mitigation and control of accident conditions within the facility. Diesel generator capacity is such that any three of the four diesel generators can supply the required loads for the safe shutdown of one unit and a design basis accident on the other unit without relying on offsite power. Performance of SR 4.8.1.1.2.d.1, SR 4.8.1.1.2.d.4, and SR 4.8.1.1.2.d.5 during power operation will not affect the operability of any other safety-related systems, nor will it create any perturbations of the electrical distribution system that could challenge plant operation. Class 1E protective relaying, at the E bus, protects the diesel from an overload condition should the normal source of power be lost while performing SR 4.8.1.1.2.d.5. There is no relaxation of any LCO as a result of the proposed license amendments. If an additional ac power source becomes inoperable during the performance of SR 4.8.1.1.2.d.1, SR 4.8.1.1.2.d.4, and SR 4.8.1.1.2.d.5, the units will be placed in the appropriate OPERATIONAL CONDITION in accordance with TS 3.8.1.1, ''A.C. Sources Operating.'' Therefore, the diesel generators' ability to perform their intended safety function, as described in Section 8.3.1.1.6.1 of the BSEP Updated Final Safety Analysis Report, is not adversely affected by the proposed license amendments.

The proposed license amendments are consistent with the guidance of Generic Letter 91–04, ''Changes In Technical Specification Surveillance Intervals To Accommodate A 24-Month Fuel Cycle,'' which concludes that TSs need not restrict surveillances to only being performed during shutdown provided that performance of the surveillance during power operations does not adversely affect safety.

The proposed license amendments reflect the clarification, previously made to Bases Section 3/4.8, in SR 4.8.1.1.2.d itself. Accordingly, SR 4.8.1.1.2.d.2, SR 4.8.1.1.2.d.3, SR 4.8.1.1.2.d.6, and SR 4.8.1.1.2.d.7 are performed for diesel generator 1 or 2 with BSEP, Unit No. 1 in OPERATIONAL CONDITION 3, 4, or 5 and for diesel generator 3 or 4 with BSEP, Unit No. 2 in OPERATIONAL CONDITION 3, 4, or 5. Defining the term ''during shutdown'' as ''OPERATIONAL CONDITION 3, 4, or 5'' is consistent with the current TS requirements of SR 4.8.1.1.2.d. TS Table 1.2, ''OPERATIONAL CONDITIONS,'' defines five OPERATIONAL CONDITIONS for the BSEP. There are two OPERATIONAL CONDITIONS applicable to power operation with the unit critical (i.e., POWER OPERATION and STARTUP) and three OPERATIONAL CONDITIONS applicable to a subcritical, shutdown unit (i.e., HOT SHUTDOWN, COLD SHUTDOWN, and REFUELING). Therefore, ''during shutdown'' and ''in OPERATIONAL CONDITION 3, 4, or 5'' have equivalent meaning.

Eliminating the expired BSEP, Unit No. 2 footnote to SR 4.8.1.1.2.d.1 is an administrative change and, therefore, cannot involve a significant reduction in a margin of safety.

Based on the above, the proposed license amendments do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

*Attorney for licensee:* William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

*NRC Project Director:* James E. Lyons.

*Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina*

*Date of amendments request:* November 6, 1997.

*Description of amendments request:* The proposed amendments to Technical Specification (TS) Limiting Conditions for Operation (LCO) 3.3.5.5, Instrumentation for Control Room Emergency Ventilation System (CREVS) and 3.7.2, Control Room Emergency Ventilation System, and associated Bases for the Brunswick Steam Electric Plant (BSEP) Units 1 and 2 would be limited in duration (approximately 3 months) and would allow operation of both BSEP units to continue while upgrades to the control building ventilation system, including new air conditioning (AC) units, are being installed. Part of the planned work requires opening the ductwork at the evaporative (i.e. cooling) coils. Temporary barriers will be constructed to preserve the leakage integrity of the control room pressure boundary; however, the temporary barriers will not be seismically qualified. While the permanent AC units are out of service, temporary AC units will be utilized. During the upgrade installation, the AC for the control room will not be protected from certain external events (e.g., seismic events, environmental hazards such as tornadoes and hurricanes, radiological sabotage, and missile hazards), as required by the system design and licensing basis, and will not fully meet single failure criteria.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not affect any component of any of the barriers to radiation release, any of the systems which protect the core from overheating, nor any system used to shut down the reactor. The proposed changes do not affect any of the chlorination system piping or the tank car, which would be the initiating components of a chlorine release event. The proposed changes affect the CREVS and CREVS instrumentation, neither of which are accident or event causing systems. Therefore, the proposed changes do not increase the probability of an accident or toxic gas release previously analyzed in the Updated Final Safety Analysis Report (FSAR).

The proposed changes do not affect the ability of the CREVS to mitigate the consequences of a design basis accident or event involving a release of radioactive material. In addition, the proposed changes do not significantly affect the ability of the system to mitigate the consequences of a toxic gas release. The following measures will be taken to minimize the consequences of accidents and events:

Temporary isolation barriers will be constructed to provide integrity of the duct during design basis radiation release events. These temporary barriers will ensure that 10 CFR Part 50, General Design Criterion 19 for Control Room operator doses is met for all design basis radiation release accidents.

During the time that the temporary barrier is used, the chlorine tank car will be removed from the exclusion area. Analyses have shown that with the chlorine tank car outside of the exclusion area, there is no threat to Control Room habitability. Removal of the chlorine tank car from the exclusion area is the current Technical Specification requirement for inoperability of the Control Room chlorine isolation mode.

The temporary condensing units for the Control Room Air Conditioning system will be installed to high quality standards, and a spare condensing unit will be provided such that two units can be maintained functional. These units will each be powered from a separate division of Class 1E power. The operation of the units will be monitored to ensure that they are in good operating order.

If two or more of the condensing units should fail, instructions have been provided to the operators for increased monitoring of temperatures, and mitigating actions are available to the operators if temperatures rise above a predetermined limit.

Therefore, the consequences of an accident or an event involving a release of radiation, toxic gas, or smoke will not be significantly increased. In addition, the change will not significantly affect the consequences of a seismic event or other severe natural phenomena, as previously analyzed in the Updated FSAR.

2. The proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes involve adjustments to the LCO requirements for CREVS relative to protection from severe natural phenomena. The proposed changes do not introduce any new modes of plant operation. The proposed changes do not involve any new modes of system operation, except that temporary condensing units will be used in place of the permanent condensing units. The temporary condensing units will interface with the permanent Control Building Heating Ventilation and Air Conditioning system in a similar manner to the permanent system. The piping connections to the permanent system will be the same, and the controls interface will be the same. No new cross-ties will be created and no new piping will be run though the habitability boundary. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. The proposed license amendments do not involve a significant reduction in a margin of safety.

The proposed changes do not represent a significant change in the assumptions and inputs to the analyses for Control Room operator doses. No increase in the doses to the Control Room operators is expected after a seismic event or tornado, since the integrity of existing barriers to release of radioactive material are not affected. Therefore, this change does not result in a significant reduction in the margin of safety for a radiological event.

The proposed change does not represent a change to the leakage criteria for the Control Room, or the Control Room ventilation ductwork, following either a toxic gas or external smoke event. The bounding analysis remains valid, unless the failure is caused by a tornado or seismic event. Due to the low probability of such an event occurring during the short time frame involved in this modification, the occurrence of such an event is not of significant concern.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

*Attorney for licensee:* William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

*NRC Project Director:* James E. Lyons.

*Consumers Energy Company, Docket No. 50–155, Big Rock Point Nuclear Plant, Charlevoix County, Michigan*

*Date of amendment request:* September 19, 1997 (Accession No. 9709240373).

*Description of amendment request:* The amendment request propose changes to the Facility Operating License and technical specifications (TS) to reflect the permanent cessation of power operations and permanent transfer of nuclear reactor fuel to the spent fuel pool (SFP). In particular, Consumers Energy requests to change: safety limits; limiting safety system settings; limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls. On November 12, 1997, Consumers Energy provided supplemental information regarding their no significant hazards determination, as requested by NRC request for additional information letter dated October 12, 1997. By letters dated June 26 and September 23, 1997, the licensee certified permanent cessation of power operations and permanent removal of all fuel from the reactor, respectively.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change provides the applicable requirements to assure safe storage of spent nuclear fuel during decommissioning following the permanent cessation of power operations at the Big Rock Point Nuclear Plant (BRP) on August 30, 1997 [see Consumers Energy letter to NRC dated June 26, 1997] and permanent removal of all fuel from the reactor vessel on September 20, 1997 [see Consumer Energy letter to NRC dated September 23, 1997]. Decommissioning activities conducted using these controls do not present undue risk to the public, and do not impact common defense and security. As such, these changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

No accidents previously evaluated in the Updated Final Hazards Summary Report (UFHSR) will have their probability of occurrence increased because the proposed controls effectively preclude the occurrence of criticality, fuel temperature exceeding limits, or fuel handling accidents. The probability of plant accidents associated with power operations have been significantly reduced. Accidents associated with spent fuel handling, including cask and single bundle drop and spent fuel cooling capability loss events, are still pertinent and were reviewed using new data on pool inventory and revised 10 CFR 20 radiological limit determinations. The probability of occurrence of accidents associated with storing 441 spent fuel assemblies in the SFP (current license limit) have not been affected by the changes in the proposed TSs.

The consequences of a fuel handling and cask drop accidents were evaluated based on the removal of all fuel from the reactor and loading spent nuclear fuel in the SFP. The removal of all fuel from the reactor vessel to storage in the SFP and the subsequent decay of the fuel in the pool result in no increase in the probability of these accidents and continuously reduced consequences from these accidents.

Analyses using the techniques in Branch Technical Position APCSB 9–2 provide the heat rate from a freshly-removed full core off-load in the SFP whose racks are filled with a total of 441 fuel assemblies as the most limiting cooling condition. Existing cooling equipment under the current TSs provide sufficient cooling to preclude spent fuel pool temperatures reaching 150 degrees-Fahrenheit with a complete loss of spent fuel cooling for 72 hours. This precludes entry into an unanalyzed condition for the SFP and provides 3 days to recover cooling flow of ''approximately 30'' gallons per minute. Since this specification change is intended for implementation following 93 days after shutdown (approximately November 30, 1997), this analysis justifies the allowance of 24 hours to re-establish cooling flow provided in specification 3.1.2.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The permanent cessation of power operation and removal of fuel from the reactor eliminates the possibility of the following categories of accidents and transients to create a hazard to the health and safety of the public: increase in heat removal by the secondary system; increase in reactor coolant inventory; decrease in heat removal by the secondary system; decrease in reactor coolant inventory; reactivity and power distribution anomalies; anticipated transient without scram; and, single loop operation. These revised TSs, in combination with requirements in the UFHSR, provide assurance that fuel handling and spent fuel cask drop accident, which represent the remaining specific pertinent accidents analyzed in the ''radioactive release from a subsystem of component'' category, will not occur. Because the revised TSs related to fuel handling, spent nuclear fuel storage, and handling of the spent fuel cask satisfy current license and UFHSR requirements, no new accidents are created.

3. Involve a significant reduction in a margin of safety.

The safety margins for analyzed accidents are maintained because the containment structures and redundant control established by the plant remain in place until the decay of spent fuel has reduced the source term to levels that analysis confirms do not require the containment features. ninety three days after permanent cessation of operations, the spent nuclear fuel at BRP will have decayed to the point where the added margin from this decay more than compensates for the removal of the containment as a safety feature, and allows relaxed controls for the cooling of the SFP.

The Big Rock Point Plant Safety Committee has reviewed this Facility Operating License and TS change request and has determined this change does not involve an unreviewed safety question and, therefore, involves no significant hazards consideration. The proposed change has been reviewed by the BRP Nuclear Performance Assessment Department.

The NRC staff has reviewed the licensee's analysis, as provided by licensee letters dated September 19 and November 12, 1997, and, based on this review, it appears that the three standards or 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room:* North Central Michigan College, 1515 Howard Street, Petosky, MI 49770.

*Attorney for licensee:* Judd L. Bacon, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

*NRC Project Director:* Seymour H. Weiss.

*Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina*

*Date of amendment request:* March 11, 1993; supplemented August 26, November 29, December 6, 1993, October 3, 1995, February 27, and September 3, 1997 (TSC 93–03).

*Description of amendment request:* The proposed changes would replace the present Electrical Power Systems section of the Technical Specifications, Sections 3.7 and 4.6, by consolidating and rearranging the present specifications, incorporating new specifications, and formating the section similar to the Babcock and Wilcox Standard Technical Specifications. The proposed changes would address such concerns as Keowee hydro station operability, Lee gas turbine operability, overhead and underground emergency power path operability, Keowee and Keowee main step-up transformer outage requirements, surveillance requirements of various components and systems, Oconee distribution system requirements, protective instrumentation system requirements, operability of 125 VDC Vital Instrument and Control power and limiting condition for operation, inverter requirements, Oconee shutdown requirements related to various components, Keowee unit extended outage, dc power operability requirements, battery cell parameter requirements, and various editorial and related Bases changes.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below.

Duke Power Company (Duke) [currently Duke Energy Corporation] has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

Each accident analysis addressed within the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to the changes proposed within this amendment request. Changes included in this amendment request are provided to assure availability of electrical power systems for mitigation of Design Basis Accidents (DBAs). As described within the technical justification, the following types of changes are included:

(1) Editorial and administrative changes associated with reformatting the Technical Specification requirements;

(2) Additional restrictions not presently included in the Technical Specifications such as the addition of requirements for electrical power systems during cold shutdown and refueling, for the 230 kV switchyard degraded grid protection system and to delete the special inoperability period for the Keowee CX transformer;

(3) Technical changes to current requirements to provide clarity and operational flexibility. These changes maintain the ability of the electrical power systems to mitigate the consequences of DBAs without a significant reduction in availability. These changes include the definition of emergency power paths to include the associated DC sources and auxiliary transformers, the combination of special inoperability periods for ''planned'' and ''unplanned'' reasons, and the ability to use the Keowee special inoperability period more than once in a three year period; and

(4) Relocation of requirements which are unnecessary for the mitigation of DBAs to licensee controlled documents. Relocated requirements include surveillance requirements for the External Grid Trouble Protection system.

Based on the above and the technical justification * * *, there is no significant increase in the probability of DBA as a result of this change, nor is there a significant increase in the consequences of a DBA as a result of this change since the proposed amendment assures availability of electrical power systems.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

The proposed changes make no physical changes to the plant configuration and do not adversely affect the performance of any equipment. Operation of ONS [Oconee Nuclear Station] in accordance with these Technical Specifications will not create any failure modes not bounded by previously evaluated accidents. Consequently, this change will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

(3) Involve a significant reduction in a margin of safety:

Margins of safety associated with these Technical Specifications have been evaluated. These changes include editorial and administrative changes associated with reformatting the Technical Specification requirements, additional restrictions not presently included in the Technical Specifications, technical changes to current requirements which maintain the ability of the electrical power systems to mitigate the consequences of DBAs, and relocation of requirements which are unnecessary for the mitigation of DBAs to licensee controlled documents. The design basis of auxiliary electrical systems is to supply the required ES [emergency system] loads of one Unit and safe shutdown loads of the other two units. The proposed amendment does not affect any safety limits, setpoints, or design parameters and assures the continued availability of electrical power systems; thus preserving the existing margin of safety. Therefore, there will be no significant reduction in any margin of safety.

Duke has concluded based on the above, and the technical justification * * * that there are no significant hazards considerations involved in this amendment request.

The NRC has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

*Attorney for licensee:* J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

*NRC Project Director:* Herbert N. Berkow.

*Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania*

*Date of amendment request:* November 4, 1997.

*Description of amendment request:* The proposed amendment would change Item 6.a.2, ''4.16 Emergency Bus (Start Diesel),'' of Table 3.3–4 of Technical Specification 3.3.2.1. The proposed change would reduce the trip setpoint for starting the emergency diesel generators on emergency bus undervoltage from a trip setpoint of greater than or equal to 83 percent with a 12-cycle delay time to greater than or equal to 75 percent of nominal bus voltage with a time delay of less than 0.9 second including auxiliary relay times. The proposed change would also reduce the allowable value from greater than or equal to 81 percent of nominal bus voltage to greater than or equal to 74 percent of nominal bus voltage with a time delay of less than 0.9 second including auxiliary relay times.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change replaces the current Engineered Safety Feature setpoint, allowable value and delay time for the diesel generator start on loss of power function. An analysis has been performed to develop the new values to minimize the diesel generator starts when a Reactor Coolant Pump (RCP) is being started or a fast bus transfer occurs. The heat generated by an increase in motor current, in response to reduced voltage, will be less than the heat generated during motor starting. The analysis results show that bus voltages may dip below the allowable setpoint value and then recover to the pick-up setpoint within the proposed delay time without stalling motors.

The proposed change does not affect the design and reliability of any plant equipment; therefore, the probability of occurrence of a previously evaluated accident is not increased. The operation of the plant will not be changed as a result of this proposed amendment, except that fewer diesel generator starts will be initiated.

This function anticipates the loss of voltage to protect equipment connected to the 4.16 Kv emergency bus. The UFSAR [Updated Final Safety Analysis Report] accident analyses do not take credit for this function; therefore, the consequences of an accident previously evaluated is not increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change to the trip setpoint, allowable value and delay time will continue to ensure that the safety-related equipment connected to the emergency bus is adequately protected from a low voltage condition. These setting changes will minimize the diesel generator starts due to voltage drops when an RCP is started or a fast bus transfer occurs.

The new setpoint and time delay allow normal voltage drops to occur during expected plant operations without causing any thermal damage to safety-related equipment. The performance of the safety system will remain unchanged and will not alter any plant equipment, performance requirements or safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not involve a significant reduction in a margin of safety since an analysis has been performed to verify that safety-related equipment connected to the emergency bus is adequately protected from a low voltage condition with the proposed settings. The proposed changes do not affect the UFSAR design bases, accident assumptions, or technical specification bases. In addition, the proposed changes do not affect release limits, monitoring equipment or plant operating practices. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

*Attorney for Licensee:* Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Project Director:* John F. Stolz.

*Entergy Operations, Inc., et al., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi*

*Date of amendment request:* October 28, 1997.

*Description of amendment request:* The amendment would (1) revise the frequency of conducting five Surveillance Requirements (SRs) and (2) add a 10 CFR Part 50, Appendix J Testing Program for Primary Containment Systems in the Technical Specifications (TSs) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The five SRs are the following: SR 3.6.1.1.1 for primary containment, SR 3.6.1.2.1

for primary containment air locks, and SRs 3.6.1.3.5, 3.6.1.3.8, and 3.6.1.3.9 for primary containment isolation valves. The proposed revisions for each of the five SRs are to delete the references to SR 3.0.2 not being applicable and change the surveillance frequency from being ''in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions'' to ''in accordance with 10 CFR Part 50, Appendix J, testing program.'' The testing program would be added to Section 5.0, Administrative Controls, of the TSs. Changes to the Bases of the TSs were also provided in the submittal.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

[On April 26, 1995, the licensee was granted an exemption to Appendix J of 10 CFR Part 50 that allowed performance-based containment leak rate testing. This exemption will expire on the startup from Refueling Outage 9, currently scheduled for the spring of 1998. The licensee's proposed changes to the TSs are to adopt Option B, Performance-Based Requirements, that is now in Appendix J, but was not in Appendix J in 1995 when the exemption was granted. The technical findings that support the rulemaking for Option B are in NUREG–1493, ''Performance-Based Containment Leak Rate Test Program,'' dated September 1995. The licensee stated in its submittal that its current containment leak rate testing program meets the requirements of Option B.]

Two initiating events were identified which could be affected by the proposed changes [in the submittal of October 28, 1997]. An interfacing system LOCA [(loss-of-coolant accident)] could be caused by significant leakage of both normally closed isolation valves in systems with high pressure/low pressure interfaces. Interfacing systems LOCAs were considered for the LPCI, LPCS, HPCS, and RCIC systems [(i.e., low pressure coolant injection, low pressure core spray, high pressure core spray, and reactor core isolation cooling)]. Because the frequency for testing of these valves will not be changed under this proposal, there is no increase in the probability or consequences of an accident [previously evaluated].

The second event evaluated was a LOCA outside containment. In this case the probability for failure of the MSIVs [(main steam isolation valves)] and the feedwater isolation valves were calculated and combined with the frequency of a pipe break outside containment and the conditional probability of a core melt given a LOCA. The increase in core damage is extremely small and therefore does not significantly increase the probability of any previously evaluated accident. Further, because the testing frequency for MSIVs and feedwater isolation valves are not being changed, the LOCA outside containment events can be discounted.

Failure of, or leakage through[,] a containment barrier can[,] however, increase the consequences of those accidents previously evaluated. Because the leakage probability for two valves in series to fail is very small and because all lines isolated by a single containment isolation valve always have a water seal and cannot act as a release pathway unless the integrity of the connected system is compromised, there is no significant increase in the consequences of any previously evaluated accident.

Containment bypass can also increase the consequences of [previously] evaluated accidents. Accident sequences involving containment have been shown to be relatively insignificant by the GGNS IPE [(Individual Plant Examination]). The potential for [containment] bypass was analyzed. The analysis showed that the probabilities for bypass were dominated by failure to close scenarios. Many programs are in place at GGNS to monitor containment component performance[,] and to ensure that proper maintenance and repairs are made during the service life of the containment. Other routine surveillances are performed periodically to ensure that the valves will close on demand. In fact, all valves that are required to close for containment isolation and that are not maintained closed at all times during power operations are stroke tested quarterly or[,] at a minimum, during each refueling outage in accordance with ASME [(American Society of Mechanical Engineers) Boiler and Pressure Vessel Code,] Section XI, Subsection IWV.

[Based on the above, the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.]

II. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The request involves the reduction in the local leak rate and the integrated leak rate testing frequencies [in accordance with Option B of Appendix J to 10 CFR Part 50]. Extending the test frequencies has no influence on, nor does it contribute in any way to, the possibility of a new or different kind of accident or malfunction from those previously analyzed. The method of performing the test is not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change.

[Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.]

III. The proposed change does not involve a significant reduction in a margin of safety.

The only margin of safety that has the potential of being impacted by the proposed changes involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to La which is defined by the GGNS TSs to be 0.437 percent by weight of the containment air [volume] per 24 hours at [the containment pressure of] 11.5 psig (Pa). The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (11.5 psig, Pa).

To provide additional conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 La during performance of the periodic integrated leakage rate test and to less than or equal to 0.60 La for type B and C leakage rate tests [of Appendix J]. This is done to account for the possible degradation of the containment leakage barriers between [the Appendix J] tests. This acceptance criteria ensures that an acceptable margin of safety is being maintained and will not be altered by the proposed changes. The preservation of this margin will continue to provide for potential degradation of the leakage barriers between tests.

No change in the method of testing is being proposed. The tests will continue to be done at full pressure (Pa) or greater [pressure]. The test pressure for primary containment isolation valves will continue to be applied in the same direction as would be required for the valve to perform its safety function (unless a different direction can be shown to be equivalent or conservative). Primary containment penetrations which require Type B leakage rate tests will be performed in the same manner as before. The Type A test [of Appendix J] will continue to be performed at full pressure (Pa). Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment[,] and systems and components penetrating the primary containment.

No change in the owners allowable leakage rate is being proposed. These conservative leakage rates ensure that[,] if every penetration were at its maximum allowable leakage rate, the total containment leakage would still be below 0.60 La. The effect of multiple penetration barriers is not considered which provides further conservatism.

The assessment of risk analysis for the proposed changes concluded that the overall risk impact of the changes are neutral and essentially negligible. Any containment isolation barrier allowed to be tested at less frequent intervals [through performance-based testing of proposed Option B of Appendix J] will have demonstrated enhanced performance which minimizes the potential for increased leakage. The assessment further shows that there is reasonable assurance that an acceptable level of performance for the containment isolation function can be maintained. The overall risk impact for the proposed changes are small enough to be almost indeterminate. No change to the leakage rate specified in the TSs is being proposed.

[The proposed changes to the TSs are in accordance with Option B of Appendix J of 10 CFR Part 50.]

[Therefore, the proposed changes do not involve a significant reduction in a margin of safety.]

Based on the above evaluation, operation in accordance with the proposed amendment involves no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room Location:* Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

*NRC Project Director:* David A. Wigginton, Acting.

*Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee, Atomic Power Station, Lincoln County, Maine*

*Date of amendment request:* September 30, 1997.

*Description of amendment request:* The proposed amendment would eliminate certain license conditions of the Maine Yankee operating license that are no longer appropriate in the permanently defueled condition of the plant. These conditions include restrictions on the Fire Protection Program and implementation of leakage reduction, airborne iodine monitoring, secondary water chemistry, and cooling water discharge monitoring programs. By letter dated August 7, 1997, the licensee certified permanent cessation of power operations and permanent removal of fuel from the reactor vessel. Most of the provisions of the Maine Yankee operating license were established to ensure protection of the public health and safety during power operations. Maine Yankee has proposed to eliminate those license requirements that are not relevant to the permanently defueled plant condition to allow the Maine Yankee staff to focus on those provisions which are still appropriate during decommissioning.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change does not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated.

The purpose of the proposed change is to eliminate requirements which are not appropriate in the permanently defueled plant condition. Since the plant has permanently ceased operation and will be maintained in a defueled condition, many provisions of the license related to operation of the plant are no longer appropriate. Elimination of these unnecessary requirements allows the plant staff to focus on those requirements which continue to be appropriate to the existing plant condition. The proposed change does not affect those Chapter 14 accidents which are appropriate to the current plant conditions: fuel handling accident, spent fuel cask drop, and radioactive liquid waste system leaks and failures, and therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The purpose of this proposed change is to eliminate requirements which are not appropriate in the permanently defueled plant condition. Since the plant has permanently ceased operation and will be maintained in a defueled condition, many provisions of the license related to operation of the plant are no longer appropriate. Elimination of these unnecessary requirements allows the plant staff to focus on those requirements which continue to be appropriate to the existing plant conditions. This proposed change does not affect storage of spent fuel and, therefore, does not create the possibility of a new or different accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The purpose of the proposed change is to eliminate requirements which are not appropriate in the permanently defueled plant condition. Since the plant has permanently ceased operation and will be maintained in a defueled condition, many provisions of the license related to operation of the plant are no longer appropriate. Elimination of these unnecessary requirements allows the plant staff to focus on those requirements which continue to be appropriate to the existing plant conditions. This proposed change does not affect storage of spent fuel and, therefore, does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

*Attorney for licensee:* Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, P.O. Box 408, Wiscasset, ME 04578.

*NRC Project Director:* Seymour H. Weiss.

*Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine*

*Date of amendment request:* October 20, 1997.

*Description of amendment request:* The proposed amendment would replace in their entirety the existing Technical Specifications incorporated in Facility Operating License No. DPR–36 as Appendix A. Maine Yankee developed the revised Technical Specifications, titled Permanently Defueled Technical Specifications, to reflect the permanently shutdown and defueled status of the plant. Changes are proposed to the definitions, limiting conditions for operation, surveillance, and administrative control sections.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). A summary of the licensee's review is presented below:

The proposed change does not,

1. Involve a significant increase in the probability or consequence of an accident previously evaluated.

This proposed change is consistent with the improved Standard Technical Specifications. The relocation of requirements from the technical specifications to the licensee controlled documents is consistent with the criteria set forth in 10 CFR 50.36 for the content of technical specifications. The removal of definitions, generic LCO actions and generic surveillance requirements has no impact on facility structures or equipment or the methods of operation of such structures or equipment. The deletion of design features and safety limits not applicable to the permanently shutdown and defueled status of the Maine Yankee reactor has no impact on the remaining applicable design basis accidents. The removal of LCO and Surveillance specifications which are related only to the operation of the nuclear reactor or only to the prevention, diagnosis or mitigation of transients or accidents primarily involving the reactor, do not affect the remaining applicable accidents previously evaluated. The critical safety functions involving core reactivity control, reactor heat removal, reactor coolant system inventory control and containment integrity are no longer necessary at the Maine Yankee facility. The postulated accidents involving damage to the reactor coolant system, main steam lines, main feed lines, steam generators or the reactor core and the subsequent release of radioactive material are no longer applicable at the Maine Yankee facility. Spent fuel pool cooling and makeup related equipment and support equipment including electrical power systems are not required to be continuously available since there is time available to effect repairs or establish alternate sources of makeup flow in the event of a loss of cooling and makeup flow to the spent fuel pool. The effect of radioactive decay since the shutdown of the reactor has reduced the consequences of the fuel handling accident to levels below those previously analyzed. The relevant parameters associated with spent fuel pool (level and boron concentration) that make up the initial conditions assumed in applicable analysis are included in the technical specifications. The deletion and modification of provisions of administrative controls do not directly affect the design of structures or equipment necessary for the safe storage of irradiated fuel or the methods used for handling and storage of such fuel in the spent fuel pool. The changes to the administrative controls are, in fact, administrative in nature and do not affect any accident applicable to the safe storage of irradiated fuel or the permanently shutdown and defueled condition of the reactor. Therefore, the proposed changes to the Maine Yankee Technical Specifications do not involve any increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes have no impact on facility structures or equipment affecting the safe storage of irradiated fuel or the methods of operation of such structures or equipment or handling and storage of such fuel. These changes are consistent with the improved Standard Technical Specifications and add to the clarity and ease of use of the proposed PDTS. The removal of technical specifications which are related only to the operation of the nuclear reactor or only to the prevention, diagnosis or mitigation of transients or accidents primarily involving the reactor, can not result in different or more adverse failure modes or accidents than previously evaluated because the reactor is permanently shutdown and defueled. The proposed deletion of provisions of the Maine Yankee Technical Specifications do not affect systems credited in the existing accident analyses for the remaining applicable postulated accidents at the Maine Yankee facility. The proposed technical specifications continue to require proper control and monitoring of safety significant parameters and activities. The proposed restrictions on boron concentration and level in the spent fuel pool are fulfilled by normal operating conditions and preserve initial conditions assumed in the analyses of postulated DBA's. Therefore, the proposed changes to the MYTS does not create the possibility of a new or different accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The deletion of provisions in the technical specifications which are not related to the storage of irradiated fuel or which are inconsistent with the scope of the improved Standard Technical Specifications will not affect the analyses of the design basis accidents remaining applicable to the Maine Yankee facility. The postulated design basis accidents involving the reactor are no longer possible due to the permanently defueled status of the Maine Yankee reactor. The requirements for systems, structures and components which have been deleted from the Maine Yankee Technical Specifications are not credited in the existing accident analysis for the remaining applicable postulated accidents and therefore do not contribute to the margin of safety associated with the accident analysis. Therefore, the proposed changes to the Maine Yankee Technical Specifications would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

*Attorney for licensee:* Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, P.O. Box 408, Wiscasset, ME 04578.

*NRC Project Director:* Seymour H. Weiss.

*Northeast Nuclear Energy Company (NNECO), et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut*

*Date of amendment request:* October 15, 1997.

*Description of amendment request:* The proposed change to Technical Specification 3/4.4.3, Pressurizer, would replace the pressurizer maximum water inventory requirement with a pressurizer maximum indicated level requirement. The proposed amendment would also modify the associated Bases section and make editorial changes.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

NNECO has reviewed the proposed revision in accordance with 10 CFR 50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated.

The Technical Specification maximum pressurizer inventory requirement in Technical Specification 3.4.3 is being changed to use the numerical value for the Reactor Trip setpoint on pressurizer high water level in Technical Specification Section 2.2. This changes the requirement from a volume to a level requirement, is consistent with the Improved Standard Technical Specifications for Westinghouse plants, and represents a more restrictive level requirement than the current technical specification. The bases change clarifies that the 89% level requirement only assures that there is a steam bubble in the pressurizer. Also, the bases change states that pressurizer level is maintained by automatic and procedural controls to provide assurance that the design basis analyses are valid. These changes do not modify plant operation. Lowering the maximum level requirement so that it is numerically consistent with the reactor trip setpoint, while clarifying the bases of the requirement, [cannot] involve a significant increase in the probability or consequences of an accident previously evaluated.

Therefore, the proposed revision does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

There are no hardware modifications associated with the change. The change does not modify the way that the plant is operated. The change modifies neither accident mitigation nor system response post-accident.

Therefore, the proposed revision does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The change places a lower maximum pressurizer level requirement for the pressurizer. The change imposes the numerical setpoint value for the reactor trip on pressurizer high water level as the restriction on the pressurizer level. The change to the bases clarifies that the 89% level requirement only ensures the existence of a steam bubble and not the validity of the design basis analyses. The design basis non-LOCA [loss-of-coolant accident] analyses use the current programmed pressurizer level and the LOCA analysis uses 62% level for full power. Those events that are analyzed to address pressurizer filling concerns are initiated assuming a higher initial pressurizer water level that accounts for 6% level uncertainty. The bases change makes it clear that the pressurizer level required to assure the validity of the design basis analyses is maintained by the automatic and procedural controls and not the less than or equal to 89% level in the requirement.

Therefore, the proposed revision does not involve a significant reduction in a margin of safety.

In conclusion, bases on the information provided, it is determined that the proposed revision does not involve an SHC.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

*Attorney for licensee:* Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

*NRC Deputy Director:* Phillip F. McKee.

*Northeast Nuclear Energy Company (NNECO), et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut*

*Date of amendment request:* November 11, 1997.

*Description of amendment request:* The proposed amendment to Technical Specifications (TS) 3.9.1.2 and 3.9.13 and their Bases will allow crediting soluble boron for maintaining k-effective at less than or equal to 0.95 within the spent fuel pool (SFP) rack matrix following a seismic event of a magnitude greater than or equal to an operating basis earthquake (OBE).

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated.

There is one spent fuel pool accident condition discussed in Chapter 15 of the FSAR [final safety analysis report]. The FSAR discusses a fuel handling accident which drops a fuel assembly onto the fuel racks during fuel movement. Degradation of the Boraflex panels in a post-seismic condition will have no effect on the probability of a fuel assembly drop onto the stored fuel, or the fuel racks. Changing the way Boraflex responds to a seismic event will have no impact on the probability of a seismic event. A misplaced fuel assembly can be postulated in the MP3 [Millstone Unit 3] fuel pool as a result of either equipment malfunction or operator error. Degradation of the Boraflex panels will have no effect on the probability of a fuel misplacement event. Therefore, the degradation of Boraflex in a post-seismic condition does not involve an increase in the probability of an accident previously evaluated.

A fuel handling accident could cause a radioactive release of fission gases, resulting in dose consequences. This radioactive release of fission gases is due to the failure of a certain number of fuel pins which are postulated to fail during the fuel handling accident. The number of fuel pins which are postulated to fail in this event is not changed by the degradation of the Boraflex panels in a post-seismic condition. There are no criticality issues with this fuel handling accident for the reason described next. Although conservative, should a fuel handling accident occur during or after a seismic event, even with no Boraflex credit, the proposed 1750 ppm [parts per million] of soluble boron is sufficient to ensure that K-effective of the SFP is maintained at less than or equal to 0.95. The 1750 ppm boron requirement also bounds any criticality concerns for a fuel handling or dropped load event due to the no Boraflex assumption. Therefore, this proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

Therefore, the proposed revision does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The change in the way Boraflex responds to a seismic event with the presence of 1750 ppm boron does not create a new accident. The use of soluble boron in the spent fuel pool is safe. There is no possibility of a dilution event during or following a seismic event up to the magnitude of an SSE [safe shutdown earthquake]. The normally filled piping systems in the vicinity of the spent fuel pool are fire protection, hot water heating, hot water preheating, domestic water, and component cooling. In addition, the roof drain system piping runs through the building. An engineering review of these systems has determined that the majority of the systems are leak tight and meet NU's [Northeast Utilities'] commitment to seismic II/I criteria for a seismic event up to and including an SSE. The analysis was performed consistent with the original design criteria for seismic II/I piping as documented in section 3.9.2 of the Millstone 3 Safety Evaluation Report (SER) Number 4.

Portions of fuel building piping systems that may not be leak tight following an SSE, and that would not leak into the spent fuel pool based on location of the potential leak, are not possible sources of dilution.

Two lines in the Hot Water Preheating system will be modified to meet the leak tight seismic II/I criteria and will not be possible sources of dilution.

A new pipe support will be added to the roof drain piping to meet the seismic II/I criteria. With the new support installed, one portion of the drain piping will still not meet leak tight requirements. The inlet opening on the roof feeding this portion of the piping will therefore be capped. Since the location of the potential cracking in the drain piping lies above the connection to the balance of the drain piping, and the system is not under pressure, water flowing from other portions of the drain system will not flow up to and out of the potentially cracked portion. This precludes a possible source of dilution.

Non borated water sources that are connected to the SFP will be isolated following a seismic event of greater than or equal to an OBE to prevent dilution. Therefore there is no possibility of a SFP boron dilution accident coincident with or following a seismic event up to an SSE, and credit for soluble boron is acceptable to meet the K-effective limit of 0.95 for the SFP. The crediting of soluble boron in the spent fuel pool to control K-effective following a seismic event does not create a new accident as boron dilution of the pool can be prevented by closing and administratively controlling the opening of dilution paths to the pool and initiating routine sampling requirements on SFP boron. At present the crediting of soluble boron following a fuel misplacement event is allowed for [in] the Millstone 3 [TS]. Analysis has shown that a seismic event of greater than an OBE level earthquake can cause Boraflex damage which can be more limiting than a fuel misplacement event. As such, the minimum boron requirement in the fuel pool will be increased from 800 ppm to 1750 ppm. As such, no new accident has been created because the crediting of boron following a malfunction/accident has always been allowed.

Therefore, the proposed revision does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The margin of safety, as defined by MP3 Technical Specifications, is to ensure that the K-effective of the MP3 SFP is maintained less than or equal to 0.95 at all times. The proposed change does not credit soluble boron during normal operations, but allows crediting soluble boron at a new higher concentration for control of K-effective during malfunction conditions. There is no reduction in the margin of safety as the result of the degradation of Boraflex following a greater than OBE seismic event, because soluble boron will compensate for the loss of Boraflex. A value of 1750 ppm of soluble boron in the SFP at all times ensures that K-effective of the MP3 SFP is maintained less than or equal to 0.95 at all times, including this new malfunction of degraded Boraflex following a greater than OBE seismic event.

Eliminating the credit for the reactivity [hold-down] effect of Boraflex panels in conjunction with 1750 ppm boron will have no effect on the probability of a seismic event. As the probability of a seismic event has not changed there is no increase in the probability of an accident or malfunction due to a seismic event. Following a seismic event, operators are presently required to make inspections of the plant to determine post seismic event plant conditions. As a result of this change, inspections will be required to review the status of the spent fuel pool and isolate potential dilution paths following a seismic event of greater than or equal to [an] OBE. These actions are consistent with present guidance in the seismic response procedure and do not create an undue burden on the operator. To compensate for the potential loss of Boraflex after a seismic event, the SFP is now required to be [] borated at all times to at least 1750 ppm to maintain the proper post seismic K-effective condition. As such, there is no mitigation equipment that has to operate in the spent fuel pool following a seismic event.

Although the Boraflex in the fuel racks is assumed to fail in a seismic event greater than an OBE, the presence of soluble boron in the fuel pool water will compensate for the loss of Boraflex. Surveillance requirements on SFP boron will ensure that there will be boron present in the SFP and ensure that the SFP is not diluted below the minimum required boron concentration during normal operation.

As the presence of SFP soluble boron during and after a seismic event maintains k-effective less than 0.95 there is no effect on the consequences of any accidents evaluated. As there are no new accidents created, there are no changes in the consequences of previously analyzed accidents, and there is no effect on the consequences of any accident. There is no reduction in the margin of safety as the result of the degradation of Boraflex following a greater than OBE seismic event, because during normal operations k-effective remains less than 0.95 without reliance on soluble boron, and during malfunction and accident conditions soluble boron can be used to compensate for the loss of Boraflex to maintain K-effective less than 0.95.

Therefore, the proposed revision does not involve a significant reduction in a margin of safety.

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

*Attorney for licensee:* Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

*NRC Deputy Director:* Phillip F. McKee.

*Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska*

*Date of amendment request:* October 3, 1997.

*Description of amendment request:* Omaha Public Power District (OPPD) proposes to change the Fort Calhoun Station Unit No. 1 Technical Specifications (TS) by revising TS Surveillance Requirement 3.9, ''Auxiliary Feedwater System,'' to clarify what flow paths are required to be tested. Additionally, OPPD proposes to revise the auxiliary feedwater pumps' surveillance requirements to delete the specific discharge pressure.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

A change to TS 3.9(2) is proposed to delete the specific discharge pressure specified for the Auxiliary Feedwater (AFW) pumps' surveillance. The developed head of the motor-driven and steam turbine-driven AFW pumps is verified quarterly. These tests are in addition to those required by TS 3.3, which implements ASME Section XI Inservice Testing (IST) to evaluate a pump's performance against its pump curve to determine operability. The IST program is controlled by TS 3.3, and requires that testing of ASME Code Class 1, Class 2, and Class 3 pumps shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to TS 3.9(4) and the Basis Section only clarify the AFW flow paths that are required to be tested. The proposed change follows the recommendations of NUREG–0635, ''Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants,'' Recommendation GS–6(2). No physical changes are proposed, information is being added to clarify the testing required to meet the recommendations of NUREG–0635, therefore these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

There will be no physical alterations to the plant configuration or changes in operating modes. The proposed change to delete the specific discharge pressure of the AFW pumps from the TS is consistent with the ASME Code Section XI requirements that are controlled by TS 3.3. Testing requirements of TS 3.3 require testing of ASME Code Class 1, Class 2, and Class 3 pumps in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC. The clarifications being provided to describe the flow paths only provide additional information for testing required to meet the recommendation of NUREG–0635.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed changes will not result in any physical alterations to the plant configuration or changes to the application of setpoints or limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

*Attorney for licensee:* Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

*NRC Project Director:* William H. Bateman.

*Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey*

*Date of amendment request:* November 4, 1997.

*Description of amendment request:* The amendments would change the Emergency Diesel Generator (EDG) Technical Specification (TS) 3/4.8.1 to (1) delete 18-month surveillance requirement 4.8.1.1.2.d.1, and (2) eliminate the accelerated testing requirement of Table 4.8–1. Both changes have been approved on other nuclear power facilities.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change deleting the requirement for an 18 month diesel inspection is consistent with the improved Standard Technical Specifications (NUREG–1433) and does not result in any changes to the existing plant design. The Salem preventive maintenance program utilizes diesel generator performance history, engineering analyses and manufacturer's recommendations as appropriate for determining diesel generator inspection requirements. The Technical Specifications will continue to contain surveillance requirements that demonstrate the functional capability of the diesel generators. The change does not impact the ability of the diesel generators or the AC electrical power sources to perform their function, nor result in a significant increase in the consequences of any accident previously evaluated. The diesel generators will continue as designed.

PSE&G has implemented the provisions of the maintenance rule for EDG's, including the appropriate regulatory guidance. This provides a program which assures EDG performance. The elements of this program include the performance of detailed root cause analysis of individual failures, effective corrective actions taken in response to individual failures, and implementation of preventive maintenance consistent with the Maintenance Rule. Additionally, the proposed changes (elimination of accelerated diesel generator testing requirements of TS 4.8.1.1.a in lieu of monthly testing and deletion of special reporting requirements for diesel failures), do not delete the surveillance requirements but rather set their frequency at every 31 days. Monitoring the effectiveness of EDG maintenance and continuing surveillance testing will ensure that the diesel generators will perform their intended functions and will minimize failures. As is noted in the recommendations of GL [Generic Letter] 94–01, because PSE&G is monitoring and maintaining EDG performance in accordance with the provisions of 10 CFR 50.65, there is no longer a need for special reporting requirements.

Since the changes do not affect the assurance of diesel generator reliability or operability as discussed above, there is no significant increase in the probability or consequences of any accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.

This request does not result in any change to the plant design or does it involve a significant change in current plant operation. The diesel generators are inspected utilizing diesel generator operating history, engineering analyses and manufacturer's recommendations as appropriate, and the remaining surveillance requirements continue to demonstrate the functional capability of the diesel generators.

Changing the surveillance of frequency of TS 4.8.1.2.a to 31 days the existing frequency as determined by Table 4.8–1, does not create a new or different kind of accident. Deleting of special reporting requirements, appropriate in light of the monitoring and maintenance in conformance with 10 CFR 50.65, and reliance on the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73, does not create the possibility of a new or different kind of accident.

The proposed changes do not result in any change to the plant design nor do they involve a significant change in current plant design. No new failure modes will be introduced. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed request does not adversely impact the reliability of the diesel generators. As stated above, the diesel generator operating history, engineering analyses and the manufacturer's recommendations will be utilized as appropriate to perform diesel generator inspections. Additionally, other Technical Specification surveillance requirements will continue to demonstrate the functional capability of the diesel generators. The diesel generators will continue to perform their design functions.

Noting the monitoring and maintenance being performed in conformance with 10 CFR 50.65, revision of the frequency of surveillance testing of 4.8.1.1.2.a does not adversely impact the reliability of the diesel generators. Deletion of the special reporting requirements of 4.8.1.1.4 does not impact the operability or the reliability of the diesel generators.

This request does not involve an adverse impact on diesel generator operation or reliability. Since the diesel generator function is not affected by the proposed change, this request does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Salem Free Public library, 112 West Broadway, Salem, NJ 08079.

*Attorney for licensee:* Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

*NRC Project Director:* John F. Stolz.

*Southern Nuclear Operating Company, Inc, Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama*

*Date of amendments request:* October 16, 1997.

*Description of amendments request:* The proposed amendments would revise the Farley Nuclear Plant (FNP) Units 1 and 2 Technical Specifications (TS) to increase the allowable number of charging pumps capable of injecting into the reactor coolant system (RCS) when the temperature of one or more of the RCS cold legs is 180°F or less. The amendments would also modify the FNP TS to allow a maximum of two charging pumps to be capable of injecting into the RCS during pump swap operations.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to TS 3.1.2.3 allow two charging pumps to be capable of injecting into the reactor coolant system (RCS) for a period not to exceed 15 minutes while RCS cold leg temperature is at or below 180 degrees F. The intent is to allow the operator to start a second pump long enough to ensure that it operates properly and then to promptly secure the pump that was originally running. This order of pump operation will allow seal injection flow to be maintained to the RCS pumps number one seal continuously, thus preventing loss of pressure to the seals and maintaining filtered water flow through the seals. The proposed revised bases address the potential for [an] RCS mass addition transient. Guidance is given to prevent the charging pump swap from being conducted while the RCS is in a condition conducive to an overpressure transient. The RCS should be in a non water solid condition and the residual heat removal (RHR) relief valves must be operable or the RCS must be vented while the pump swap evolution is in progress. The proposed revision to TS 3.1.2.3 allows 15 minutes to have two pumps capable of injecting into the RCS, although two pumps will be running only momentarily, the remaining time is needed to perform the charging pump circuit breaker racking operations needed to render one of the two pumps incapable of injecting into the RCS. The proposed actions statement 3.1.2.3b directs that immediate action be taken to render all but one pump inoperable should the allotted 15 minutes be exceeded. This action is more appropriate than is currently specified. These proposed changes include sufficient controls to prevent an RCS overpressurization event.

Therefore, the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

This proposed change involves no change to the physical plant. It allows for a very limited and controlled operational change. The change increases the potential for a mass addition transient while the RCS is [at or] below 180 degrees F; however, sufficient controls are proposed to prevent a cold overpressure event.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The proposed change includes controls sufficient to prevent a significant reduction in the possibility or consequences of an accident. The proposed change specifies that the pump swap evolution be performed under conditions that will prevent an adverse plant transient. In addition, the proposed revision provides appropriate operator action that does not currently exist. This change is consistent with NUREG 1431, Standard Technical Specifications-Westinghouse Plants.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama.

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama.

*NRC Project Director:* Herbert N. Berkow.

**Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing**

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

*Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida*

*Date of application for amendment:* October 4, 1997.

*Brief description of amendment:* The proposed amendment revises the description of the electrical controls for Operating Reactor Building Recirculation System Fan/Cooler contained in the Final Safety Analysis Report and Improved Technical Specification Bases.

*Date of publication of individual notice in the* **Federal Register**: November 13, 1997 (62 FR 60921).

*Expiration date of individual notice:* December 15, 1997.

*Local Public Document Room location:* Coastal Region Library, 8619 W. Crystal River, Florida 34428.

*Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida*

*Date of application for amendment:* October 31, 1997.

*Brief description of amendment:* The proposed amendment revises Operating License No. DPR–72, License Condition 2.C.(5) and deletes the requirement for installation and testing of flow indicators in the emergency core cooling system.

*Date of publication of individual notice in the* **Federal Register**: November 12, 1997 (62 FR 60733).

*Expiration date of individual notice:* December 12, 1997.

*Local Public Document Room location:* Coastal Region Library, 8619 W. Crystal River, Florida 34428.

*Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida*

*Date of application for amendment:* October 31, 1997.

*Brief description of amendment:* The proposed amendment involves revisions to the Crystal River 3 Technical Specifications (TS) relating to decay heat removal requirements in Mode 4.

*Date of publication of individual notice in the* **Federal Register**: November 12, 1997 (62 FR 60735).

*Expiration date of individual notice:* December 12, 1997.

*Local Public Document Room location:* Coastal Region Library, 8619 W. Crystal River, Florida 34428.

*Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida*

*Date of application for amendment:* October 31, 1997.

*Brief description of amendment:* The proposed amendment involves revisions to the Crystal River 3 Technical Specifications (TS) relating to the methodology for post-loss of coolant accident boron precipitation prevention.

*Date of publication of individual notice in the* **Federal Register**: November 12, 1997 (62 FR 60731).

*Expiration date of individual notice:* December 12, 1997.

*Local Public Document Room location:* Coastal Region Library, 8619 W., Crystal River, Florida 34428.

**Notice of Issuance of Amendments to Facility Operating Licenses**

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document

Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

*Duke Energy Corporation, et al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina*

*Date of application for amendment:* May 27, 1997.

*Brief description of amendment:* The amendment deletes references to steam generator tube sleeving and repair criteria that will not be used for the Westinghouse Model D5 steam generators in use at Catawba Unit 2. Also, unused paragraph numbers have been deleted and a typographical error has been corrected.

*Date of issuance:* November 13, 1997.

*Effective date:* As of the date of issuance to be implemented within 30 days.

*Amendment No.:* 154.

*Facility Operating License No. NPF–52:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: June 18, 1997 (62 FR 33122).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* York County Library, 138 East Black Street, Rock Hill, South Carolina.

*Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana*

*Date of amendment request:* October 7, 1997.

*Brief description of amendment:* The amendment changes the Appendix A Technical Specifications (TSs) by modifying TS 3.3.3.7.3, and Surveillance Requirements (SR) 4.3.3.7.3 for the broad range gas detection system. Also it makes some changes to the Bases in section 3/4.3.3.7 to incorporate information associated with the existing toxic gas monitors.

*Date of issuance:* November 14, 1997.

*Effective date:* November 14, 1997, to be implemented within 60 days.

*Amendment No.:* 135.

*Facility Operating License No. NPF–38:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: October 15, 1997 (62 FR 53660).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

*Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana*

*Date of amendment request:* February 6, 1996.

*Brief description of amendment:* The proposed change will amend the Allowable Values of parameters in Table 3.3–4 of Waterford Steam Electric Station, Unit 3, (Waterford 3) Technical Specifications (TSs) to make it consistent with the identical parameters in Table 2.2–1 of TSs for Waterford 3. The proposed change will add Mode 4 to surveillance requirements of Table 4.3–2, Item 5.c (Safety Injection System Automatic Actuation Logic) that was inadvertently removed. Finally, the proposed change removes a reference to TS 3.3.3.2 in Surveillance Requirements TS 4.10.2.2 and 4.10.4.2 since Incore Detectors has been removed from the TSs.

*Date of issuance:* November 20, 1997.

*Effective date:* November 20, 1997, to be implemented within 60 days.

*Amendment No.:* 136.

*Facility Operating License No. NPF–38:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: June 5, 1996 (61 FR 28615).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 20, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

*Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas*

*Date of amendment request:* August 23, 1996, as supplemented by letters dated October 1 and 15, 1996, and January 28, 1997.

*Brief description of amendments:* The amendments reflect the approval of the transfer of the authority to operate South Texas Project, Units 1 and 2, under the licenses to a new operating company, South Texas Project Nuclear Operating Company.

*Date of issuance:* November 17, 1997.

*Effective date:* November 17, 1997.

*Amendment Nos.:* Unit 1—Amendment No. 93; Unit 2—Amendment No. 80.

*Facility Operating License Nos. NPF–76 and NPF–80.* The amendments revised the Technical Specifications and the operating licenses.

*Date of initial notice in* **Federal Register**: November 7, 1996 (61 FR 57719).

The additional information contained in the supplemental letter dated January 28, 1997, was clarifying in nature and thus, it was within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 17, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

*Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut*

*Date of application for amendment:* May 5, 1997.

*Brief description of amendment:* Technical Specification Surveillance 4.8.4.1 requires periodic testing of lower voltage circuit breakers for all containment penetration conductor overcurrent protective devices. The amendment modifies the requirements for determining the operability of lower voltage circuit breakers by using the manufacturer's curve of current versus time to test delay trip elements, clarifies the use of two pole in series testing, and expands the Bases description of the testing.

*Date of issuance:* November 14, 1997.

*Effective date:* As of the date of issuance, to be implemented within 60 days.

*Amendment No.:* 153.

*Facility Operating License No. NPF–49:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: June 4, 1997 (62 FR 30637).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385.

*Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota*

*Date of application for amendments:* May 15, 1997, as supplemented August 29, October 20, October 24, and October 28, 1997.

*Brief description of amendments:* The amendments revise certain Technical Specification (TS) limitations on reactor coolant system leakage and steam generator tube surveillance, and implement a voltage-based repair criteria per requirements of NRC Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.'' In addition, the amendments correct a typographical error in TS Section 4.12.c.

*Date of issuance:* November 18, 1997.

*Effective date:* November 18, 1997, with full implementation of the Technical Specifications within 30 days. License Condition 5 of Appendix B shall be implemented immediately upon issuance of the amendments.

*Amendment Nos.:* 133 and 125.

*Facility Operating License Nos. DPR–42 and DPR–60:* Amendments revised the licenses and the Technical Specifications.

*Date of initial notice in* **Federal Register**: August 13, 1997 (62 FR 43371).

The August 29, October 20, October 24, and October 28, 1997, supplements provided clarifying information that did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

*Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York*

*Date of application for amendment:* August 19, 1997, as supplemented September 17, 1997.

*Brief description of amendment:* The amendment revised the Ginna Station Improved Technical Specifications to correct an error in the required accumulator borated water volume specified in Surveillance Requirement 3.5.1.2.

*Date of issuance:* November 10, 1997.

*Effective date:* November 10, 1997.

*Amendment No.:* 69.

*Facility Operating License No. DPR–18:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: October 8, 1997 (62 FR 52587).

The September 17, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 10, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

*The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio*

*Date of application for amendment:* October 24, 1996, as supplemented June 16 and October 2, 1997.

*Brief description of amendment:* This amendment revised the minimum critical power ratio safety limit to reflect the 10 CFR Part 21 condition reported by General Electric in their letter to the NRC dated May 24, 1996.

*Date of issuance:* November 7, 1997.

*Effective date:* November 7, 1997.

*Amendment No.:* 91.

*Facility Operating License No. NPF–58:* This amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: February 12, 1997 (62 FR 6569). The June 16 and October 2, 1997, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 7, 1997.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

Dated at Rockville, Maryland, this 25th day of November 1997.

For the Nuclear Regulatory Commission.

**Elinor G. Adensam,**

*Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.*

[FR Doc. 97–31522 Filed 12–2–97; 8:45 am]

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## SECURITIES AND EXCHANGE COMMISSION

### Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension: Rule 13e–1, SEC File No. 270–255, OMB Control No. 3235–0305; Rule 12g3–2, SEC File No. 270–104, OMB Control No. 3235–0119; Trust Indenture: Act Rules, SEC File No. 270–115, OMB Control No. 3235–0132.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.) the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget requests for extension of the previously approved collections of information discussed below.

''Purchase of Securities by issuer thereof under the Securities Exchange Act of 1934''. Rule 13e–1 under the Exchange Act is designed to provide shareholders and the marketplace with relevant information concerning issuer repurchases during a tender offer for its securities by a third party. Public companies are the respondents. An estimated 20 respondents will file submissions annually at an estimated 13 hours per response for a total annual burden of 260 hours.

''Securities Exchange Act of 1934—Rule 12g3–2.'' Rule 12g3–2 provides an exemption for certain foreign securities. It affects approximately 1,800 foreign issuer respondents at an estimated one burden hour per response for a total annual burden of 1,800 hours.

''Requirements as to Form and Content of Applications, Statements and Reports under the Trust Indenture Act of 1939.'' Rules 7a–15 through 7a–37 under the Trust Indenture Act of 1939 (''TIA'') provides guidance for complying with requirements under the TIA. Persons and entities subject to TIA requirements are the respondents. No information collection burdens are imposed directly by these rules so they are assigned only one burden hour for administrative convenience.

An agency may not conduct or sponsor, and a person is not required to respond to, a collection of information