additional information that is developed. As a result of this withdrawal action, the NRC is postponing the public meeting announced in the notice published on September 4. In addition, the opportunity for a hearing that was published as part of the notice is withdrawn pending further NRC action on the matter.

Dated at Rockville, Md., this 18th day of September 1997.

For the Nuclear Regulatory Commission,

John W.N. Hickey,

Chief, Low-Level Waste and Decommissioning Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 97–25317 Filed 9–23–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of September 22, 29, October 6, and 13, 1997.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of September 22

There are no meetings scheduled for the week of September 22.

Week of September 29—Tentative

There are no meetings scheduled for the week of September 29.

Week of October 6—Tentative

There are no meetings scheduled for the week of October 6.

Week of October 13—Tentative

Tuesday, October 14

10:00 a.m.
Briefing on EEO Program (Public Meeting) (Contact: Irene Little, 301–415–7380)

1:00 p.m.
Briefing on Severe Accident Master Integration Plan (Public Meeting)

Wednesday, October 15

10:00 a.m.
Briefing on PRA Implementation Plan (Public Meeting) (Contact: Gary Holahan, 301–415–2884)

11:30 a.m.
Affirmation Session (Public Meeting) (if needed)

*The schedule for commission meetings is subject to change on short notice. To verify the status of meetings call (recording) (301) 415–1292. Contact person for more information: Bill Hill (301) 415–1661.

* * * * *

The NRC Commission Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm.

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661).

In addition, distribution of this meeting notice over the internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

* * * * *

Dated: September 19, 1997.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 97–25444 Filed 9–22–97; 8:45 am]

BILLING CODE 7590–01–M

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 August 29, 1997, through September 12, 1997. The last biweekly notice was published on September 10, 1997 (62 FR 47696).

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 October 24, 1997, 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.

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: July 15, 1997

Description of amendment request: The proposed amendments would: (1) add Technical Specification (TS) 3.5.7, ''Main Steam Line Break Detection and Feedwater Isolation,'' to identify operability requirements and Bases for the main steamline break (MSLB) detection isolation circuitry, the feedwater isolation circuitry, the main feedwater main control valves, and the main feedwater startup control valves; (2) revise TS 3.5.1, ''Operation Safety Instrumentation'' to add a reference to TS 3.5.7; (3) revise Table 3.5.1-1, ''Instruments Operating Conditions,'' to reflect operability requirements for the main steam header pressure and MSLB detection channels, the feedwater isolation channels, and the feedwater isolation channels manual pushbuttons; and (4) revise Table 4.1-1, ''Instrument Surveillance Requirements,'' and Table 4.1-2, ''Minimum Equipment Test Frequency,'' to include surveillance requirements for the subject circuitry and components.

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:

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

NO

This proposed Technical Specification amendment does not create any conditions or events which lead to accidents (events) previously evaluated in the UFSAR [Updated Final Safety Analysis Report], other than a loss of Main Feedwater (FDW). The new MSLB detection and feedwater isolation circuitry addressed by this change is designed so that a credible single failure will not cause a loss of FDW to the steam generator unless [an] MSLB is detected. Single failures are not assumed if entry into a Technical Specification action statement occurs.

During [an] MSLB, the circuitry is intentionally stopping and isolating FDW. Operators are currently instructed to isolate FDW on indication of [an] MSLB. The new circuitry will automatically stop FDW to eliminate the need for this operator action. Thus the probability of the stopping (loss) of FDW is not increased. The NRC has also stated that the stopping of FDW to mitigate [an] MSLB is an acceptable response to address the concerns of Inspection and Enforcement Bulletin 80-04.

The Emergency Feedwater (EFW) System is an accident mitigation system. The MSLB modification and associated Technical Specification to keep the turbine driven emergency feedwater pump (TDEFW) pump from starting following [an] MSLB will not initiate any accidents.

The potential for containment overpressurization currently exists without the installed modification and associated Technical Specification. The new MSLB detection and feedwater isolation circuitry will assist in reducing the potential for the overpressurization of containment. The EFW circuitry is designed so that the TDEFWP will still auto start for any event other than [an] MSLB. The TDEFWP can still be manually started during [an] MSLB or FDW line break accident as needed. This action is similar to other manual actions to align EFW for the MSLB scenarios that are already described in the ONS [Oconee Nuclear Station] UFSAR. This new circuitry and associated Technical Specification creates no new credible single failures that could prevent the TDEFWP from auto starting (except for the MSLB). The motor driven EFW pumps and EFW flow control valves are not adversely affected by this change and will provide EFW flow for scenarios other than Station Blackout. Both FDW and EFW will still provide their design functions of supplying feedwater to the steam generators, as evaluated in the UFSAR. The ability to shut down following a 10CFR50 Appendix R fire is not adversely affected. This Technical Specification change does not adversely affect containment integrity and radiological release pathways.

B. Create the possibility of a new or different kind of accident from the accident previously evaluated?

NO

No accidents different than already evaluated in the UFSAR are postulated. The FDW System will still perform its design function of supplying feedwater to the steam generators as evaluated in the UFSAR. The EFW System will still provide its function of supplying feedwater to the steam generators, as evaluated in the UFSAR for events resulting in the loss of the FDW System.

C. Involve a significant reduction in a margin of safety?

NO

The design pressure of containment is specified to be 59 psig in the bases to several Technical Specifications. With the potential for unrestricted FDW and EFW flow during [an] MSLB inside containment, the design pressure of the containment could be exceeded. The proposed Technical Specifications address equipment which will function to isolate FDW in the unlikely event of [an] MSLB accident. Therefore, the proposed Technical Specifications do not increase the potential for the containment to be pressurized or increase the expected pressure of containment following [an] MSLB. No plant safety limits, set points, or design parameters are adversely affected. The fuel, fuel cladding, and Reactor Coolant System are not impacted.

Duke [Duke Energy Corporation] has concluded based on the above that there are no significant hazards considerations involved in this amendment request.

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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

Duke Power Company, Docket Nos. 50-269, 270 and 50-287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South Carolina

Date of amendment request: August 28, 1997 (TSC 96-09)

Description of amendment request: The proposed changes would add new limiting conditions for operation and new surveillance requirements for the Emergency Condenser Circulating Water System, the Essential Siphon Vacuum System, and the Siphon Seal Water System to reflect design changes and modifications to these systems.

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

NO.

This Technical Specification change does not create any conditions or events which lead to accidents previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. The new ECCW [Emergency Condenser Circulating Water] System Technical Specification 3.19, along with the new ECCW Surveillance requirements specified in Technical Specification Table 4.1-2, are conservative in nature. No existing Technical Specification requirements are being deleted with this revision. Surveillance and operability requirements are being added for the upgraded ECCW System.

The ECCW System is only required following the occurrence of loss of offsite power (LOOP) events. The most limiting of these LOOP events is the loss of coolant accident concurrent with the LOOP (LOCA/LOOP). Therefore, the ECCW System is not considered to be an accident initiator. As a result, the proposed new ECCW Technical Specification requirements will not result in any increase in the probability of any design basis accidents or events evaluated in the UFSAR.

The credit for restarting a CCW [Condenser Circulating Water] pump within 1.5 hours following a LOOP, to ensure suction to LPSW [Low Pressure Service Water] is maintained, is being replaced by credit for maintaining the ECCW siphon using the new siphon support systems (ESV [Essential Siphon Vacuum] System and SSW [Siphon Seal Water] System) in conjunction with the upgraded ECCW System. Therefore, obsolete requirements specified in Selected Licensee Commitments (SLCs) 16.9.7 and 16.9.8 will be revised or deleted accordingly. Replacement of the CCW pump restart during a LOOP with the ability to maintain ECCW siphon flow will not create any conditions or events which lead to accidents previously evaluated in the UFSAR.

The modifications to upgrade the ECCW System were performed to improve the reliability of the ECCW System. The proposed new ECCW Technical Specification provides additional surveillance and operability requirements to ensure that the upgraded ECCW System will function reliably during the design basis events which require its operation. Therefore, these proposed new Technical Specification requirements will not increase the consequences of any accidents previously evaluated in the UFSAR.

[2. Will the change] create the possibility of a new or different kind of accident from the accident previously evaluated?

NO.

No accidents different than those already evaluated in the UFSAR are postulated. The upgraded ECCW System will more reliably perform its design function of supplying water to the suction of the Low Pressure Service Water (LPSW) System as evaluated in the UFSAR. The new Technical Specification requirements will increase the reliability of the upgraded ECCW System. In addition, the ECCW System is not an accident initiator since it is used following certain design basis events such as a LOCA/LOOP.

[3. Will the change] involve a significant reduction in a margin of safety?

NO.

The proposed Technical Specifications address equipment which will function in certain design basis events, such as a LOCA/LOOP, to ensure a reliable water supply to the LPSW System. The LPSW System must function to remove decay heat from primary systems and the reactor building during a LOCA/LOOP. The proposed Technical Specifications addressing the upgraded ECCW System will further enhance the reliability of the ECCW System and will result in greater assurance that the LPSW System can perform its safety functions. No plant safety limits, setpoints, or design parameters are adversely affected. The fuel, fuel cladding, and Reactor Coolant System are not impacted. The proposed Technical Specifications provide additional, conservative, operational requirements beyond the current Technical Specifications which address the ECCW System.

Duke [Duke Energy Corporation] has concluded based on this information that there are no significant hazards considerations involved in this amendment request.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

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: September 4, 1997

Description of amendment request: The proposed changes would incorporate changes to the Oconee Final Safety Analysis Report and Technical Specification Bases to address a potential unreviewed safety question associated with implementation of revised small break loss-of-coolant accident analysis. The proposed changes would address operation of the facility and single failure criteria related to the high pressure injection system.

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

No. None of the proposed changes [have] any impact upon the probability of any accident which has been evaluated in the UFSAR [Updated Final Safety Analysis Report].

None of these changes have any impact upon the ability of the HPI [high-pressure injection] System to mitigate the consequences of a small break LOCA [loss-of-coolant accident], which is addressed below. The small break LOCA is the limiting design basis accident with respect to the HPI System operability requirements.

The proposed changes to the Bases of Specification 3.3.1 and Chapter 15 of the Oconee UFSAR include operator actions that have not previously been reviewed and approved by the [NRC] staff for licensing basis small break LOCA analyses. However, these operator actions have been included in the Emergency Operating Procedure for over 10 years and crediting these actions in the safety analyses does not result in any change to the operator's response to a small break LOCA. These actions are simply changes to the assumptions contained in the licensing basis small break LOCA analyses. The operability requirements for the HPI System contained in Specification 3.3.1 are supported by a spectrum of small break LOCA analyses based on the approved Evaluation Model described in FTI [Framatome Technologies, Inc.] topical report BAW-10192P. These small break LOCA analyses demonstrate that the acceptance criteria of 10CFR 50.46 are satisfied.

The operability requirements in Technical Specification 3.3.1.c assure that the HPI System can withstand the worst single failure and still result in two HPI pumps injecting through two trains. The full power small break LOCA analyses supporting this proposed license amendment have been performed in accordance with the approved Evaluation Model described in FTI topical report BAW-10192P.

When at or below 75% FP [full power], one HPI train provides sufficient flow to mitigate a small break LOCA. The 60% power level currently in Specification 3.3.1 is justified by analyses using the Evaluation Model described in FTI topical report BAW-10192P, considering the worst case break location and size described in LER [Licensee Event Report] 269/90-15 and Attachment 2 to this submittal. The proposed changes to the Bases of Technical Specification 3.3.1 describe the operator actions credited to justify the adequacy of the current specification and eliminate the need for the administrative restrictions imposed by LER 269/90-15. These requirements ensure that, following the worst single failure, one train of HPI would remain available to mitigate a small break LOCA.

In summary, the technical analyses described in this license amendment justify the adequacy of this specification and assure that operability of the HPI System is maintained in a manner consistent with the requirements of the design basis accidents. Therefore, it is concluded that this amendment request will not significantly increase the probability or consequences of an accident previously evaluated.

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

No. The proposed changes to the Bases of Technical Specification 3.3.1 and Chapter 15 of the Oconee UFSAR do not result in any new operator actions or changes in plant operation. The proposed changes involve crediting operator actions in the licensing basis small break LOCA analyses that have been included in the Emergency Operating Procedure for years. No new initiating events or potentially unanalyzed conditions have been created. Therefore, this proposed amendment will not create the possibility of any new or different kind of accident.

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

No. The HPI System requirements associated with the proposed UFSAR and Technical Specification Bases changes are supported by analyses which demonstrate that the acceptance criteria of 10 CFR 50.46 are not violated for any small break LOCA. These analyses were performed in accordance with the Evaluation Model described in FTI topical report BAW-10192P. Therefore, it is concluded that the proposed amendment request will not result in a significant decrease in the margin of safety.

Duke [Duke Energy Corporation] has concluded, based on the above, 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 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: August 6, 1997

Description of amendment request: The proposed amendment would eliminate the provisions in Technical Specification 3.8.1, ''AC Sources - Operating,'' for accelerated testing of the emergency diesel generators (DG). The proposed changes are the following: (1) the frequency of verifying DG starts and operation in Surveillance Requirements 3.8.1.2 and 3.8.1.3, respectively, would be changed to 31 days, from the present reference to Table 3.8.1-1, and (2) Table 3.8.1-1, ''Diesel Generator Test

Schedule,'' would be deleted. The emergency DG provide emergency AC power to the site with the loss of offsite AC power.

Basis for proposed no significant hazardsconsideration 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. This request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

[These] change[s] will provide flexibility to structure the standby diesel generator maintenance program based on the risk significance of the structures, systems, and components [(SSCs)] that are within the scope of the Maintenance Rule [(10 CFR 50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants)]. The removal of the diesel generator accelerated testing is acceptable as the maintenance rule applies site and system specific performance criteria to monitor diesel generator performance. This criteria includes a running availability and reliability goal as well as specific goals to monitor maintenance preventable functional failures. The performance criteria for the diesel generator reliability and availability established by the maintenance rule and the causal determinations and corrective actions required for maintenance preventable functional failures are considered to be an acceptable method for monitoring diesel generator performance.

The proposed change[s] [have] no effect on the probability of the initiation of an accident, because the emergency diesel generators do not serve as the initiator of any event. Additionally, as diesel generator performance will continue to be [ensured] by the maintenance rule, the proposed changes do not affect the ability to mitigate the consequences of an accident previously evaluated. The changes do not impact the diesel [generator]'s design sources, operating characteristics, system functions, or system interrelationships. The failure mechanisms for the accident previously evaluated are not affected and no additional failure modes are created that could cause an accident that has been previously evaluated. Since the diesel generator performance and reliability will continue to be [ensured] by the maintenance rule, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

[These] proposed change[s] [do] not involve a change to the plant design or operation. As a result, the proposed change[s] [do] not affect any of the parameters or conditions that could contribute to the initiation of any accidents. The proposed changes only affect the methods used to monitor and [ensure] diesel generator performance. The performance criteria for both the diesel generator reliability and unavailability established by the maintenance rule, and the causal determinations and corrective actions required for maintenance preventable functional failures, [are] considered by [the Nuclear Regulatory Commission (NRC) in] GL [(Generic Letter)] 94-01[, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,'' issued May 31, 1994,] to be an acceptable method for monitoring diesel generator performance.

No SSC, method of operation, or system interface is altered by [these] change[s]. The changes do not impact the diesel [generator]'s design sources, operating characteristics, system functions, or system interrelationships. The failure mechanisms for the accidents are not affected, and no additional failure modes are created. Because the diesel generator performance and reliability will continue to be [ensured] by the maintenance rule, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. This request does not involve a significant reduction in a margin [of] safety.

The proposed changes only affect the methods used to monitor and [ensure] diesel generator performance and reliability. The performance criteria for the diesel generator reliability and availability established by the maintenance rule, and the causal determinations and corrective actions required for maintenance preventable functional failures, [are] considered by [NRC in] GL 94-01 to be an acceptable method for monitoring diesel generator performance. No margin [of] safety as defined in the bases for any technical specification is impacted by these changes. [These] change[s] [do] not impact any uncertainty in the design, construction, or operation of any SSC. Diesel generator response to accident initiators is unchanged. No SSC, method of operating, or system interface is altered by [these] change[s]. The changes do not impact the diesel [generator]'s design sources, operating characteristics, system functions, or system interrelationships. Because the diesel generator performance and reliability will continue to be [ensured] by the maintenance rule, the proposed changes do not involve a significant reduction in the 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: 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, N.W., 12th Floor, Washington, DC 20005-3502

NRC Project Director: James W. Clifford, Acting

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

Date of amendment request: August 26, 1997

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 (CR3) Technical Specifications Bases (TSB) to change the design basis of the Emergency Diesel Generator (EDG) Air Handling System. Specifically, TSB Sections B 3.8.1 and B 3.8.2 would be revised to indicate that a single or dual fan operation depending upon fan supply air temperature, would maintain the temperature of the EDG engine and control rooms within the EDG manufacturer's limits.

Basis for proposed no significant hazardsconsideration determination:

The EDG Air Handling System provides continuous ventilation, and dissipates internal heat gains in the EDG engine and control rooms when the diesel is operating. Presently, the CR3 plant documentation requires operation of only one cooling fan per room to maintain the EDG room temperature within the manufacturer's limit and is inconsistent with the Final Safety Analysis Report (FSAR) which requires operation of two fans.

As part of its EDG upgrade to increase their service ratings and associated cooling analysis, the licensee has determined that operation of either a single or dual cooling fans depending upon fan supply air temperature, would achieve the required room cooling limits. The licensee has determined that reliance on the operation of two cooling fans instead of one involves an unreviewed safety question and requires a license amendment.

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 not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a significant increase in the probability of an accident previously evaluated. The EDG room cooling fans support operation of the EDGs which are used to mitigate design basis accidents. Although EDG availability is a contributor to the risk of station blackout, the CR-3 licensing basis assumes a station blackout without regard to EDG reliability. Therefore, the probability of previously evaluated accidents is not significantly increased.

For design basis accidents, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated. The proposed change to operate both cooling fans for each EDG to provide adequate ventilation potentially increases the probability of malfunction of equipment important to safety. However, the proposed changes do not affect the independence of the EDGs or the independence of the EDG Air Handling System and, based on single failure criteria, one EDG will be fully operable and capable of meeting its mission at all times as required by the CR-3 Technical Specifications. Therefore, no significant increase in the consequences of an accident previously evaluated, including the offsite radiological dose exists.

Based on the above, the probability of an accident previously evaluated has not been significantly increased, and this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does not create the possibility of a new or different kind of accident from any accident previously evaluated

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Neither the fans nor the EDGs are initiators of any new accidents. The EDG room cooling fans support operation of the EDGs, which are used to mitigate design basis accidents. Reliance on two fans rather than one has reduced the redundancy of the EDG Air Handling System and increased the probability of a malfunction of an EDG. However, the proposed changes do not affect the independence of the EDGs or the independence of the EDG Air Handling System and, based on single failure criteria, one EDG will be fully operable and capable of meeting its mission at all times as required by the CR-3 Technical Specifications. Results of analyses to evaluate the failure of an EDG to operate following a design basis accident are documented in the FSAR. Therefore, this change does not create the possibility of a new or different kind of accident.

3. Does not involve a significant reduction in the margin of safety

The proposed change does not involve a significant reduction in the margin of safety. The EDG room cooling fans support operation of the EDGs. Following this change, two fans will be required to maintain the EDG engine room and EDG control room temperatures within the design basis limit when the fan supply air temperature is greater than or equal to 85°F. Reliance on two fans rather than one has reduced the redundancy of the EDG Air Handling System and slightly increased the probability of malfunction of an EDG, but only after it has run for some period of time. However, the proposed changes do not affect the independence of the EDGs or the independence of the EDG Air Handling System and, based on single failure criteria, one EDG will be fully operable and capable of meeting its mission at all times as required by the CR-3 Technical Specifications. Therefore, this change does not result in a significant reduction to the margin safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC - A5A, P. O. Box 14042, St. Petersburg, Florida 33733-4042

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: September 9, 1997

Description of amendment request: The proposed amendment would revise the Crystal River 3 (CR3) Final Safety Analysis Report (FSAR) to reflect the revised analysis for the hypothetical Makeup System Letdown Line Failure Accident. In the original analysis, the event was modeled as being terminated by an automatic isolation of the failed letdown line on low reactor coolant system pressure. The revised analysis has modeled the event as being terminated by manual operator action to isolate the line. The licensee has determined that reliance on a manual operator action in place of the automatic action involves an unreviewed safety question (USQ) and requires prior Nuclear Regulatory Commission (NRC) approval. Other FSAR changes are being proposed to clarify that this accident is a hypothetical event that is presented only to demonstrate that the dose consequences are below 10 CFR Part 100 limits. The licensee submitted its proposed FSAR changes which, upon NRC approval, will be incorporated in the next revision to the FSAR.

Basis for proposed no significant hazardsconsideration 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 not involve a significant increase in the probability or consequences of an accident previously evaluated.

This change involves a revision to the analysis for the Makeup System Letdown Line Failure Accident. The revised analysis assesses the resultant change in consequences of this event based on the actions specified in EOP-3 [Emergency Operating Procedure - 3] to manually isolate the letdown line failure. No changes have been made to any precursors to this event. Therefore, the probability of an accident previously evaluated has not been increased.

This change has resulted in an increase in the calculated doses due to the greater release of reactor coolant prior to termination of the leak. Although the doses have increased, they remain significantly less than the limits of 10 CFR 100. These doses also remain lower than the resultant doses for the design basis LOCA [loss-of-coolant-accident].

The revised analysis evaluates the consequences of this accident based on the replacement of the automatic isolation of the letdown line with a manual operator action to isolate the letdown line. This action was added to EOP-3 when it was identified that the manual initiation of the HPI [high pressure injection] system directed by the EOP would interfere with the automatic isolation signal assumed to terminate this event. Manual initiation of the HPI system for a LSCM [loss of subcooling margin] event is consistent with the symptomatic philosophy of the EOPs. This philosophy is utilized in order to manage a wide range of event/leaks that would be indicated by a LSCM. Early initiation of the HPI system is intended to ensure adequate core cooling as the primary concern during a LSCM event.

Prior to the addition of the EOP step to manually isolate the letdown line, the EOP directed actions towards locating and isolating the source of the leak resulting in the LSCM. However, due to the potential significance of the letdown line failure which can result in RCS [reactor coolant system] leakage outside the reactor building, the manual action was added early in EOP-3 to isolate the letdown line. This action is proactive in ensuring early isolation of the potential leakage path and is consistent with the concept of a ''simple'' operator action (Reference 9) [NRC to Florida Power Corporation letter, Long-term modifications regarding emergency core cooling system Small Break Analysis problem, dated September 26, 1978].

Crediting a manual operator action instead of the automatic isolation introduces the possibility of a malfunction of a different type (i.e., operator error). The revised analysis assumes that operator action to isolate the letdown line occurs 10 minutes following a LSCM. Although the probability of operator error during this action may be greater than the probability of the failure of the automatic function, the consequences of this error would be small. Several indications would be available to the operator to identify the continued loss of coolant through this line. As discussed above, the radiological dose calculated by this event remains a small fraction of the limits of 10 CFR Part 100. Therefore, adequate time would exist for the identification of an operator error and correction of this error before any significant increase in the consequences of this event would occur.

Additionally, the probability for operator error in this event is considered to be small due to the extensive training plant operators receive regarding the EOPs and the simple nature of the action. Validation of the required actions in the EOPs, including isolation of the letdown line, is performed on the plant simulator to ensure the validity of the EOPs as well as to ensure that these actions can be performed as required.

The clarification added to FSAR Section 5.4.4.2 and 14.2.2.6.1 reflects the previously approved evaluation for pipe rupture criteria outside the reactor building for CR-3. A break in the high energy portion of the letdown line outside containment is not considered a credible event. This accident is presented only to demonstrate that the dose consequences from a postulated break in the letdown line outside containment remain below the 10 CFR Part 100 limits.

Based on the above, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This change does not involve any modification to the plant nor a change in the operation of the plant prior to the postulated failure of the letdown line. This change only evaluates the radiological dose consequences of the actions taken following the line failure. The addition of the action to manually isolate the letdown line for a LSCM event is consistent with the need to isolate potential RCS leakage paths and replaces the automatic isolation that was previously assumed to occur. Therefore, this change does not create the possibility of a new or different kind of accident.

3. Does not involve a significant reduction in the margin of safety.

This change does not result in a reduction to the margin of safety as defined in the Bases for any Technical Specifications. As discussed above, the radiological doses for the revised analysis have increased but remain a small fraction of the 10 CFR Part 100 limits.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC - A5A, P. O. Box 14042, St. Petersburg, Florida 33733-4042

NRC Project Director: Frederick J. Hebdon

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: August 22, 1997

Description of amendment request: The proposed amendment revises Technical Specification (TS) 4.0.5, Surveillance Requirements for Inservice Inspection and Testing of ASME Code Class 1, 2, and 3 components, to relocate the Inservice Testing Program requirements from TS 4.0.5 to the Administrative Controls Section 6.8, Procedures and Programs. The proposed amendment also provides conforming changes to several Surveillance Requirements to change the reference from TS 4.0.5 to the Inservice Testing Program.

Basis for proposed no significant hazardsconsideration 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) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no changes to the testing and evaluation related to pumps and valves in the Inservice Testing Program. The only substantive change allows the implementation of alternate testing provisions where Code-requirements are impractical and the NRC has not formally provided written approval. Since impractical testing would not be performed in any event, the actual testing program is unaffected.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of the modified specifications cannot create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to implementation of this administrative change since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The operating limits and functional capabilities of the affected systems, structures, and components remain unchanged by the proposed amendments, therefore, these changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981-5596

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408-0420

NRC Project Director: Frederick J. Hebdon

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Dates of amendment request: August 27, 1997

Description of amendment request: The licensee proposed modifying the Turkey Point Units 3 and 4 Technical Specifications (TS) to delete a sentence from section 6.2.2.f and add clarification to section 6.2.2.f of the Administrative section of TS to allow the use of up to 12 hour shifts without routine heavy use of overtime.

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) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a physical or procedural change to any structure, system or component that significantly effects the probability or consequences of any accident or malfunction of equipment important to safety. The proposed changes will allow the use of 12 hour shifts for a nominal 40 hours per week.

This change is only administrative in nature and has no significant impact on the probabilities or consequences of any evaluated accident or malfunction of equipment important to safety.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment will not change the physical plant or modes of plant operation defined in the Turkey Point Units 3 and 4 operating license. The proposed amendment will not involve addition or modification of permanent equipment for any systems structures or components at Turkey Point.

The change does modify the controls on working shift hours for operating personnel without significantly changing the hours worked per week and retains the current limitations on excessive overtime. The changes are administrative in nature.

Consequently, operation of either unit in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed amendment will allow the use of 12 hour shifts by virtue of the administrative change. This will result in fewer turnovers per day and will allow more contiguous days off between work shifts. The sum of these 12 hour work shift features will be more rested crews with better communications between shifts. The proposed change will not alter the basis for any Technical Specification that is related to the establishment of, or maintenance of, a nuclear safety margin.

Consequently, operation of Turkey Point Units 3 and 4 in accordance with this proposed amendment would not involve a significant reduction in margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Florida International University, University Park, Miami, Florida 33199

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: Frederick J. Hebdon

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 18, 1997

Description of amendment request: The proposed amendment would revise Technical Specification 3.7.1.6, Atmospheric Steam Relief Valves, to ensure the automatic feature of the steam generator power operated relief valve remains operable during Modes 1 and 2. In addition, the proposed change adds a surveillance requiring that a channel calibration on the steam generator power operated relief valve be performed every 18 months.

Basis for proposed no significant hazardsconsideration 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 South Texas Project proposed to revise Technical Specification 3.7.1.6 to ensure the automatic feature of the Steam Generator Power Operated Relief Valve remains operable during Modes 1 and 2. The South Texas Project has evaluated this proposed amendment and determined that it involves no significant hazards considerations based on the following:

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

The methodologies used in the accident analyses remain unchanged. The automatic actuation of the Steam Generator Power Operated Relief Valves is not a new design feature. The effects of the inadvertent opening of a Steam Generator Power Operated Relief Valve are currently analyzed as described in Section 15.1.4 of the Updated Final Safety Analysis Report. The radiological consequences for the SBLOCA [small-break loss-of-coolant accident] event presented in the Updated Final Safety Analysis Report remain unchanged. The calculated Peak Clad Temperature remains substantially below the 2200°F acceptance limit of 10[]CFR[]50.46.

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

The automatic actuation of the steam generator power operated relief valves is not an accident initiator for the SBLOCA event. The automatic actuation of the steam generator power operated relief valves currently exists at the South Texas Project and is not a new design feature. The description of the Steam Generator Power Operated Relief Valves currently exists in the Updated Final Safety Analysis Report. This change does not represent a change to the facility and does not affect the safety functions and reliability of systems, structures, or components in any new manner. Operating procedures have a temporary administrative control to ensure the automatic actuation of the Steam Generator Power Operated Relief Valves remains operable in Modes 1 and 2. This condition will become permanent with the approval of the Technical Specification Amendment proposal.

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

The proposed change results in the calculated Peak Clad Temperature remaining well below the acceptance limit of 10[]CFR[]50.46 and comparable to the results currently described in the Updated Final Safety Analysis Report.

Therefore, the South Texas Project has concluded that the proposed change does not involve a significant hazards considerations.

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

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

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036-5869

NRC Project Director: James W. Clifford, Acting

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London, Connecticut

Date of amendment request: September 2, 1997

Description of amendment request: The proposed changes to the Technical Specifications (TSs) would modify the maximum allowed containment pressure specified in TS 3.6.1.4, ''Containment Systems Internal Pressure,'' from 2.1 psig to 1.0 psig. The TS Bases, Section 3/4.6.1.4, would also be revised to reflect the new maximum allowed containment pressure.

Basis for proposed no significant hazardsconsideration 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 involve an SHC [significant hazards consideration] because the changes would not:

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

The proposed change will reduce the maximum allowed value for containment pressure specified in Technical Specification 3.6.1.4, ''Containment Systems Internal Pressure.'' This change will improve the margin between the peak containment pressure following a main steam line break (most limiting accident for peak containment pressure at Millstone Unit No. 2) and the containment design pressure limit of 54 psig. Reducing the initial containment pressure will result in a reduction in peak containment pressure.

To ensure the assumption of a lower initial containment pressure is maintained, a change to Technical Specification 3.6.1.4 is necessary.

The proposed change to Technical Specification 3.6.1.4 will allow one of the initial assumptions used in the analysis for peak containment pressure following a main steam line break to be changed. However, this change will not affect how any of the plant systems function to mitigate design basis accidents and will not require any changes to mitigation procedures. The acceptance criteria of a peak containment pressure less than the design limit of 54 psig remains the same. Therefore, this change does not significantly increase 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 proposed change does not alter the way any structure, system, or component functions and does not alter the manner in which the plant is operated. It does not introduce any new failure modes and conservatively alters an assumption made in the main steam line break safety analysis.

Therefore, the change will 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.

This proposed change will reduce the maximum allowed value for containment pressure specified in Technical Specification 3.6.1.4, ''Containment Systems Internal Pressure.'' This change will improve the margin between the peak containment pressure following a main steam line break (most limiting accident for peak containment pressure at Millstone Unit No. 2) and the containment design pressure limit 54 psig. Starting at a lower initial containment pressure will result in a lower peak containment pressure. To ensure the assumption of a lower initial containment pressure is maintained, a change to Technical Specification 3.6.1.4 is necessary.

This more restrictive change in the maximum allowed containment pressure will result in the use of a lower initial containment pressure in the analysis of a main steam line break accident. However, the analysis acceptance criteria of a peak accident containment pressure less than 54 psig, will remain the same. Therefore, there is no significant reduction in a margin of safety as defined in the Bases of Technical Specification 3.6.1.4.

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, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Deputy Director: Phillip F. McKee

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: September 2, 1997

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to: (1) Combine TS 3.6.2.1, ''Containment Spray System,'' and TS 3.6.2.2, ''Containment Air Recirculation System,'' into one specification which would reduce the allowed outage time for one inoperable containment spray (CS) train or one inoperable containment air recirculation (CAR) cooler from 30 days to 7 days; increase the allowed outage time for two inoperable CAR coolers from 48 hours to 7 days; add an allowed outage time of 48 hours (instead of entering TS 3.0.3) for one inoperable CS train and two inoperable CAR coolers or three or four inoperable CAR coolers; provide specific guidance on when to enter TS 3.0.3; and expand the applicable TS Bases to reflect these changes; (2) Modify the definition of containment integrity and TS 3.6.1.1, ''Containment Integrity,'' to indicate that the operability of the automatic isolation valve system is satisfied by the use of the containment isolation trip push buttons in Mode 4, and expand the TS Bases to reflect these change; (3) Add an exception to the reactor coolant flow rate surveillance requirement, TS 4.1.1.3, whenever there is a reduction in reactor coolant system boration while in Modes 2 and 3 because the reactor coolant pumps are required to be in operation; (4) Delete the reactor coolant system leakage surveillance requirements, TS 4.4.6.2.a and TS 4.4.6.2.b, which require monitoring the containment atmosphere particulate radioactivity and containment sump inventory, respectively; (5) Modify emergency core cooling system surveillance requirement, TS 4.5.2.e, to allow the use of alternative methods to verify that the throttle valves in Table 4.5-1 are in the correct position and expand the TS bases to address the alternative methods; (6) Modify TS 5.5.1,'' Emergency Core Cooling Systems,'' by deleting the word ''original'' since the design has been modified; and (7) Make editorial changes to terminology and item numbering.

Basis for proposed no significant hazardsconsideration 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 changes do not involve an SHC [significant hazards consideration] because the changes would not:

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

The proposed change to combine Technical Specifications 3.6.2.1 and 3.6.2.2 into one specification reduces the allowed outage time for one inoperable containment spray (CS) train or one inoperable containment air recirculation (CAR) cooler from 30 days to 7 days; increases the allowed outage time for two inoperable CAR coolers from 48 hours to 7 days; adds an allowed outage time of 48 hours (instead of entering Technical Specification 3.0.3) for one inoperable CS train and two inoperable CAR coolers, or three or four inoperable CAR coolers; and provides specific guidance when it is necessary to enter Technical Specification 3.0.3 will not affect how these systems function to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes to modify the definition of containment integrity, modify the Technical Specification 3.6.1.1, ''Containment Integrity,'' and expand the Bases to explain why automatic containment isolation valves are operable in Mode 4 have no affect on any containment isolation valve or Engineered Safety Feature Actuation System (ESFAS) component. These components will still function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to provide an exception to Surveillance Requirement 4.1.1.3 when the plant is in Modes 1 and 2 will not result in any new approach to plant operation, it simply removes the requirement to perform an unnecessary surveillance. The minimum coolant flow through the core during a reduction in Reactor Coolant System (RCS) boron concentration will still be met. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to delete Surveillance Requirements (SRs) 4.4.6.2.a and 4.4.6.2.b does not reduce the operability requirements for any equipment used to monitor RCS leakage. The equipment covered by these 2 SRs, containment atmosphere particulate radioactivity monitors and containment sump inventory monitor, provide early indication that RCS leakage exists, but do not provide the specific information (amount of leakage) necessary to verify operation within the leakage limits contained in Technical Specification 3.4.6.2, ''Reactor Coolant System Leakage.'' Operability of the containment atmosphere particulate radioactivity monitors and containment sump inventory monitor is verified by SRs 4.4.6.1.a and 4.4.6.1.b. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to Surveillance Requirement 4.5.2.e. to allow the use of alternate methods does not reduce operability or surveillance requirements for any of the Emergency Core Cooling System (ECCS) throttle valves. Therefore, these ECCS throttle valves will continue to function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to Technical Specification 5.5.1 has no affect on how the ECCS operates. The ECCS will still function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes to add information to the Bases of the affected Technical Specifications, and make editorial changes to terminology and item numbering will have no affect on equipment operation. Therefore, all associated equipment will continue to function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

Thus, this License Amendment Request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. They will not alter assumptions made in the safety analysis and licensing basis. The affected components and systems will still function as designed to mitigate design basis accidents.

Therefore, these changes do 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 proposed changes will not reduce the margin of safety since they have no impact on any safety analysis assumption. The proposed changes do not decrease the scope of equipment currently required to be operable or subject to surveillance testing, nor do the proposed changes affect any instrument setpoints or equipment safety functions. The requirement to check containment radiation and containment sump level every 12 hours has been eliminated. However, this equipment is still required to be operable, and the surveillance requirements to verify operability have not been changed. Therefore, this equipment will be available to provide early indication of RCS leakage.

The effectiveness of Technical Specifications will be maintained since the changes will not alter the operation of any component or system. In addition, the changes are consistent with the new, improved Standard Technical Specifications (STS) for Combustion Engineering plants (NUREG-1432).

Therefore, there is not 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: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Deputy Director: Phillip F. McKee

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London, Connecticut

Date of amendment request: September 3, 1997

Description of amendment request: The proposed amendment would revise the Updated Final Safety Analysis Report (UFSAR) by changing the length of time the emergency diesel generators (EDGs) would operate following a loss-of-coolant accident (LOCA) based on the capacity of the onsite diesel fuel oil supply required by the current Technical Specifications (TSs). The UFSAR indicates that the diesel fuel oil supply tanks contain a sufficient amount of fuel to operate one EDG for about 7 days and the other EDG 1 hour following a LOCA based on the TS minimum limit of 24,000 gallons of diesel fuel oil stored onsite. Northeast Nuclear Energy Company (the licensee) has performed calculations indicating that both EDGs can initially operate, following a LOCA, for 24 hours and one EDG can continue to operate for an additional 3.5 days based on the TS requirement to have a minimum of 24,000 gallons of fuel oil stored onsite. The licensee has determined that the difference in the EDGs operating time, as a result of the new calculations, constitutes an unreviewed safety question and requests approval to revise the UFSAR.

Specifically, the proposed license amendment would revise the UFSAR, Section 8.3, ''Emergency Generators,'' to reflect the operating times for the EDGs based on the TS-required onsite fuel oil supply. Additional requirements would also be added indicating that the existing nonsafety-related underground fuel oil storage tank would be required to maintain about 17,700 gallons of fuel oil when the unit is operating in Modes 1 through 4. This requirement would be included in the Technical Requirements Manual, which also will require that the amount of stored fuel oil be verified by surveillance requirements similar to the TS-required surveillances for the safety-related fuel oil supply. This change will increase the total time that one EDG can continue to operate following a LOCA from 3.5 to 7 days. The Emergency Plan (EP) procedures require that an evaluation be performed within 4 hours following a LOCA or loss of normal power (offsite power) to determine if additional fuel oil is needed from an offsite source. The licensee has a contract with a supplier for the delivery of fuel oil to the Millstone site. The EP procedures also require that load shedding recommendations be made within 24 hours. The recommendations will vary depending on the situation and are another way to extend the operating times for the EDGs.

Basis for proposed no significant hazardsconsideration 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 involve an SHC [significant hazards consideration] because the changes would not:

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

The proposed change expands FSAR Section 8.3, ''Emergency Generators,'' to discuss the length of time the emergency diesel generators (EDGs) will operate following a loss of coolant accident (LOCA) and a loss of normal power (LNP), utilizing only onsite diesel fuel oil sources. The onsite sources include the Technical Specification required volume of 12,000 gallons in each diesel oil supply tank and an additional approximate 17,700 gallons that will be maintained in the underground diesel oil storage tank. This onsite volume of diesel fuel oil is sufficient to allow two EDGs to operate at rated load (2750 KW) for 24 hours following a design basis LOCA and LNP. The remaining diesel fuel oil will be sufficient for one EDG to continue operation at rated load for a total of 7 days from event initiation.

The proposed change to the FSAR has no effect on EDG operation and reliability. The EDGs will continue to operate as designed to supply the electrical loads assumed to mitigate the design basis accidents. Therefore, there is no 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 proposed change will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. Plant operating procedures will be changed. However, the changes will not require the performance of any task not currently performed by the plant operators. Emergency Plan procedures already specify the action to provide load shedding recommendations within 24 hours of a LOCA and LNP, and to evaluate the need to order additional fuel from offsite sources within four hours after the accident.

The proposed change does not alter the way any structure, system, or component functions and does not alter the manner in which the plant is operated. It does not introduce any new failure modes and does not alter assumptions made in the safety analysis.

Therefore, the change will 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 length of time the emergency diesel generators (EDGs) will operate following a Loss of Coolant Accident and a Loss of Normal Power, utilizing only the onsite diesel fuel oil sources required by Technical Specifications has been recalculated. The new EDG run times do not agree with the current EDG run times contained in the Millstone Unit No. 2 Final Safety Analysis Report (FSAR), and therefore do not agree with the current Technical Specification Bases for 3.8.1.1, ''A.C. Sources - Operating,'' and 3.8.1.2, ''A.C. Sources - Shutdown.''

This deviation does result in a reduction in the margin of safety as defined in the Technical Specification Bases for 3.8.1.1, ''A.C. Sources - Operating,'' and 3.8.1.2, ''A.C. Sources - Shutdown.'' However, this proposed change will require additional diesel fuel oil to be maintained onsite in the non-seismic underground diesel oil storage tank. This will ensure sufficient diesel fuel oil will be maintained onsite to provide a 7 day supply, assuming a seismic event does not occur. Therefore, this is not a significant reduction in the margin of safety as defined in the Technical Specification Bases for 3.8.1.1, ''A.C. Sources - Operating,'' and 3.8.1.2, ''A.C. Sources - Shutdown.''

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, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Deputy Director: Phillip F. McKee

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: August 20, 1997

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to provide for: (1) the relocation of suppression pool volume references in Limiting Condition for Operation (LCO) 3.5.3 to the Hope Creek (HC) Updated Final Safety Analysis Report (UFSAR) and TS Bases as appropriate; (2) the revision of the suppression pool volume currently listed in LCO 3.5.3.b; (3) the relocation of the suppression pool volume references in LCO 3.6.2.1.a.1 to the UFSAR and TS Bases; and (4) the revision to the suppression pool volume reference in TS 5.2.1 to reference the TS Bases section where this information will reside.

Basis for proposed no significant hazardsconsideration 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 TS revisions involve: 1) no changes to the operation of any systems or components in normal or accident operating conditions; and 2) no significant changes to existing structures, systems or components. The installation of the new strainers will be justified separately using the provisions of 10CFR50.59. The relocation of Technical Specification references to suppression pool volume to the UFSAR and/or TS Bases will not adversely impact the safety-related functions of the suppression pool or its supported systems since any changes to suppression pool volume will be subject to 10CFR50.59 provisions. The impact of the new strainers on ECCS [emergency core cooling system] performance in Operational Conditions 4 and 5 has been determined to be negligible, with less than a 0.3% decrease in suppression pool water volume at the minimum specified suppression pool water level limit. In addition, suppression pool volume is not a parameter involved in the initiation of any accident. Therefore these changes will not significantly increase the probability of an accident previously evaluated. To the extent practicable, these proposed changes were developed consistent with the changes approved by the NRC when developing NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4'', with the intent of having the relocated information controlled in other plant documents subject to 10CFR50.59 provisions. Since the plant systems associated with these proposed changes will still be capable of: 1) meeting all applicable design basis requirements; and 2) retain the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. Therefore, these changes will not involve a significant increase in the 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.

Neither the relocation of Technical Specification references to suppression pool volume nor the revision of the suppression pool volume references for Operational Conditions 4 and 5 (COLD SHUTDOWN and REFUELING) will adversely impact the operation of any safety related component or equipment. Since the proposed changes involve: 1) no changes to the operation of any systems or components; and 2) no significant changes to existing structures, systems or components, there can be no impact on the occurrence of any accident. To the extent practicable, these proposed changes were developed consistent with the changes approved by the NRC when developing NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4'', with the intent of having the relocated information controlled in other plant documents subject to 10CFR50.59 provisions. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. Therefore, these 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.

Removal and relocation of the Technical Specification references to suppression pool volume is consistent, to the extent practicable, with the changes approved by the NRC when developing NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4''. The information retained in the Technical Specifications for minimum suppression pool water level and the information retained in the UFSAR and Technical Specification Bases will ensure that the suppression pool and supported components will remain capable of performing their intended safety functions. Any changes to suppression pool volume information retained in the UFSAR or Technical Specification Bases will be subject to the provisions of 10CFR50.59 and a separate safety evaluation would be developed to support any proposed changes that would subsequently be made. The impact of the new strainers on ECCS performance in Operational Conditions 4 and 5 has been determined to be negligible, with less than a 0.3% decrease in suppression pool water volume in the minimum specified suppression pool water level limit. By retaining the 5 inch minimum suppression pool water level limit within the TS, adequate provisions for: 1) NPSH [net-positive suction head] for ECCS pump suction; 2) recirculation volume; and 3) vortex prevention are maintained. Therefore, the changes contained in this request do not result in 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: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070

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

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

Date of amendment request: August 19, 1997

Description of amendment request: The proposed amendment would revise the Ginna Station Improved Technical Specifications (ITSs) by adding a note to the Containment Spray (CS) Limiting Condition for Operation (LCO) 3.6.6 which would allow the CS pumps in MODE 4 to be placed in pull-stop, and motor-operated valves (MOVs) 896A and 896B to have their DC control power restored with the valves placed in the closed position in order to perform interlock and valve testing of MOVs 857A, 857B, and 857C. A time limit of 2 hours is placed on this configuration for each test.

Basis for proposed no significant hazardsconsideration 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. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change is to add a note to LCO 3.6.6 which allows the CS pumps to be placed in pull-stop and MOVs 896A and 896B to have power restored and closed in MODE 4. This does not increase the probability of any accident previously evaluated since the CS system provides mitigation capability only (i.e., does not initiate any accident). In addition, there is no design basis accident previously evaluated in MODE 4 which would require the use of CS. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because the CS function is not required for any design basis accident in MODE 4. In addition, time restraints [are] placed on the proposed plant configuration. As such, no question of safety is involved, and the 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: Rochester Public Library, 115 South Avenue, Rochester, New York 14610

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005

NRC Project Director: Alexander W. Dromerick, Acting Director

TU Electric Company, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: August 2, 1996 (TXX-96434)

Brief description of amendments: The proposed changes would increase the allowed outage time (AOT) for a centrifugal charging pump from 72 hours to 7 days.

Basis for proposed no significant hazardsconsideration 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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

There is no effect on the probability of an event; the only potential effect is on the capability to mitigate the event. The centrifugal charging pumps are credited in the Final Safety Analysis Report Chapter 15 LOCA analysis for ECCS injection and for the containment sump recirculation mode for the design-basis LOCA. Increasing the AOT for the centrifugal charging pumps does not affect analysis assumptions regarding functioning of required equipment designed to mitigate the consequences of accidents. Further, the severity of postulated accidents and resulting radiological effluent releases will not be affected by the increased AOT.

A reliability analysis of the charging system found the change to have no significant impact on normal operation or on the RCP seal cooling function. Therefore, the change would not significantly increase in the probability of a seal LOCA.

The change potentially affects only the availability of the charging system for accident mitigation and has no effect on the ability of other ECCS systems to perform their functions. Through the use of a probabilistic risk assessment, it was determined that the proposed change would have an insignificant effect on the core damage frequency.

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

2. Do the proposed changes create the possibility of a new or different type of accident from any accident previously evaluated?

Unavailability of one centrifugal charging pump for a finite period of time is currently allowed by the Technical Specifications. Increasing the AOT from 72 hours to 7 days would not change the method that TU Electric operates CPSES, thus would not create a new condition. Further, the proposed change would not result in any physical alteration to any plant system, and there would not be a change in the method by which any safety related system performs its function. The ECCS would still be capable of mitigating the consequences of the design-basis accident LOCA with the one centrifugal charging pump operable. No new unanalyzed accident would be created.

3. Do the proposed changes involve a significant reduction in a margin of safety?

The proposed change does not impact either the physical protective boundaries or performance of safety systems for accident mitigation. There is no safety analysis impact since the extension of the centrifugal charging pump AOT interval will have no effect on any safety limit, protection system setpoint, or limiting condition of operation. There is no hardware change that would impact existing safety analysis acceptance criteria, therefore there is no significant change in the margin of safety.

In summary, the proposed change would not have a significant impact on the 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 Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, N.W., Washington, DC 20036

NRC Project Director: James W. Clifford, Acting

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.

Detroit Edison Company, Docket No. 50-341, Fermi 2, Monroe County, Michigan

Date of amendment request: September 5, 1997 (NRC-97-0107)

Description of amendment request: The proposed amendment would add Special Test Exception 3/4.10.7, ''Inservice Leak and Hydrostatic Testing,'' that allows the performance of pressure testing at a reactor coolant temperature up to 212 °F while remaining in Operational Condition 4. This special test exception would also require that certain Operational Condition 3 specifications for Secondary Containment Isolation, Secondary Containment Integrity, Secondary Containment Automatic Isolation Dampers, and Standby Gas Treatment System operability be met. This change would also revise the Index, Table 1.2, ''Operational Conditions,'' and the Bases to incorporate the reference to the proposed special test exception. The licensee requested that this amendment be reviewed under exigent circumstances.

Date of individual notice in the **Federal Register:** September 12, 1997 (62 FR 48113)

Expiration date of individual notice: October 14, 1997 NSHC comments: September 29, 1997

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226

NRC Project Director: John N. Hannon

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 14, 1997

Brief description of amendment request: The proposed amendments would revise the allowed tolerance of the reactor coolant system volume provided in Technical Specification 5.4.2 to account for steam generator tube plugging.

Date of individual notice in the **Federal Register:** August 26, 1997 (62 FR 45278)

Expiration date of individual notice: September 25, 1997

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

Notice Of Issuance Of Amendments ToFacility 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.

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 application for amendments: January 15, 1997, as supplemented on August 22, 1997.

Brief description of amendments: The amendments revise the minimum and maximum allowed values in Technical Specification 3.6.2.1 for suppression chamber water volume. The amendments correct an error identified by Carolina Power & Light Company in the previous calculation of water volume and correct an error in the value listed in the associated TS Bases for Unit 1 for primary system operating pressure.

Date of issuance: August 28, 1997

Effective date: August 28, 1997

Amendment Nos.: 186 and 217

Facility Operating License Nos. DPR-71 and DPR-62: Amendments change the Technical Specifications

Date of initial notice in **Federal Register:** March 26, 1997 (62 FR 14458) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 28, 1997. No significant hazards consideration comments received: No.

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

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 application for amendments: December 4, 1996

Brief description of amendments: The amendments revise the approach in Technical Specification 3/4.1.2 for determining a reactivity anomaly by changing from control rod density comparison to direct comparison of reactivity status.

Date of issuance: September 5, 1997

Effective date: September 5, 1997

Amendment Nos.: 187 and 218

Facility Operating License Nos. DPR-71 and DPR-62: Amendments change the Technical Specifications

Date of initial notice in **Federal Register:** March 12, 1997 (62 FR 11484) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at

Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: January 24, 1997

Brief description of amendments: The amendments revise the Technical Specification (TS) required surveillance calibration to be performed on the reactor water level instrumentation to reflect the modifications made to the Unit 3 instrumentation. The modifications were made during the recent Unit 3 refueling outage to improve the reliability of emergency core cooling system (ECCS) initiation on low low reactor water level. The surveillance requirement for calibration of the new level instrumentation is consistent with the ECCS low reactor water level initiation transmitter calibration requirements of NUREG 1433, ''Standard Technical Specifications, General Electric Plants, BWR/4'' for similar instrumentation. The same TS change for Unit 2 has been previously reviewed and approved by the NRC staff in Amendment No. 145 dated June 28, 1996. In addition minor editorial changes were made to the TS.

Date of issuance: September 10, 1997

Effective date: September 10, 1997, with full implementation within 60 days.

Amendment Nos.: 162 and 157

Facility Operating License Nos. DPR-19 and DPR-25: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 18, 1997 (62 FR 19143) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 10, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450

oit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan

application for amendment: December 15, 1994, as revised July 25, 1996, and supplemented December 13, 1996, and June 18, 1997

Brief description of amendment: The amendment revises Technical Specification (TS) Section 6.0, Administrative Controls, by (1) removing requirements that are adequately controlled by existing regulations other than 10 CFR 50.36 and the TS and (2) relocating selected requirements from TS Section 6.0 to licensee-controlled documents or programs.

Date of issuance: September 10, 1997

Effective date: September 10, 1997, with full implementation within 90 days. Implementation of this amendment shall include the relocation of the TS requirements to the appropriate licensee-controlled documents, as described in the licensee's application dated December 15, 1994, as revised July 25, 1996, and supplemented December 13, 1996, and June 18, 1997, and evaluated in the staff's safety evaluation dated September 10, 1997.

Amendment No.: 113

Facility Operating License No. NPF-43. Amendment revises the TS.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29873) and August 14, 1996 (61 FR 42279). The December 13, 1996, and June 18, 1997, letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 10, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161

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

Date of application for amendment: March 10, 1997

Brief description of amendment: The amendment modifies the Technical Specifications (TSs) by reducing the reactor coolant system specific activity limits in accordance with the NRC's guidance provided in Generic Letter 95-05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes by Outside Diameter Stress Corrosion Cracking.'' The definition of DOSE EQUIVALENT I-131 is replaced with the Improved Standard TS definition in the first sentence and an equation is added based on dose conversion factors derived from the International Commission on Radiation Protection (ICRP) ICRP-30. TS 3.4.8, Specific Activity, is revised by reducing the DOSE EQUIVALENT I-131 limit from 1.0 micro Ci/gram to 0.35 micro Ci/gram for the 48-hour limit and from 60 micro Ci/gram to 21 micro Ci/gram for the maximum instantaneous limit. Item 4.a in TS Table 4.4-12, Primary Coolant Specific Activity Sample and Analysis Program, TS Figure 3.4-1, and the Bases for TS 3/4.4.8 are also modified to reflect the reduced DOSE EQUIVALENT I-131 limit.

Date of issuance: September 10, 1997

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

Amendment No: 205

Facility Operating License No. DPR-66. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 7, 1997 (62 FR 24985) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 10, 1997. No significant hazards consideration comments received: No

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

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: August 4, 1997, as supplemented August 16, 1997.

Brief description of amendment: Temporary change to Technical Specification Surveillance Requirement (SR) 3.3.8.1. The change will allow the licensee to extend the frequency of SR 3.3.8.1 from 31 to 60 days.

Date of issuance: August 29, 1997

Effective date: August 29, 1997

Amendment No.: 157

Facility Operating License No. DPR-72. Amendment temporarily revises Technical Specifications Surveillance Requirement 3.3.8.1.

Date of initial notice in **Federal Register:** August 12, 1997 (62 FR 43189) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629

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: May 17, 1996, as supplemented June 14, 1996, March 17, July 29, and July 30, 1997

Brief description of amendments: The amendments modify Technical

Specification Section 3/4.4.5 Steam Generators, 3/4.4.6 Reactor Coolant System Leakage, and associated Bases to allow the installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes.

Date of issuance: September 4, 1997

Effective date: September 4, 1997

Amendment Nos.: Unit 1 - Amendment No. 90; Unit 2 - Amendment No. 77

Facility Operating License Nos. NPF-76 and NPF-80: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 29, 1996 (61 FR 25938) and April 9, 1997 (62 FR 17235). The June 14, 1996, and July 29, and July 30, 1997, submittals provided additional information that did not affect the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 4, 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: June 30, 1997

Brief description of amendment: Technical Specification Surveillance Requirements 4.7.1.5.1 and 4.7.1.5.2 require the periodic testing of the main steam isolation valves (MSIVs) to demonstrate operability. The amendment (1) clarifies when the MSIVs are partial stroked or full closure tested, (2) adds a note to the Mode 4 applicability of Technical Specification 3.7.1.5 to require that the MSIVs be closed and deactivated at less than 320 degrees F, (3) makes editorial changes, and (4) makes changes to the associated Bases sections.

Date of issuance: September 3, 1997

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

Amendment No.: 148

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

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40853) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 3, 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

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 14, 1997, as supplemented by letter dated July 30, 1997

Brief description of amendment: Technical Specification Surveillance Requirement 4.8.2.1.c.4 requires that each battery charger be tested to verify that it can supply a specified current at 125 volts. The amendment increases the required test voltage.

Date of issuance: September 5, 1997

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

Amendment No.: 149

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33130) The July 30, 1997, letter provided clarifying information that did not change the scope of the May 14, 1997, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 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

PECO Energy Company, Public Service Electric and Gas Company Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: May 9, 1997, as supplemented by letter dated July 14, 1997

Brief description of amendments: The proposed change revises the Peach Bottom Atomic Power Station, Units 2 and 3, technical specifications to extend the interval for replacing the primary containment purge and exhaust valve inflatable seals.

Date of issuance: September 4, 1997

Effective date: Both units, as of date of issuance, to be implemented within 30 days.

Amendments Nos.: 220 and 223

Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 2, 1997 (62 FR 35851) The supplemental letter provided clarifying information that did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 4, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: September 25, 1996

Brief description of amendments: These amendments (1) revise the required number of operable gaseous radioactivity monitoring system channels and particulate radioactivity monitoring system channels from one in each of the monitoring systems to one in either of the monitoring systems, (2) allow both the gaseous radioactivity monitoring system and the particulate monitoring system to be inoperable for up to 30 days provided that grab samples are obtained and analyzed at least once per 12 hours, and (3) add an action for the loss of all reactor coolant system leakage detection systems (drywell floor sump level monitoring system, gaseous radioactivity monitoring system and particulate radioactivity monitoring system).

Date of issuance: September 3, 1997

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

Amendment Nos.: 168 and 142

Facility Operating License Nos. NPF-14 and NPF-22: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 19, 1996 (61 FR 58904) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated

September 3, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701

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 application for amendments: April 25, 1997, as supplemented June 6, 1997

Brief description of amendments: The amendments revise Technical Specification 3.5.2 to eliminate reference to the flow path from the residual heat removal system to the reactor coolant system hot legs. This flow path is being eliminated to prevent excessive flow through the residual heat removal system during all hot leg recirculation configurations assuming worst-case single failures that could result in excessive flow during hot leg recirculation following a loss-of-coolant accident.

Date of issuance: September 11, 1997

Effective date: Both units, as of the date of issuance, to be implemented within 60 days of issuance.

Amendment Nos.: 200 and 184

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 14, 1997 (62 FR 26574) The June 6, 1997, supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 11, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079Sacramento Municipal Utility District, Docket No. 312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: December 9, 1993, as superseded December 19, 1995, and as supplemented on January 22, 1996.

Brief description of amendment: This amendment changes the Technical Specifications to incorporate the revised 10 CFR Part 20, Standards for Protection Against Radiation. The amendment corrects references from Semiannual Radioactive Effluent Release Report to Annual Radioactive Effluent Release Report. The amendment also corrects references from NRC Region V to NRC Region IV.

Date of issuance: August 22, 1997

Effective date: August 22, 1997

Amendment No.: 125

Facility Operating License No. NPF-1: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 2, 1994 (59 FR 10015) The information provided in the licensee's letters of December 19, 1995 and January 22, 1996 contained editorial changes and did not involve significant changes to the original Federal Register notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: : Central Library, Government Documents, 828 I Street, Sacramento, California 95814

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia Date of application for amendments: January 7, 1997, as supplemented July 2, 1997

Brief description of amendments: The amendments revise plant Technical Specifications associated with surveillance requirements testing that requires manually actuating every safety/relief valve during each unit startup from a refueling outage.

Date of issuance: September 5, 1997

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

Amendment Nos.: 208 and 150

Facility Operating License Nos. DPR-57 and NPF-5: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4350) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

Tennessee Valley Authority, Docket No. 50-260 Browns Ferry Nuclear Plant, Unit 2, Limestone County, Alabama

Date of application for amendment: June 2, 1995, revised March 3, 1997, as supplemented May 13 and August 20, 1997 (TS 353)

Brief description of amendment: The amendment provides technical specification (TS) changes for an upgrade of the power range neutron monitor instrumentation. Changes to thermal limits specifications were also proposed to implement average power range monitor and rod block monitor ts improvements, and maximum extended load line limit analyses.

Date of issuance: September 11, 1997

Effective Date: September 11, 1997

Amendment No.: 249

Facility Operating License No. DPR-52: Amendment revised the TS.

Date of initial notice in **Federal Register:** August 16, 1995 (60 FR 42609) The March 3, 1997 revision, as supplemented May 13 and August 20, 1997, does not affect the staff's proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1997. No significant hazards consideration comments received: None.

Local Public Document Room location: Athens Public library, 405 E. South Street, Athens, Alabama 35611

Tennessee Valley Authority, Docket No. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: March 27, 1997, as supplemented May 28, June 4, and July 30, 1997.

Brief description of amendment: The amendment pertains to Cycle 2 core design changes and provides operational enhancements for reactor trip setpoints. Part 1 addresses an increase in the containment sump boron concentration during a large break loss-of-coolant accident and describes changes to Technical Specification (TS) 3.5.1 and 3.5.4 regarding boron concentration. Part 2 addresses changes to TS Figure 2.1.1-1, TS Table 3.3.1-1, and TS 3.4.1 on safety limits, the trip system and pressure, temperature and flow limits, respectively.

Date of issuance: September 11, 1997

Effective date: Sepember 11, 1997

Amendment No.: 7

Facility Operating License No. NPF-90: Amendment revises the TS.

Date of initial notice in **Federal Register:** July 2, 1997 (62 FR 35852) The July 30, 1997 submittal provided clarifying information which did not affect the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1997. No significant hazards consideration comments received: None

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402

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 No. 1, Lake County, Ohio

Date of application for amendment: January 31, 1997, supplemented August 6, 1997.

Brief description of amendment: The amendment approves the use of Option B, ''Performance-Based Requirements,'' to 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.''

Date of issuance: September 9, 1997

Effective date: September 9, 1997

Amendment No.: 86

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

Date of initial notice in **Federal Register:** March 12, 1997 (62 FR 11492). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 9, 1997. No significant hazards consideration comments received: No.

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

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 No. 1, Lake County, Ohio

Date of application for amendment: May 2, 1997

Brief description of amendment: The amendment allows the leakage rate of one or more main steam lines to be up to 35 standard cubic feet per hour (scfh), as long as the total leakage rate through all four main steam lines is less than or equal to 100 scfh.

Date of issuance: September 11, 1997

Effective date: September 11, 1997

Amendment No.: 87

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33136). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1997. No significant hazards consideration comments received: No.

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

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: April 14, 1997 (TSCR 198)

Brief description of amendments: These amendments revise Technical Specification Section 15.3.1, ''Reactor Coolant System,'' to eliminate the provisions for operation of the units at below 3.5 percent rated power with a single reactor coolant pump.

Date of issuance: September 3, 1997

Effective date: September 3, 1997, with full implementation within 45 days

Amendment Nos.: 178 and 182

Facility Operating License Nos. DPR-24 and DPR-27: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27802) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 3, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: January 24, 1997, as supplemented on May 15 and August 5, 1997 (TSCR 193)

Brief description of amendments: These amendments revise TS 15.5.4, ''Fuel Storage,'' to increase fuel assembly enrichment limits to 5.0 weight percent uranium-235 while maintaining K_{eff} in the storage pools (spent fuel pool and new fuel storage racks) less than 0.95. Date of issuance: September 4, 1997

Effective date: September 4, 1997, with full implementation within 45 days

Amendment Nos.: 179 and 183

Facility Operating License Nos. DPR-24 and DPR-27: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30647) The August 5, 1997, submittal provided clarifying information within the scope of the original application and did not affect the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 4, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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 room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By October 24, 1997, 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

Arizona Public Service Company, et al., Docket No. STN 50-529, Palo Verde Nuclear Generating Station, Unit No. 2, Maricopa County, Arizona

Date of application for amendment: August 28, 1997, as supplemented by letter dated September 3, 1997.

Brief description of amendment: The amendment revises Technical Specification Table 4.3-2 to allow for a one-time, five-day extension of the required surveillance interval for the main steam isolation system portion of the engineered safety feature actuation system logic.

Date of issuance: September 4, 1997

Effective date: September 4, 1997

Amendment No.: 105

Facility Operating License No. NPF-51: The amendment revised the Technical Specifications. Press release issued requesting comments as to proposed no significant hazards consideration: Yes. September 1, 1997. Arizona Republic Newspaper (Arizona). Comments received: No. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Arizona and final determination of no significant hazards consideration are contained in a Safety Evaluation dated September 4, 1997.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: William H. Bateman

Public Service Electric & Gas Company, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey Date of application for amendment: August 19, 1997, as supplemented August 20, 1997.

Brief description of amendment: This amendment to the Technical Specifications increases the allowable band for control and shutdown rod demanded position versus indication position from plus or minus 12 steps to plus or minus 18 steps when the power level is not greater than 85% rated thermal power.

Date of issuance: September 10, 1997

Effective date: As of date of issuance, to be implemented within 7 days.

Amendment No. 183

Facility Operating License No. DPR-75: This amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes. The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on September 3, 1997, and stated that, should circumstances change during the notice period, such that a 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 notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The notice was published in the Wilmington News Journal on August 22, 1997, and in Today's Sunbeam on August 24, 1997. No public comments were received. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of New Jersey and final no significant hazards consideration determination are contained in a Safety Evaluation dated September 10, 1997.

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

Dated at Rockville, Maryland, this 17th day of September 1997.

For the Nuclear Regulatory Commission

Elinor G. Adensam,

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

[Doc. 97-25210 Filed 9–23–97; 8:45 am]

BILLING CODE 7590–01–F

NUCLEAR REGULATORY COMMISSION

[Docket No. 030–01786]

National Institutes of Health Issuance of Director's Decision Under 10 CFR § 2.206

Notice is hereby given that the Director, Office of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission (NRC), has acted on a Petition for action dated October 10, 1995, submitted by Maryann Wenli Ma, M.D., Ph.D., and Bill Wenling Zheng, M.D., Ph.D. (Dr. Ma and Dr. Zheng or Petitioners), as supplemented by letters dated March 25, 1996, and July 10, 1997, with regard to NRC Licensee, the National Institutes of Health (NIH or the Licensee).

Petitioners requested, pursuant to 10 C.F.R. 2.206, that NRC suspend or revoke the materials license of NIH, NRC License No. 19–00296–10, pending resolution of the issues raised by the Petition, and that NRC take other appropriate enforcement action, including the imposition of civil penalties against NIH for willful and reckless violations of 10 CFR part 20. Broadly stated, the Petitioners assert that, as the direct and proximate result of NIH's: (1) Deliberate failure to control and secure radioactive materials in violation of 10 CFR 20.1801 and 20.1802; (2) failure to maintain an effective bioassay program; and (3) failure to otherwise adhere to the requirements of 10 CFR part 20, Dr. Ma was contaminated with phosphorus-32 (P–32), resulting in both her and her unborn fetus receiving intakes of radioactive material significantly in excess of regulatory limits, additional NIH employees were also internally contaminated with P–32, and NIH failed to take proper actions to assess accurately the level of Dr. Ma's internal contamination or provide appropriate medical care and follow-up treatment.

In their March 25, 1996, supplemental Petition, Petitioners state that NIH's repeated denials that it has any problem with its security over radioactive materials suggests that the NIH radioactive materials license should be suspended or revoked, because the Licensee poses a threat to public health and safety, the Licensee has not responded adequately to other enforcement actions, and is unwilling or unable to comply with NRC requirements. On July 10, 1997, Petitioners submitted another supplement to their Petition, requesting immediate revocation or suspension of the NIH license on the grounds that NIH continues in its failure to implement and maintain a program to oversee licensed radioactive materials sufficiently secure to prevent another contamination incident of the type Dr. Ma experienced in 1995.

For the reasons stated in the ''Director's Decision Under 10 CFR 2.206,'' (DD–97–22) the Director of the Office of Nuclear Material Safety and Safeguards has granted the following requests of Petitioners in part: for enforcement action against NIH for violations of NRC security and control requirements and for violation of NRC requirements related to radiation safety training, ordering radioactive materials, inventory control of radioactive materials, monitoring, and the issuance, use, and collection of dosimetry.