LTR) (62 FR 39058) for a two year interim use period. The Commission also directed the staff to maintain a dialogue with the public through the use of a website and public workshops. In addition, the Nuclear Regulatory Commission (NRC) staff is developing a standard review plan (SRP) for use in reviewing licensee submittals related to the LTR.

Workshops on Guidance for Radiological Criteria for License Termination

The NRC has scheduled six workshops during the period 12/98 to 10/99. All of the workshops will be held at NRC Headquarters in the auditorium of the Two White Flint North building. The address is 11545 Rockville Pike, Rockville MD, 20852. The dates for the workshops are listed below.

Workshop Dates: December 1–2, 1998, January 21–22, 1999, March 18–19, 1999, June 16–17, 1999, August 18–19, 1999, October 20–21, 1999.

The final workshop agendas will depend on the issues that emerge as industry, NRC, and other stakeholders review, and gain experience using, the draft guidance. However, the general topics to be covered are dose modeling, demonstrating as low as is reasonably achievable (ALARA), final status surveys, and restricted use/alternate criteria. Issues of concern that emerge from industry and stakeholder review and use of the guidance will be posted and discussed on the web site, and during any additional meetings held between the workshops. The workshops will be focused on specific technical or policy issues. The agendas will be posted 6–8 weeks in advance of the scheduled date. The final agenda for the first workshop, to be held on December 1–2, 1998, is not yet finalized, but is expected to include the following topics:

1. Overview of the process to solicit stakeholder input on the draft guidance,
2. NRC test cases,
3. resuspension factor parameter in the building occupancy model,
4. measuremements when the compliance levels are close to background,
5. NRC's approach to refining the screening model for alpha emitting radionuclides,
6. licensee test cases.

The address for the web site containing the technical conference on the draft guidance for the License Termination Rule is *HTTP://TECHCONF.LLNL.GOV/INDEX.HTML.* The site contains seven major functional areas. Four separate areas have been created for discussion on the major topics in Draft Regulatory Guide DG–4006, ''Demonstrating Compliance With The Radiological Criteria For Decommissioning.'' The four areas are: 1) dose modeling, 2) final status survey, 3) ALARA, and 4) restricted use/alternate criteria. Comments, questions, and case-specific experiences can be posted in these areas by any interested party. The issues raised in these discussion areas will be considered as topics for workshops, or for one of the periodic meetings or telephone conferences. The web site will also contain an area where NRC will post draft agendas for meetings and workshops for review and comment. The final agenda, including workshop and meeting dates, times, and locations will also be posted. Finally, the site will contain a Question and Answer (Q&A) area where NRC will post the resolution to issues raised during workshops and meetings. During a public meeting held on August 14, 1998, the Q&A format was suggested by the Nuclear Energy Institute as a useful format for publishing NRC's resolution of issues.

NRC strongly encourages stakeholder participation in this process to finalize RG–4006 and develop an SRP for the license termination rule. The data and information generated during the review and implementation of the draft guidance, as well as the results of industry research and test cases, will play a significant role in the development of effective final guidance documents.

FOR FURTHER INFORMATION CONTACT: For more information, contact Mr. David N. Fauver, Sr. Health Physicist, Low-Level Waste and Decommissioning Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington DC, 20555–0001, telephone number at (301) 415–6625.

Dated at Rockville, Maryland, this 14th day of October, 1998.

For the Nuclear Regulatory Commission.

Lawrence Bell,

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

[FR Doc. 98–28191 Filed 10–20–98; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

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 September 26, 1998, through October 8, 1998. The last biweekly notice was published on October 7, 1998 (63 FR 53943).

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 Administration 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 November 20, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: September 23, 1998.

Description of amendment request: Carolina Power & Light (CP&L) proposes to revise the Harris Nuclear Plant Technical Specification (TS) 3/4.6.1.3, ''Containment Air Locks,'' to clarify the requirements for locking an air lock door shut. CP&L also proposes to revise TS 3/4.6.1.3 to be consistent with NUREG 1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995.

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

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

Containment Air Locks are not an accident initiating system as described in the Final Safety Analysis Report [FSAR]. The proposed change implements guidance for Technical Specifications associated with air lock doors consistent with NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995. Additionally, clarification is provided to permit locking an inoperable air lock door as required by Technical Specifications [TS]. The proposed change does not affect another Structure, System, or Component. The operation and design of containment air locks will not be affected by this proposed change. The ability of containment to mitigate an accident will not be affected by this change.

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

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

Containment Air Locks are designed to form part of the containment pressure boundary. The proposed change provides for administrative controls and operating restrictions for air lock doors consistent with guidance provided by the Commission. Containment Air Locks are not an accident initiating system as described in the Final Safety Analysis Report. The proposed change does not affect another Structure, System, or Component. The operation and design of containment air locks will not be affected by this proposed change.

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

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

The proposed change to containment air locks does not affect any of the parameters that relate to the margin of safety as described in the Bases of the TS or the FSAR. Accordingly, NRC Acceptance Limits are not affected by this change.

Therefore, the proposed change does 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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

NRC Project Director: Pao-Tsin Kuo (Acting).

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1, Monroe County, Michigan

Date of amendment request: July 17, 1998 (Reference NRC–98–0044).

Description of amendment request: The proposed amendment will revise the License to allow the licensee to possess special nuclear material in a quantity totaling no more than 15 grams of uranium-235, uranium-233, or plutonium, or any combination thereof and with plutonium totaling no more than 2 curies.

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 using the standards in 10 CFR 50.92(c). The licensee's analysis is presented below:

(1) *Does the proposed change significantly increase the probability or consequences of an accident previously evaluated?*

The proposed changes do not involve a significant increase in the probability or consequences of an accident. Possessing trace amounts of special nuclear material cannot affect the probability of the analyzed sodium or liquid waste accidents. The ability to possess such material does not itself change any methods of handling liquid waste or sodium. Possession of special nuclear material could potentially increase the consequences of an accident if it was in use or in the vicinity if an accident occurs. However, the increase in consequences would not be significant due to the limitations on radioactivity content of such special nuclear material. The special nuclear material limit is below that requiring an emergency plan or maximum dose evaluation per 10 CFR 70.22(i). Since the quantity is below that requiring an offsite emergency plan or evaluation, even if all the special nuclear material allowed to be possessed by the proposed amendment were released during a postulated accident, the consequences would not be significantly increased. If the provision allowing for possession of more than 15 grams of special nuclear material or 2 curies of plutonium were to be used in the future due to identified plant contamination, the requirements of 10 CFR 70.22(i) would need to be assessed and a dose evaluation performed or an emergency plan submitted if required to ensure the analyzed accident is appropriately addressed and mitigated. Any such special nuclear material would be contained in the remaining plant contamination, since fuel and blanket material were shipped offsite during 1973–1975. Therefore, this amendment does not involve a significant increase in the probability or consequences of an accident.

(2) *Will the proposed amendment create the possibility of a new or different kind of accident from any accident previously analyzed?*

The proposed changes do not create the possibility of a new or different type of accident from any previously evaluated. Allowing possession of small amounts of special nuclear material does not change methods of monitoring the facility or operations or surveillance of any systems at Fermi 1. The amount requested is below that requiring criticality monitoring per 10 CFR 70.24, and the separation of the special nuclear material will not be permitted. Thus, there is no identified physical mechanism for creating an accident based on the existence of such material in the quantities specified. If the provision allowing for possession of more than 15 grams of special nuclear material or 2 curies of plutonium if is identified in plant contamination in the future were to be invoked, applicable provisions to ensure public safety per 10 CFR Part 70, Part 73, and Part 74 will apply. For these reasons, allowing Detroit Edison to possess very limited amounts of special nuclear material at Fermi 1 will not create the possibility of a new or different type of accident.

(3) *Will the proposed change significantly reduce the margin of safety at the facility?*

The proposed changes do not involve a significant reduction in the margin of safety at Fermi 1. No changes to any systems, or the status of any systems or structures, are created by this amendment. Being able to have a very limited amount of special nuclear material at Fermi 1 will not significantly reduce the margin of safety because a 10 CFR Part 20 program is already in place, and the amount of special nuclear material is being limited below criteria requiring an emergency plan, special nuclear material control program, or criticality monitoring. If more than 15 grams of special nuclear material or 2 curies of plutonium is identified in plant contamination in the future, the proposed license amendment will require the applicable portions of 10 CFR Part 70, Part 73, and Part 74 to apply for the amount identified. For these reasons, this amendment will not significantly reduce the margin of safety at Fermi 1.

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, NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Branch Chief: John W.N. Hickey.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: April 30, 1998.

Description of amendment request: Arkansas Nuclear One—Unit 2 (ANO–2) Technical Specification (TS) 4.8.1.1.2.c.3 has been revised to relocate the specific value for the single largest post-accident load to the Bases associated with TS 4.8. The revised TS 4.8.1.1.2.c.3 would require the licensee to verify the generator capability to reject a load greater than or equal to its associated single largest post-accident load.

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

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

The Diesel Generators (DGs) are not identified as the initiator of any accident previously analyzed. The design and function of the DGs are unaffected by this proposed change. Applying more restrictive acceptance criterion to the single largest load rejection test can not result in an increase in the probability of accidents previously evaluated and will provide increased assurance that the DGs will perform as intended to support the mitigation of accidents previously evaluated.

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

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

The proposed change corrects information contained in the technical specification and does not involve any design change, plant modification, change in analyzed DG performance, or change in plant operation. Since the DGs are not considered to be event initiators, their accident mitigation function is unaffected, and normal operation is unaffected, the proposed change does not result in new or different accidents from those previously analyzed.

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

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

The design and function of the DGs are unaffected by the proposed change. Applying more restrictive acceptance criterion to the single largest load rejection test will provide increased assurance that the DGs will perform as intended to support the mitigation of postulated accidents. DG performance is proposed to meet a more stringent standard.

Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: May 18, 1998.

Description of amendment request: The proposed changes delete the ANO–2 TS 3.6.2.2 and 4.6.2.2 requirements, and their associated bases, for the sodium hydroxide addition system and add new limiting conditions for operation, action statements, surveillance requirements, and bases information for trisodium phosphate baskets which will be installed during the next ANO–2 refueling outage (2R13). The capability to add sodium hydroxide to the containment spray system during the initial phase of a loss-of-coolant accident will be replaced with crystalline trisodium phosphate (TSP) dodecahydrate stored in containers located on the floor of the containment building.

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

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

The proposed change modifies the method of containment spray sump pH control. The containment spray function is important for containment heat removal/pressure mitigation. However, this change does not affect the probability of occurrence of the accident initiators which result in the need for containment heat removal and pressure mitigation. Since the TSP baskets are seismically mounted passive devices located inside the containment, they cannot initiate a transient or affect the probability of occurrence of any previously analyzed accident.

The proposed change only modifies the chemical composition of the containment spray and sump fluid. The proposed changes do not affect the heat removal/pressure mitigation functions of the system since the spray flow rate and droplet size are unchanged. The proposed change also will not adversely affect the radiological doses for the design basis accident (DBA) loss-of-coolant accident (LOCA) at the exclusion area boundary, low population zone, control room, or emergency response facility. The change does not adversely affect the calculated peak clad temperature for the DBA LOCA or the environmental qualification (EQ) of components located inside containment.

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

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

The proposed change allows the use of TSP as a buffering agent for the containment sump instead of sodium hydroxide (NaOH) added via the containment spray system. The TSP baskets are passive devices that have minimal impact on any other system except through water chemistry. The change in water chemistry does not adversely affect any safety system or required safety functions. The replacement of NaOH additive with TSP will not change the probability of a malfunction of safety-related equipment.

Potential malfunctions relating to the proposed modification have been evaluated for their effect on plant safety and have been found to be non-significant. Additionally, the transient pH behavior of the containment spray flow does not adversely affect the EQ of components located inside containment.

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

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

The proposed change does not adversely affect the ability of the containment spray system to perform the functions of containment heat removal, pressure mitigation, and fission product (iodine) retention. The proposed change does not adversely affect any equipment credited in the safety analysis. Also, the proposed change does not increase the peak clad temperature or the offsite doses due to the DBA LOCA.

Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: June 29, 1998.

Description of amendment request: The proposed amendment would revise the as-found lift setting tolerance for the ANO–2 main steam safety valves (MSSVs) and pressurizer safety valves (PSVs) will be increased. The proposed increase in the lift setting tolerance is contingent upon a reduction in a linear power level-high setpoint and use of the latest small break loss of coolant accident (SBLOCA) methodology for development of the Core Operating Limits Report (COLR).

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

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

This change allows for a larger ±3% tolerance versus ±1%, −3% as-found lift setting tolerance. The proposed change does not involve any change to the physical characteristics of the main steam safety valves (MSSVs) and pressurizer safety valves (PSVs), and will have no impact on the as-left settings. During testing, the MSSVs and PSVs will continue to adjusted to ±1% of the Technical Specification (TS) lift setting.

The impact on the Safety Analysis Report (SAR) analyses when the as-found lift setting tolerances are increased has been evaluated and the effects upon the impacted events have been found to be within acceptable limits, providing the allowable linear power level with three inoperable MSSVs is revised from 45% to 36%, and that the latest NRC approved C-E small break loss of coolant analysis (LOCA) evaluation model, CENPD–137, Supplement 2–P–A, is included as a methodology for determination of operating parameters identified within the core operating limits report (COLR). With these concurrent changes, plant systems required for safe operation and shutdown will continue to be available to fulfill their safety function as described in the SAR. Steam production in excess of relief capacity is precluded by the physical design of the plant and operation of the reactor protection system. Revision of the MSSV as-found lift setting tolerance from ±1%, ±3% to ±3% does not alter safety analyses conclusions.

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

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

This change does not create any new plant configuration or operational mode. This proposal to increase the MSSV and PSV as-found lift setting tolerance does not modify equipment or change the manner in which the MSSVs and PSVs will be operated. ASME design requirements for maintaining system operating pressure limits below the maximum design pressure of 1210 psia for plant secondary systems, and 2750 psia for the reactor coolant system (RCS) are not impacted. The reduction in allowable linear power level when three MSSVs are inoperable assures plant operation within current analysis assumptions. The addition of topical report CENPD–137, Supplement 2–P–A, as a reference to develop the COLR is bounded by assumptions within the existing safety analysis. The cycle specific COLR analyses will continue to be performed utilizing NRC approved methodologies. The TS changes do not require any new equipment be included in the design basis, and current equipment will continue to be operated in a manner consistent with its design.

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

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

The upper tolerance limit for design pressure is not affected by this change. During the most severe anticipated operational transient, the Secondary System pressure and RCS pressure will not exceed 110% of design pressure. The MSSV and PSV lift settings will continue to be set within −1% of the TS lift setting during surveillance testing.

The decrease in the peak cladding temperature of the reactor fuel, due to a change in the methodology for analysis, does not significantly impact previous analytical results. The current and previous analytical methodologies are approved by the Staff.

The impact of the proposed changes on the ANO–2 SAR analyses have been evaluated. The evaluation demonstrates that the results of the impacted events remained within the acceptable limits providing the maximum linear power level percentage for three inoperable MSSVs is reduced. This reduction in maximum allowable linear power level assures that adequate steam relief capacity will be available to prevent overpressurizing the secondary steam system during the most severe anticipated operational transient.

Addition of topical report CENPD–137, Supplement 2-P-A, will not reduce the existing TS operability and surveillance requirements. The cycle specific COLR limits for future reloads will continue to be developed based on NRC-approved methodologies. The ANO–2 TSs will continue to require that the core be operated within these limits.

The cumulative impact of all of the proposed changes and the results of the impacted events have been found to be within acceptable limits. The system capabilities to mitigate and/or prevent accidents will be the same as they were prior to these changes.

Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: June 29, 1998.

Description of amendment request: These proposed changes are in Technical Specification 3.4.2, ''Reactor Coolant System—Safety Valves—Shutdown,'' and Technical Specification 3.4.12, ''Reactor Coolant System—Overpressure Protection'' regarding the low temperature overpressure protection system. The specific changes include modifying the requirements for the pressurizer code safety valve requirements specified by Technical Specification 3.4.2 and a modification of the safety injection tank isolation requirements specified in Technical Specification 3.4.12.

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

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

The reactor coolant system (RCS) is designed with overpressure protection devices to be used in all modes of operation. The changes to Technical specification (TS) 3.4.2 will ensure that, if no pressurizer code safety valves are operable, the RCS will be cooled down to the mode of applicability of the low temperature overpressure protection (LTOP) system (TS 3.4.12) within 12 hours. The LTOP relief valves provide sufficient relief capacity to protect the RCS from overpressurization when the RCS inlet temperature (T_c) less than or equal to 220° F. Therefore, this change will ensure the proper actions will be taken that will ensure adequate overpressure protection of the RCS. These actions are not accident initiators, and therefore do not involve a significant increase in the probability of any accident previously evaluated.

The proposed change to TS 3.4.12 provides additional operational flexibility for the use of the safety injection tanks (SITs) as an additional inventory source during Modes 4, 5, and 6 when the RCS is in LTOP conditions. The ability to use the SITs, with a pressure less than 300 psig is within the existing LTOP analysis. The LTOP analysis ensures that under the analyzed worst case overpressurization event, the RCS is protected. The 300 psig SIT pressure limit, corrected for instrument uncertainty, will prevent a challenge to the LTOP relief valves and therefore the RCS will be assured of overpressure protection. The SIT pressure limit will also be low enough to prevent an inadvertent isolation of the shutdown cooling system and thus prevent a loss of shutdown cooling due to placing an SIT in service. The remaining changes included in this amendment request are considered administrative in nature and are therefore considered acceptable.

Based on the above discussions, these changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes included in this amendment request provide additional operational flexibility for the use of the SITs and specify the proper actions to be taken that will ensure adequate overpressure protection of the RCS. The LTOP relief valves have already been evaluated for operation below 220° F. The changes do not introduce any new plant configurations. No new accident possibilities are being introduced by these changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to the TS 3.4.2 action statement requires the T_c be less than or equal to 220° F when no pressurizer code safety valves are available. When Tc is less than or equal to 220° F, the LTOP system operability is required by TS 3.4.12. This action will provide assurance that the RCS will be protected from an overpressurization event and therefore increases the margin of safety.

The requirements to maintain one pressurizer code safety valve in Mode 4 when T_c is less than or equal to 220° F and in Mode 5 has been removed by the proposed revision to TS 3.4.2. The LTOPs provide adequate RCS over pressure protection during these modes without reliance on the pressurizer code safeties. Maintaining the requirement to require one pressurizer code safety to be operable at the same time as the LTOP system is required to be operable, provides no additional plant safety. An operable LTOP system prevents RCS pressure from increasing high enough to challenge the pressurizer code safety lift setpoints.

The current TS 3.4.12 LTOP limits are based on an analysis that uses the methodology outlined in the ASME Code Case N–514. This code case defines the margin of safety for the current LTOP limits. This code case was utilized in the development of TS 3.4.12. The safety factor utilized by the code case provides a reasonable vessel overpressure allowance for conditions expected during a low temperature transient. The margin of safety is not reduced with SITs in service and pressurized to less than 300 psig because this condition is bounded by the existing LTOP analysis. Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: June 29, 1998.

Description of amendment request: The proposed change to the Arkansas Nuclear One Unit 2 Technical Specifications would provide a range of acceptable values for the 4160 Volt bus loss of voltage values. The present Technical Specification Table 3.3–4, item 7.a provides a single value for both the trip and the allowable values for the 4160 Volt bus loss of voltage requirements. These table entries do not include an acceptable range or an explicit indication of the allowed tolerance that the actual setting is allowed to vary from the indicated value. The proposed change replaces the specific trip value with an explicit range of acceptable allowable values.

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

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

The two 4160 Volt (V) vital bus loss of voltage protection relays that are provided on each of the 4160 V safety buses are provided to detect loss of voltage, isolate the safety buses, initiate load shedding, and start the associated emergency diesel generator. This safety function is unchanged by the proposed setpoint revisions. The revised settings for the loss of voltage protection relays will continue to provide the safety function with no appreciable additional time delay. The proposed time delays are within those assumed in the ANO–2 safety analyses. Additionally, the lower voltage settings will prevent unnecessary isolations from the off-site power sources which will contribute to reducing the probability of a loss of off-site power due to off-site power system transients.

The ANO–2 technical specifications will continue to require the 4160 V loss of voltage functions to be surveillance tested at their present frequency without changing the modes in which the surveillance is required or the modes of applicability for these components. The technical specifications will continue to require the same actions as currently exist for the inoperability of one or more of the 4160 V loss of voltage channels. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change introduces no new modes of plant operation or new plant configuration. The 4160 V vital bus loss of voltage protection relays are required to operate following a complete loss of off-site power to initiate the bus power source transfer to on-site power, i.e., the emergency diesel generators, to prevent a loss of all AC power. This safety function is unchanged by the proposed setpoint revisions, and the proposed setpoints continue to provide the required actions consistent with the ANO–2 safety analysis. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The two undervoltage relays located on each 4160 V safety bus are provided to detect loss of voltage, isolate the safety buses, initiate load shedding, and start the emergency diesel generators. This safety function is unchanged by the proposed setpoint revisions.

The lower loss of voltage values do not affect the safety function since there is no appreciable time difference in reaching the lower setpoints during a loss of voltage event. The maximum proposed time delay setting with the minimum loss of voltage relay setting is within those used in the ANO–2 safety analysis. The revised settings for the relays will continue to provide the safety function with no appreciable additional time delay.

Removal of the trip value from the technical specifications is consistent with that which is presented in NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' The current ANO–2 technical specifications and NUREG–1432 both indicate that if the setpoint is outside the allowable value column, the associated channel is declared inoperable. This approach is consistent with this proposed technical specification change.

The trip and allowable values listed in the technical specifications for the loss of voltage protection for the 4160 V buses are presently the same. With these values being the same, if the trip value is exceeded, the allowable value will also be exceeded. This change provides a range of acceptable allowable values for these relays. By relocating the trip values in the surveillance test procedures, the procedural limits for the voltage and time delay settings can be adjusted to ensure margin to the allowable values. Additionally, the lower voltage settings will help to prevent unnecessary isolation from the off-site power sources due to off-site perturbations in the electrical grid, and thus contribute to increasing the margin of safety. Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: June 29, 1998.

Description of amendment request: The proposed Technical Specification change revises the surveillance testing requirements for the Arkansas Nuclear One—Unit 2 (ANO–2) direct current (DC) electrical distribution system. ANO–2 is planning on modifying the 120 volt vital alternating current (AC) electrical distribution system by installing new inverters during the next scheduled refueling outage (2R13). This modification will increase the normal 125 volt vital DC system loads by adding the inverters as a normal load. The power for each 125 volt vital DC system is normally supplied by its associated battery charger. ANO–2 is in the process of replacing the vital DC battery chargers by plant modification to ensure all the battery chargers are of sufficient capacity to provide the necessary current requirements for the normal 125 volt vital DC loads. The proposed change to specification 4.8.2.3.c.4 is required to ensure the new chargers are adequately tested to support the associated inverter replacement.

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

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

Technical Specification (TS) surveillance requirement (SR) 4.8.2.3.b.2 requires the battery banks for each of the vital 125 volt direct current (DC) systems to be inspected to ensure that no visible corrosion exists at the terminals or the connectors. This SR has been modified to allow the present corrosion inspection, or the measurement of the resistance of the associated battery connections. The resistance measurement provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance and has been an accepted alternative to the visual inspection requirement.

The Bases change associated with TS 3.8.2.3 Action ''b'' is considered administrative in nature and simply clarifies the intent of the action without changing the requirements of the action or its required completion time. The station batteries are not classified as accident initiators in the ANO–2 accident analysis. The 125 volt class 1E batteries are credited for accident mitigation in the accident analysis. The above described changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Each battery charger is required to have sufficient capacity to restore the battery from the design minimum charge to its fully charged state while supplying normal steady state loads. The minimum specified TS surveillance required charger amperage limit will ensure this capacity. The additional charger output is presently accounted for in the emergency diesel generator loading tables in the Safety Analysis Report (SAR). Loss of one train of the vital 125 volt DC system is an accident that has been evaluated in the SAR. The capacity of the battery chargers is not a factor in the probability of this accident occurring. Therefore, the changes associated with this technical specification amendment request do not increase the probability of any accident previously evaluated.

The proposed technical specification changes do not modify the limiting condition for operation or the associated action statements regarding operability of the battery chargers other than clarifying these requirements. The frequency at which the battery charger operability is demonstrated by surveillance testing is not being modified by this technical specification change request. The proposed battery charger surveillance testing acceptance criterion will more appropriately demonstrate the capability of this equipment. This change does not affect the consequences of any of the previously evaluated accidents.

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

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

Technical specification SR 4.8.2.3.b.2 requires the battery banks for each of the 125 volt systems to be inspected to ensure that no visible corrosion exists at the terminals or the connectors. This SR has been modified to allow the present corrosion inspection, or to perform resistance readings on the associated battery connections. The visual inspection is required to detect corrosion of the battery connections. The resistance measurement of the associated battery connections provides an acceptable alternative to the visual inspection requirement and provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The availability of an extra battery charger for each train following the plant modification provides a more reliable configuration without introduction of any new modes of plant operation. No new accident possibilities are being introduced by the proposed change to the surveillance testing specification for battery charger amperage. Increasing the surveillance testing amperage limit for the battery chargers does not create the potential for any different accident since the new value remains within the design capacity of the components.

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

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

TS SR 4.8.2.3.b.2 has been modified to allow resistance readings on the associated battery connections or the performance of the present visual inspection requirements. The resistance measurement of the associated battery connections provides an acceptable alternative to the visual inspection requirement and provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance without a significant reduction in the margin of safety.

The proposed technical specification surveillance requirements for the battery chargers continues to require testing of battery chargers at the present duration and frequency. These requirements will also apply to the second charger being installed for each Class 1E battery train. Each of the new battery chargers has sufficient capacity to restore the battery from the design minimum charge to its fully charged state while supplying normal steady state loads. The proposed surveillance specification change does not involve a significant reduction in the margin to safety since the demonstrated capacity will be of a higher amperage requirement than is demonstrated during the surveillance test with the existing configuration. Increasing the required amperage value assures the surveillance test will continue to demonstrate the chargers can provide significantly more current than is necessary to meet the design requirements. Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: August 6, 1998.

Description of amendment request: The proposed technical specification change revises the Action requirements for the Arkansas Nuclear One—Unit 2 (ANO–2) Control Element Assembly (CEA) position indicator channels. The Action requirements listed in Specification 3.1.3.2 are being modified consistent with the requirements of NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' The proposed changes also include the relocation of Technical Specification Table 3.8–1, ''Containment Penetration Conductor Overcurrent Protective Devices'' per

NRC Generic Letter 91–08, ''Removal of Component Lists From Technical Specifications.''

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

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

This technical specification (TS) change request contains the relocation of Table 3.8–1, Containment Penetration Conductor Overcurrent Protective Devices, and changes to the control element assembly (CEA) position indication.

Generic Letter (GL) 91–08, ''Removal of Component Lists From Technical Specifications,'' was issued as a TS line item improvement by the NRC. Table 3.8–1 is one of the specific lists of components contained in the GL. TS Table 3.8–1 and all its references have been removed from Specification 3/4.8.2.5 in accordance with the GL. This change is considered administrative in nature because the requirements for operability, the limiting conditions for operation, the surveillance requirements and their frequencies for the containment penetration conductor overcurrent protective devices remains the same. This amendment request fundamentally modifies the physical location of the devices listed in Table 3.8–1 from the TS to the plant procedures. These changes have no affect on the probability or consequences of any accident previously evaluated.

The remaining changes included in this amendment request are those relating to the CEA position indication. The Action requirements for TS 3.1.3.2 were modified to be consistent with the requirements of NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' The most recent revision of NUREG–1432 was used to produce this change because it represents the latest guidance for the TS CEA position indication requirements that are applicable to ANO–2 and acceptable to the NRC.

The requirement was removed from TS 3.1.3.2 that restricted each CEA group to a maximum of one CEA with less than two of the required position indicator channels. NUREG–1432 places no requirements on the number of CEAs in a group with less than two of the required position indicator channels. NUREG–1432 would allow all the CEAs in a group to have only one of the required CEA position indications operable. In this situation, the associated CEAs with less than two of the required position indicator channels would have to be placed at their ''Full In'' or ''Full Out'' limits.

TS 3.1.3.2 was modified to allow the use of the ''Full In'' or ''Full Out'' limits which ensures this specification is consistent with its bases and NUREG–1432. The TS will still maintain the requirements for two independent means of determining CEA position with this amendment request. With two independent means of determining CEA position, reliable determination of actual CEA position will be maintained.

Additionally, NUREG–1432 does not require the placement of any other CEAs in the associated group at the ''Full Out'' limit when one of the CEAs in the group has only one of the required position indication systems operable. All of the remaining CEAs in the associated group still have at least two independent means of CEA position indication or they would already be required to be positioned to the ''Full Out'' limit to restore the second position indication. The TS retains the requirements for the individual and group CEA alignment in accordance with Specifications 3.1.3.1 and 3.1.3.6. These requirements also eliminate the need for pulling the remaining CEAs in the group to the ''Full Out'' limit as long as the alignment requirements are maintained.

These changes will allow the operator more time to focus on the individual CEA position indication problem rather than moving the remainder of the CEAs in the group unnecessarily. Anytime that a CEA is moved, a small probability exists for it to slip or drop into the core. If this were to occur while attempting to align the group to the ''Full Out'' limit, a reactor transient would be initiated. Additionally, anytime the CEAs are operated, a small probability of an error exists. Removing the unnecessary requirement for the group withdrawal could decrease the probability of CEA misoperation. CEA position indication is not considered as an accident initiator. Retaining the requirements to maintain at least two independent means of determining CEA position will ensure the consequences of all the accidents previously evaluated remain unchanged.

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

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

The portions of this change that are made in accordance with GL 91–08 are considered administrative in nature and do not result in the creation of a new or different kind of accident from any previously evaluated.

The bases for TS 3.1.3.2 state that the action statements applicable to inoperable CEA position indicators permit continued operation when the positions of CEAs with inoperable position indicators can be verified by the ''Full In'' or ''Full Out'' limits. Although TS 3.1.3.2 may have originally been intended to allow continued operation using the ''Full In'' limits, it has never been clearly addressed in the specification. NUREG–1432 allows the use of both the ''Full In'' or ''Full Out'' limits. This amendment request will not change the methods for CEA operation, although it will reduce unnecessary CEA manipulations due to CEA position indication problems.

The requirements of Specification 3.1.3.1 will ensure that an individual CEA is maintained in proper alignment with the remaining CEAs in the group. Specification 3.1.3.6 will ensure the CEA groups are maintained within the proper withdrawal sequence and insertion limits. Specification 3.1.3.5 will ensure the shutdown CEA groups are maintained in the ''Full Out'' position. The CEA position indication changes allowed by this amendment request, including the allowance to use the ''Full In'' limits, can produce a CEA configuration that is different from that allowed by the current TSs. However, the allowed configurations will be bounded by the TS 3.1.3.2 Action ''c'' requirements for compliance with Specifications 3.1.3.1, 3.1.3.5, and 3.1.3.6. Therefore, the action requirements of TS 3.1.3.2 will ensure the CEAs are operated consistent with the safety analysis assumptions.

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

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

The portions of this change that are made in accordance with GL 91–08 are considered administrative in nature and have no effect on the margin of safety. The remaining changes can result in a lower probability of CEA misoperation and reduce the potential of plant transients due to CEAs that slip or drop into the core while performing unnecessary group realignments. These changes can also reduce unnecessary plant shutdowns, due to unneeded restrictions on CEA position indication. An unnecessary plant shutdown produces an opportunity for plant upsets that can be avoided by this change. The proposed TS provide an equivalent level of safety as those specifications that currently exist. Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: September 17, 1998.

Description of amendment request: The proposed amendment addresses a problem associated with the existing technical specifications being inconsistent with the design of the plant protection system (PPS). The PPS uses a design in which a single bistable is used to automatically enable the selected core protection calculator (CPC) trip functions whenever a permissive exists to bypass the high logarithmic power level trip function. The technical specifications allow the bypass of the high logarithmic power trip when power is above 10^{-4} percent power and allow bypasses of the affected CPC trips when power is below 10^{-4} percent power. The proposed technical specification change establishes a range for the bistable setpoint to be within such that it is possible to meet both of its design functions while also meeting the technical specification requirements.

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

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

This technical specification (TS) change request modifies the power level at which two of the three operating bypasses can be set to operate. This change is necessary because the present plant bistable design requires a range for this bistable to operate within rather than a specific setpoint as required by the present TS. The single bistable associated with these operating bypasses is designed with an inherent hysteresis loop and therefore requires an operating range. The band of 10^{-4}% to 10^{-2}% of rated thermal power provides the bistable an adequate operating range to account for the inherent bistable hysteresis, allow for bistable drift, and provides margin for the applicable uncertainties. Regardless of the actual bistable setpoint within this band, the bistable design ensures that either the high logarithmic power level or the core protection calculator (CPC) generated trips are available to provide reactor trip protection. The CPC and logarithmic power operating bypasses and their setpoints are not considered credible accident initiators and therefore modifying their setpoints does not involve a significant increase in the probability of an accident previously evaluated.

The automatic removal function of these operating bypasses is designed to mitigate the consequences of accidents. As described within the background section of the TS change request, the safety analyses associated with operating bypasses have been reviewed for the acceptability of these changes. This review concluded that these changes are considered bounded by the existing safety analyses. Since these TS changes are bounded within the present safety analyses, they do not involve a significant increase in the consequences of an accident previously evaluated.

The remaining changes included in this TS change request are being made to clarify the existing requirements for the operating bypasses and to establish consistency with the above described changes. The remaining changes have been found acceptable because they are considered administrative in nature and have no effect on the probability or consequences of an accident previously evaluated.

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

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

There are no physical plant modifications being made to the plant as a result of this change. The only function that is required by the TS and modified by this change is associated with the allowed setpoint for the automatic bypass removal feature of the CPCs. This feature will still be required by the TS, but will be allowed a slightly higher setpoint. The system connections and the reactor trip setpoints are not affected by this change. The CPC and logarithmic power operating bypasses and their setpoints are not considered as credible accident initiators. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The safety analyses associated with these operating bypasses have been reviewed for the acceptability of these changes. This review concluded that the changes associated with this TS change request are considered bounded within the existing safety analyses. The associated safety analyses have been considered to be acceptable because they have produced acceptable results and thus provide an acceptable margin to safety. Therefore, this change does 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

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

Date of amendment request: June 29, 1998.

Description of amendment request: The proposed changes modify Technical Specification (TS) 3.7.6.1 (Control Room Emergency Air Filtration System—Modes 1–4), TS 3.7.6.2 (Control Room Emergency Air Filtration System—Modes 5 and 6), TS 3.7.6.3 (Control Room Air Temperature—Modes 1–4), TS 3.7.6.4 (Control Room Air Temperature—Modes 5 & 6), and TS 3.7.6.5 (Control Room Isolation and Pressurization), and the associated Bases.

The proposed changes to the control room ventilation TS affects the Applicability and the Actions. These changes will make the TS consistent with NUREG–1432 (Standard Technical Specifications Combustion Engineering

Plants), as applicable, and the accident analysis. The proposed changes to the TS Bases make the Bases consistent with the TS and also clarify that suspending movement of irradiated fuel assemblies shall not preclude movement to a safe conservative position.

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 operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes revise the control room ventilation Technical Specifications (TS) Actions to delete the Action statement to suspend all operations involving positive reactivity changes, and adds an Applicability and Action related to the movement of irradiated fuel assemblies. The changes also add an Applicability footnote and revise the Bases to allow irradiated fuel assemblies to be placed in a safe conservative position when movement is required to be suspended. Other changes to the Bases are being made to be consistent with the TS. These changes do not affect the probability of an accident. The control room ventilation systems (ventilation, temperature, or envelope) do not affect the initiators of an accident; therefore, the changes do not alter the initiators of any analyzed events.

The administrative and more restrictive changes do not affect the consequences of an accident. The administrative changes add an Applicability footnote and revise the TS Bases to make them consistent with the TS. This will ensure the applicable control room ventilation system TS are entered during movement of irradiated fuel assemblies and that there is no confusion associated with the Bases being inconsistent. The more restrictive change of adding the Applicability during movement of irradiated fuel assemblies and the Action to suspend movement of irradiated fuel assemblies eliminates the precursor to the fuel handling accident which prevents the fuel handling accident from occurring when the control room ventilation systems are inoperable. The addition of this Action ensures the event that may release radioactivity is precluded when the control room ventilation systems are inoperable.

The less restrictive changes (deleting the requirement to suspend positive reactivity changes and a Bases change which allows irradiated fuel assemblies to be placed in a safe conservative position when movement has been suspended) do not affect the consequences of an accident because no accident mitigator is affected. The safety analysis credits instrumentation to detect a boron dilution accident and alert the control room staff. After the control room staff is alerted, the accident is terminated without a radioactive consequence. These instruments are required to be Operable and if one is inoperable, positive reactivity changes are required to be suspended. If both instruments become inoperable, along with suspension of positive reactivity additions, boron concentration is required to be determined at frequencies specified in the Core Operating Limits Report (only when source range neutron flux monitors are inoperable). Also, the shutdown margin (SDM) is required to be met. If the SDM requirements are not met, action must be taken to borate (addition of negative reactivity) until the SDM is restored. Therefore, if the control room ventilation systems are inoperable, suspension of positive reactivity changes are not required. The added statement in the Bases allows irradiated fuel assemblies to be placed in a safe conservative position to preclude a fuel handling accident from occurring. These Actions ensure that appropriate measures are taken to preclude events that would require the control room to be isolated when any of the control room ventilation systems are inoperable.

Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

The proposed changes revise the control room ventilation TS Actions to delete the Action statement to suspend all operations involving positive reactivity changes, and adds an Applicability and Action related to the movement of irradiated fuel assemblies. The changes also add an Applicability footnote and revise the Bases to allow irradiated fuel assemblies to be placed in a safe conservative position when movement is required to be suspended. Other changes to the Bases are being made to be consistent with the TS. These changes do not alter the design or configuration of the plant. There has been no physical change to plant systems, structures, or components. The proposed changes will not reduce the ability of any of the safety-related equipment required to mitigate Anticipated Operational Occurrences (AOOs) or accidents. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes revise the control room TS Actions to delete the Action statement to suspend all operations involving positive reactivity changes, and adds an Applicability and Action related to the movement of irradiated fuel assemblies. The changes also add an Applicability footnote and revise the Bases to allow irradiated fuel assemblies to be placed in a safe conservative position when movement is required to be suspended. Other changes to the Bases are being made to be consistent with the TS. The margin of safety is not affected because the proposed changes to delete one Action and add an Applicability and Action ensures the assumptions of the accident analysis are being met. The administrative changes ensure the applicable TS are entered and eliminate confusion associated with the discrepancies between the TS and Bases. The more restrictive changes of adding an Applicability and Action eliminates the precursor to an event (fuel handling accident) that may release radioactivity when the control room ventilation systems are inoperable. The less restrictive changes revises the TS to rely on the instrumentation credited in the accident analysis and to allow irradiated fuel assemblies to be placed in a safe position to preclude a fuel handling accident. The instruments are required to be operable per TS. Compliance with these TS and also the SDM TS ensures that boron dilution event is precluded or can be mitigated. Therefore, suspension of positive reactivity changes is not required when the control room ventilation systems are inoperable. These Actions ensure that appropriate measures are taken to preclude events that would require the control room to be isolated when any of the control room ventilation systems are inoperable. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005–3502.

NRC Project Director: John N. Hannon.

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

Date of amendment request: August 12, 1998

Description of amendment request: The proposed amendment will change Technical Specifications (TS) 3.1.2.8, 3.5.1, 3.5.4, Figure 3.1–1, and Bases 3/4.5.2 for Waterford 3. It increases the maximum boron concentration in the Safety Injection Tanks (SITs) and the Refueling Water Storage Pool (RWSP) from 2300 ppm to 2900 ppm.

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 operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequence of any accident previously evaluated?

Response: No.

The proposed change increases the maximum boron concentration in the SITs and the RWSP from 2300 ppm to 2900 ppm. This change does not affect the probability of any accident. This increase in boron concentration affects the pH of water in the safety injection sump during a LOCA [Loss of Coolant Accident] and the potential for boron precipitation. The amount of TSP in containment is adequate to maintain the pH above 7.0. The revised long term cooling analysis shows that boron precipitation will not occur at the higher boron concentrations. Therefore, this change will not adversely impact post-LOCA core cooling. Thus, the consequences of a LOCA are not affected.

Therefore, the proposed change will not involve a significant increase in the probability or consequence of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change will not create any new system connection or interactions. Thus, no new modes of failure are introduced. There is no significant impact on the corrosion rate in the safety injection system due to the slightly higher acidic solution with the higher boron concentration.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in margin of safety?

Response: No.

Sufficient TSP [Trisodium Phosphate Dodecahydrate] is provided in the containment to ensure that the pH of the safety injection sump water during a LOCA remains above 7.0 as stated in the Technical Specification bases. Adequate time and HPSI [High Pressure Safety Injection] flow exist to avoid boron precipitation during a LOCA. The higher boron concentration limit will also allow higher refueling boron concentrations which will increase the available shutdown margin.

Therefore, the proposed change does 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 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 New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005–3502

NRC Project Director: John N. Hannon

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 31, 1998.

Description of amendment request: The proposed amendment would revise Improved Technical Specification (ITS) 5.6.2.10, ''Steam Generator (OTSG [once-through steam generator]) Tube Surveillance Program,'' to include a new repair process, called a ''repair roll'' or ''re-roll.'' The process would be used to repair steam generator tubes with defects within the upper tubesheet. Changes to inservice inspection and reporting requirements are proposed for tubes which are repaired using this process. The proposed revision would also require inspection of both OTSGs during each inservice inspection. In addition, several format and editorial changes are proposed to ITS 5.6.2.10 and to ITS 5.7.2, ''Special Reports,'' for clarification purposes.

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.

The proposed LAR [license amendment request] addresses several editorial and format changes which do not impact accident analyses. LAR #235 also proposes to implement the repair roll (re-roll) process.

The qualification of the re-roll joint is based on establishing a mechanical roll length which will carry all structural loads imposed on the tubes with required margins. A series of tests and analyses were performed to establish this length. Tests that were performed included leak, tensile, fatigue, ultimate load and eddy current measurement uncertainty. The analyses evaluated plant operating and faulted loads in addition to tubesheet bow effects. Any tube leakage will be bounded by the main steam line break (MSLB) evaluation presented in the Final Safety Analysis Report (FSAR). The proposed change also requires inspections of the joints created by the repair roll process. The addition of this inspection does not change any accident initiators. The proposed inspections after re-roll installation, and during future inservice inspections, assure continuous monitoring of these tubes such that inservice degradation of tubes repaired by the re-roll process will be detected. Based on the Framatome Technologies qualification, as well as the history for similar industry repair rolls, there are no new safety issues, as defined in BAW–2303P, Revision 3, associated with the repair roll. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

No new failure modes or accident scenarios are created by the re-roll process. The new pressure boundary joint created by the repair roll process has been shown by testing and analysis to provide structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. Furthermore, the testing and analysis demonstrate the repair roll process creates no new adverse effects for the repaired tube and does not change the design or operating characteristics of the OTSGs. In the unlikely event that a tube with a repair roll should fail and sever completely at the transition of the re-roll region, the tube would remain engaged in the tubesheet bore, preventing interaction with other surrounding tubes. In this case, leakage is bounded by the steam generator tube rupture (SGTR) accident analysis. Therefore, this change does not create a possibility of a new or different kind of accident from any previously evaluated.

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

The repair roll process effectively removes the defective/degraded area of the tube from service. The new roll expanded interface created with the tubesheet satisfies all the necessary structural, leakage and heat transfer requirements. The joint is constrained within the tubesheet bore; thus, there is no additional risk associated with tube rupture. The accident leakage is shown to be well within the initial assumption of the MSLB analysis of one gallon per minute primary-to-secondary leakage. Therefore, the FSAR analyzed accident scenarios remain bounding, and the use of the repair roll process does not reduce 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: 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 (CR–3), Citrus County, Florida

Date of amendment request: August 31, 1998.

Description of amendment request: The proposed amendment will change the Improved Technical Specifications (ITS) to add three additional Regulatory Guide (RG) 1.97 Type A Category 1 post-accident monitoring (PAM) instrumentation variables and one Type B Category 1 PAM instrumentation variable to ITS Table 3.3.17–1, Post-Accident Monitoring Instrumentation. The Type A Category 1 variables added are low pressure injection (LPI) pump run status, LPI suction from reactor building (RB) sump isolation valves DHV–42 and DHV–43 open position, and high pressure injection (HPI) pump run status. The Type B Category 1 variable added is reactor coolant system (RCS) low range pressure.

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

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

The addition of post-accident monitoring instrumentation to the CR–3 ITS and ITS Bases is to ensure instrumentation is available for use by the operators for performing manual actions, or to verify automatic actions have occurred, which are required to mitigate the effects of a design basis accident. The instrumentation is used for monitoring by the operators only after an accident occurs, performs no automatic functions, and there are no credible failures of this instrumentation which could initiate any accident previously evaluated. Therefore, the probability of occurrence of any accident previously evaluated is unaffected.

The availability and use of this instrumentation ensures that the prescribed manual operator actions for mitigating the consequences of an accident will be implemented when necessary, and that the operator has sufficient information to verify required automatic actions have occurred when necessary. Therefore, the availability and use of the instrumentation provides assurance that the consequences of accidents will not be greater than that previously evaluated.

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

The addition of post-accident monitoring instrumentation to the CR–3 ITS and ITS Bases is to ensure instrumentation is available for use by the operators for performing manual actions, or to verify automatic actions have occurred, which are required to mitigate the effects of a design basis accident. The instrumentation is used for monitoring by the operators only after an accident occurs, performs no automatic functions, and there are no credible failures of this instrumentation which could initiate a new or different kind of accident. Therefore, the possibility of a new or different kind of accident occurring as a result of this passive instrumentation is not created.

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

The addition of post-accident monitoring instrumentation to the CR–3 ITS and ITS Bases is to ensure instrumentation is available for use by the operators for performing manual actions, or to verify automatic actions have occurred, which are required to mitigate the effects of a design basis accident. The instrumentation is used for monitoring by the operators only after an accident occurs, and performs no automatic functions. The availability and use of this instrumentation ensures that the prescribed manual operator actions for mitigating the consequences of an accident will be implemented when necessary, and that the operator has sufficient information to verify required automatic actions have occurred when necessary. These required manual and automatic actions are necessary to preserve the margin of safety as defined in the CR–3 ITS and ITS Bases. The availability and use of this instrumentation provides assurance that the existing margin of safety will be maintained, and assumptions related to the margin of safety during mitigation of design basis accidents will be preserved. Therefore, the existing margin of safety will not be reduced.

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.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey.

Date of amendment request: May 5, 1998.

Description of amendment request: This request is to change the licensing basis to allow for a small amount of containment overpressure to ensure sufficient net positive suction head for the Emergency Core Cooling System pumps under post Loss of Cooling Accident (LOCA) conditions.

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

The proposed change to the licensing basis does not ''Involve a significant increase in the probability or consequences of an accident previously evaluated * * *''. As the strainers have no function until after the design basis LOCA occurs, the design of the strainer cannot affect the probability of a Large Break LOCA.

The requested change to raise the assumed containment overpressure for suction strainer design to 1.25 psig is less than that which is already used in LOCA analyses for offsite releases. Therefore, this change will not increase the offsite consequences of any previously analyzed accident. The frequency of a design basis LOCA occurrence at the Oyster Creek Nuclear Generating Station is conservatively estimated at 5.67×10^{-4} per year. The frequency of a design basis LOCA with a loss of containment overpressure is conservatively estimated at 2.46×10^{-7} per year.

Since the frequency of the design basis LOCA coincident with a loss of containment overpressure is insignificant (2.46×10^{-7}), the requested increase does not significantly impact the probability of exceeding the existing design bases. The core damage frequency increase due to the request for overpressure is mitigated, in part, by the current procedural requirement to flood containment following the design basis LOCA, thereby obviating the need for over pressure in the long term. The risk evaluation, performed in support of the request for over pressure, indicated a non-risk significant change in the core damage frequency.

The proposed change to the licensing bases does not ''Create the possibility of a new or different kind of accident from any accident previously evaluated * * *''. Both the new and existing strainers are passive. They function solely to prevent debris from entering the suction of the core and containment spray pumps. The only significant difference is that the new strainers can remove more debris without clogging. The slight amount of containment overpressure does not affect the operation of the strainers, and improves the ability of the core spray and containment spray systems to continue operation. Therefore, no new or different kind of accident is created or possible.

The proposed change to the licensing bases does not ''Involve a significant reduction in a margin of safety * * *.'' The modification increases the amount of debris that can be removed while maintaining core spray system operation. The requested change takes credit for 1.25 psig of wetwell overpressure. However, as the requested change is bounded by existing calculations for offsite release, no significant reduction in the margin of safety can occur. Additionally, as demonstrated in Attachment III, the probability of a LOCA with a loss of containment overpressure is not significant.

Guidance has been provided in ''Final Procedures and Standards on No Significant Hazards Considerations,'' Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provided examples of amendments which are and are not considered likely to involve significant hazards considerations.

Based on the above evaluation and the review of 51 FR 7744, this proposed change to the licensing basis of the Oyster Creek Nuclear Generating Station does not involve irreversible changes, a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the **Federal Register** and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazard.

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: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Attorney for Licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cecil O. Thomas.

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 9, 1998.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) by: (1) Changing the TS Definitions 1.24, ''Core Operating Limits Report,'' 1.27, ''Engineering Safety Feature Response Time,'' and 1.31, ''Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)''; (2) changing TS 3.0.2, ''Limiting Condition For Operation,'' by adding a new TS 3.0.6 to the Limiting Condition For Operation TS section; (3) changing TS 4.0.5, ''Surveillance Requirements''; (4) changing the mode applicability of TS 3.2.3, ''Total Unrodded Integrated Radial Peaking Factor—F_r^T''; (5) changing TS 3.3.2.1, ''Engineered Safety Features Actuation System Instrumentation,'' by modifying TS Table 4.3–2 Table Notation (1) which it references; (6) changing TS 3.4.1.1, ''Reactor Coolant System—Coolant Loops and Coolant Circulation Startup and Power Operation'; and (7) changing TS 3.4.11, ''Reactor Coolant System—Reactor Coolant System Vents.'' The associated TS Bases sections would also be updated to reflect the proposed changes. The proposed changes would resolve identified compliance issues.

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.

Technical Specification Definitions

The minor editorial and non-technical changes to correct reference, spelling and terminology errors contained in the definitions will not result in any technical changes to the Millstone Unit No. 2 Technical Specifications. The proposed changes will have no adverse effect on plant operation. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.0.6

The new Technical Specification, 3.0.6, will provide guidance on returning inoperable equipment to service, under administrative control, to demonstrate operability of that equipment, or the operability or other equipment. Various Technical Specification Actions require inoperable equipment to be removed from service, such as maintaining a containment isolation valve closed or tripping/bypassing a failed instrument channel. An exception to these required actions is necessary to allow the performance of testing to demonstrate the operability of the equipment being returned to service. Specifically, this Technical Specification addresses the situation where the inoperable equipment has been repaired, tested to the extent possible, and believed to be capable of performing its function. At this point, a presumption of the operability of the equipment is reasonable, and is supported by experience. Therefore, it is acceptable to place the equipment in service for testing under administrative control. Administrative controls will be used to ensure the time the equipment is returned to service is consistent with the Action Statements and is limited to the time necessary to perform the surveillance requirements.

This specification will also allow the inoperable equipment to be placed in a condition different from that required by the action statement to demonstrate the operability of other equipment. An example would be during the performance of an operability test on one reactor protection channel while another channel associated with the same function is inoperable. In this situation only one of the channels could be in the tripped condition, otherwise a reactor trip would be initiated. This is already permitted for reactor protection channels by Technical Specifications 3.3.1.1, ''Instrumentation—Reactor Protective Instrumentation,'' Action 2, and for engineered safety features channels by 3.3.2.1, ''Instrumentation—Engineered Safety Feature Actuation System Instrumentation,'' Action 2.

This provision is provided only to perform surveillance requirements to prove operability, and not to provide time to perform any other preventive or corrective maintenance. The testing will be performed consistent with the current Technical Specification Action Statement and will be limited to the time necessary to perform the surveillance requirement. The proposed changes will have no adverse effect on plant operations. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 4.0.5

The proposed changes will revise Technical Specification 4.0.5.a and Bases 3/4.4.10, ''Structural Integrity,'' by removing the phrase ''(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).'' The changes to Technical Specifications clarify that all applicable requirements in 10 CFR 50.55a apply. The changes relate to inservice inspection (ISI) and inservice testing (IST) requirements which are specified in 10 CFR 50.55a, ''Codes and Standards.'' The ISI and IST requirements are given in 10 CFR 50.55a, which the licensee documents via its 10 year interval program requirements. Upon finding a Code requirement impractical because of limitations in the design (including prohibitive dose rates), construction, or system configurations, NNECO [Northeast Nuclear Energy Company] would be required to prepare the determination describing the impractical condition(s) and the applicable code requirements that cannot be met in accordance with 10 CFR 50.55a, paragraphs (f)(5)(iii) and (iv), and (g)(5)(iii) and (iv) if within the first 12 months of a new interval. For example, 10 CFR 50.55a(f)(5)(iv), and (g)(5)(iv) allow a licensee up to a full year after the beginning of an updated interval to inform the NRC of the new Code requirements which cannot be met and to request relief. If an impracticality is identified after the first 12 months, the guidance contained in NUREG–1482 will be followed. This will eliminate inconsistencies between the Technical Specifications and the regulations. There will be no adverse effect on plant operations. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.2.3

The proposed change will change the mode of applicability for Technical Specification 3.2.3 from Mode 1 to Mode 1 with thermal power >20%. Data from the incore detectors are used for determining the measured radial peaking factors to verify compliance with Technical Specification 3.2.3. However, the accuracy of the neutron flux information from the incore detectors is not reliable below 20% power. The proposed change acknowledges this limitation of the incore detectors by changing the applicability of this specification to power levels where the data from the incore detectors is reliable. This will have no adverse effect on plant operations since the current Technical Specification surveillance requirements do not require the verification of this limit until prior to operation above 70% following each fuel loading, prior to 31 days accumulated operation in Mode 1, or if the azimuthal power tilt limit is exceeded (Technical Specification 3.2.4 which is applicable in Mode 1 above 50% power). Therefore, the proposed change has no impact on the initial conditions, with respect to power distribution, assumed in the accident analysis. Thus, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.3.2.1

The proposed change will add an exception to Technical Specification 4.0.4 that will allow the channel functional test of the automatic actuation logic associated with ESF [engineered safety feature] actuations for safety injection, containment spray, containment isolation, main steam line isolation, enclosure building filtration, and containment sump recirculation to be delayed during plant startup until the actuation blocks are removed. This will allow entry into Mode 3 where plant conditions (sufficient pressurizer and steam generator pressure) can be established that will automatically remove the blocks of these ESF actuations. The channel functional test of the automatic actuation logic, using the ATI [Automatic Testing Insertor] circuit, will then be performed. In addition, the channel functional tests of the automatic actuation logic must be performed prior to entering Mode 2.

The exception to Technical Specification 4.0.4 allows a mode change with equipment that is inoperable only because conditions [cannot] be established to perform the SR [surveillance requirement] until after the mode is entered. All other equipment operability requirements must be met. Even though operability of the automatic actuation logic for the affected ESF actuations cannot be verified prior to entering Mode 3, this equipment is still expected to be operable. The ESFAS [engineered safety feature actuation system] will continue to function as before. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.1.1

The Flow Dependent Setpoint Selector Switch was installed to allow power operation with less than four reactor coolant pumps (RCPs) in operation by changing the reactor trip setpoints for the variable high power, Reactor Coolant System (RCS) low flow, and thermal margin low pressure (TM/LP) reactor trips. Millstone Unit No. 2 is not currently licensed to operate with less than four RCPs in operation. Therefore, this switch should be maintained in the four pump position.

The use of the switch position to ensure compliance with Technical Specification 3.4.1.1 provides an indirect verification of LCO [limiting condition for operation] compliance since the loss of an RCP will result in a reactor trip when in the four pump position. The proposed change will replace the method used for LCO verification with one that is more consistent with the LCO. Verification of switch position is performed as a prerequisite prior to reactor startup (entering Mode 2). It is not necessary to verify the switch position every 12 hours as currently required. The position of this switch is important to the operability of the associated Reactor Protection System (RPS) trips variable high power, RCS low flow, and TM/LP). The operability of these RPS trips and associated setpoints is already covered by Technical Specifications 2.2.1, ''Reactor Trip Setpoints,'' and 3.3.1.1, ''Reactor Protective Instrumentation.''

It is not necessary to verify the position of this switch fifteen minutes prior to reactor criticality since the switch position is verified prior to a reactor startup, and is not expected to be changed during power operation. If surveillance testing or maintenance activities are to be performed which may require the switch to be in other than the four pump position, the affected RPS channels will already have been removed from service (declared inoperable and placed in the tripped or bypassed condition) prior to commencing the activities. In addition, a light (''PUMP SETPOINT ERROR'') on each of the RPS Calibration and Indication Panels will illuminate if the switch is not in the four pump position.

It is also not necessary to verify compliance with the requirements of Technical Specification 3.4.1.1 within fifteen minutes prior to reactor criticality since this condition is verified prior to a reactor startup, and the RPS will initiate a reactor trip if less than four RCPs are in operation.

The proposed change will replace SR 4.4.1.1, verification of the Flow Dependent Setpoint Selector Switch position, with a verification check of the required RCS loops. This verification is more consistent with the Limiting Condition for Operation (LCO). This will not change the requirement that both RCS loops be operable and operating in Modes 1 and 2. The Technical Specification will continue to assure that the initial condition, with respect to RCS loops in service, in the accident analysis is applicable. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.11

The proposed change to modify the wording of SR 4.4.11.3 will not affect the operability requirements of the RCS Vent System. This change will provide operational flexibility to use a series of overlapping tests to verify flow through sections of the vent system, such that when completed, flow will be verified through all parts of the vent system. This will minimize potential contamination of the area surrounding the sparger and will eliminate the need to establish solid water conditions in the RCS.

The proposed surveillance requirement will still verify the ability of the vent valves to operate. This will provide reasonable assurance of system operability and availability if needed to mitigate the consequences of design basis accidents. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no adverse effect on any of the design basis accidents previously evaluated or on any equipment important to safety. Therefore, the license amendment request does not impact the probability of an accident previously evaluate 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 will 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. Therefore, the proposed changes 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 proposed changes will correct reference, spelling, and terminology errors in various Technical Specification Definitions; add a new Technical Specification, 3.0.6; modify Technical Specification 4.0.5 to remove an inconsistency between the Technical Specification and the regulations; change the applicability of Technical Specification 3.2.3; add an exception to Technical Specification 4.0.4 to Technical Specification 3.3.2.1; modify the wording of a surveillance requirement associated with RCS Technical Specification 3.4.1.1; and modify the wording of a surveillance requirement associated with the RCS Vent System, Technical Specification 3.4.11 to provide operational flexibility in the performance of the test. These changes will have no adverse effect on equipment important to safety. The equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction of the margin of safety as defined in the Bases for the Technical Specifications affected by these proposed changes.

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

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

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

NRC Project Director: William M. Dean.

Philadelphia Electric Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of amendment request: September 14, 1998.

Description of amendment request: The proposed amendment to the Limerick Generating Station (LGS), Unit 2, Technical Specifications (TS) would revise TS Table 4.4.6.1.3–1, ''Reactor Vessel Material Surveillance Program—Withdrawal Schedule.'' This table provides the schedule for withdrawing the reactor pressure vessel material surveillance program capsules. This proposed TS change involves revising the schedule for withdrawing the first surveillance capsule from 8 Effective Full Power years (EFPY) to 15 EFPY, and the second surveillance capsule from 20 EFPY to 30 EFPY.

A revision to TS Surveillance Requirement (SR) 4.4.6.1.4 is also proposed. This revision will remove the reference to flux wire removal and analysis that was originally required following the first cycle of operation. TS SR 4.4.6.1.4 will be changed to refer to the flux wires that are located within the surveillance capsules, which will be removed and analyzed in accordance with the surveillance capsule removal schedule, located in Table 4.4.6.1.3–1.

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

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

The proposed changes do not increase the probability of occurrence of an accident previously evaluated in the safety analysis report and do not affect any accident initiators as described in the Safety Analysis Report (SAR). The change revises the withdrawal schedule for the reactor vessel material surveillance capsules. The capsules are not an initiator of any previously analyzed accident nor does the withdrawal schedule of the surveillance capsules affect the probability or consequences of any previously analyzed accident.

The proposed changes will not affect the Pressure-Temperature (P-T) limits as specified in LGS TS Figure 3.4.6.1–1 and Updated Final Safety Analysis Report (UFSAR) Figure 5.3–4. P-T limits are imposed on the reactor coolant system to ensure that adequate safety margins exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P-T limits are related to the RT_{NDT} [reference temperatures], as described in ASME Section III, Appendix G. Changes in the fracture toughness properties of RPV [reactor pressure vessel] beltline materials, resulting from neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10 CFR 50 Appendix H. The effect of neutron fluence on the shift in the RT_{NDT} is predicted by methods given in Regulatory Guide 1.99, Rev.2.

As detailed in Attachment 3 [of the September 14, 1998, submittal], for LGS, Unit 2, the combination of low expected RT_{NDT} shift for the plate material due to low predicted fluence and excellent material chemistry; Supplemental Surveillance Program (SSP) data on similar material; and the inherent margin in the P-T curve calculations, with the withdrawal schedule of the first surveillance capsule modified from 8 EFPY to 15 EFPY and the second surveillance capsule modified from 20 EFPY to 30 EFPY, will result in more credible sets of surveillance data, while ensuring the continued safe operation of LGS, Unit 2.

The current LGS P–T limits were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in Regulatory Guide 1.99, Revision 2, Regulatory Position 1, ''Surveillance Data Not Available.'' Calculation of adjusted reference temperature by these procedures includes a conservative base fluence estimate; power rerate adjustment of a 110% fluence multiplier from startup, instead of a 105% fluence multiplier since 2R03 [third refueling outage]; and a margin term to ensure conservative, upper-bound values are used for the calculation of the P–T limits. Revision of the first capsule withdrawal schedule will not affect the P–T limits because they will continue to be established in accordance with Regulatory Position 1 guidance. Also, as indicated in Attachment 3, it is also appropriate to extend the withdrawal of the LGS, Unit 2, second capsule. The current schedule specifies withdrawal of the second capsule at 20 EFPY. Based upon the information provided in Attachment 3 supporting withdrawal of the first capsule at 15 EFPY, there will be an insignificant shift in material properties at 20 EFPY, after only an additional exposure of 5 EFPY. It is appropriate to extend this schedule to 30 EFPY which meets the intent of ASTM E185–82, such that the withdrawal of the second capsule occurs before the accumulated neutron fluence of the capsule corresponds to the approximate EOL [end of life] fluence at the reactor pressure vessel inner wall location, and provides consistency with the LGS, Unit 1, withdrawal schedule.

In accordance with the guidance stipulated in Regulatory Guide 1.99, ''Radiation Embrittlement of Reactor Vessel Materials,'' Revision 2, Regulatory Position 2, ''Surveillance Data Available,'' the collection of two (2) or more sets of credible surveillance data is necessary to empirically calculate the adjusted reference temperature (ART). Each surveillance capsule constitutes one set of credible surveillance data. This calculated ART can be used to revise the P–T curves (TS Figure 3.4.6.1–1). Without two (2) or more sets of credible data, the ART must be calculated and the P-T curves revised, based upon the calculational methodologies as provided in the Regulatory Guide 1.99, Revision 2, Regulatory Position 1, ''Surveillance Data Not Available.'' These methodologies use plant specific chemistry and fluence values to determine a calculated shift in RT_{NDT}. A ''margin'' term is then added, to obtain conservative, upper-bound values of adjusted reference temperature.

The existing LGS, Unit 2, P–T curves are based upon the Regulatory Position 1 methodology, and are currently valid up to 10 EFPY. With first capsule removal at either 8 or 15 EFPY, the existing P–T curves will require a revision, prior to reaching 10 EFPY, based upon the calculational methodologies as contained in the Regulatory Guide 1.99, Revision 2, Regulatory Position 1, ''Surveillance Data Not Available.'' Therefore, the Technical Specification revision to the first capsule withdrawal schedule, as supported by this Safety Evaluation [supporting information described in attachments 1 and 3 of the September 14, 1998, submittal], results in no impact to the calculational methodologies that will be used for the P-T curve revision that will be necessary to extend the curves beyond 10 EFPY.

The fluence data as determined from the surveillance capsule flux wires at 15 EFPY will provide an accurate indication of neutron fluence. In accordance with Regulatory Guide 1.99, Revision 2, Regulatory Position 1 methodology, data from these flux wires will permit an adjustment of TS Figure 3.4.6.1–1 in accordance with TS SR 4.4.6.1.3, if required, and will meet the requirements of 10 CFR 50, Appendix H, and ASTM E–185.

The proposed changes will not affect any plant safety limits or limiting conditions of operation. The proposed changes will not affect reactor pressure vessel performance as it involves no physical changes and LGS P–T limits will remain conservative in accordance with Regulatory Guide 1.99, Revision 2, guidance. The proposed changes will not cause the reactor pressure vessel or interfacing systems to be operated outside of their design or testing limits.

The proposed changes do not increase the probability of the occurrence of a malfunction, or consequences of a malfunction, of equipment important to safety previously evaluated in the SAR. The proposed changes do not involve any physical changes to equipment important to safety. The potential for reactor vessel failure will be adequately assessed by the proposed withdrawal schedule. In addition, the results from the Supplemental Surveillance Program (SSP) will provide industry data that bounds the materials used in the LGS vessel until the data from the first LGS capsule is available. The proposed change provides the same level of confidence in the integrity of the vessel. The P–T curves are currently controlled by the TS and are determined using the conservative methodology delineated in Regulatory Guide 1.99. Therefore, the possibility of failure of the reactor vessel is not increased. The current P–T limit curves are inherently conservative and will continue to be adhered to.

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

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

The proposed changes do not create the possibility of a different type of accident than any previously evaluated in the SAR. The proposed changes are a revision of the withdrawal schedule for the first reactor pressure vessel material surveillance capsule from 8 EFPY to 15 EFPY, and for the second capsule from 20 EFPY to 30 EFPY. The proposed changes do not involve a physical modification of the design of plant structures, systems, or components. The proposed changes will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. No new accident types or failure modes will be introduced as a result of the proposed change.

LGS's current P–T limits were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in Regulatory Guide 1.99, Revision 2, Regulatory Position 1, ''Surveillance Data Not Available.'' Calculation of adjusted reference temperature by these procedures includes a conservative base fluence estimate; power rerate adjustment of a 110% fluence multiplier from startup, instead of a 105% fluence multiplier since 2R03; and a margin term to ensure conservative, upper-bound values are used for the calculation of the P–T limits. Revision of the first capsule withdrawal schedule will not affect the P–T limits because they will continue to be established in accordance with the guidance of Regulatory Position 1 of Regulatory Guide 1.99. Also, as specified in Attachment 3, it is appropriate to extend the withdrawal of the LGS, Unit 2, second capsule. The current schedule specifies withdrawal of the second capsule at 20 EFPY. Based upon the information provided in Attachment 3 supporting withdrawal of the first capsule at 15 EFPY, there will be an insignificant shift in material properties at 20 EFPY, after only an additional exposure of 5 EFPY. It is appropriate to extend this schedule to 30 EFPY which meets the intent of ASTM E185–82, such that the withdrawal of the second capsule occurs before the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor inner wall location, and provides consistency with the LGS, Unit 1, withdrawal schedule.

The existing LGS, Unit 2, P–T curves are based upon the Regulatory Position 1 methodology, and are currently valid up to 10 EFPY. With first capsule removal at either 8 or 15 EFPY, the existing P–T curves will require a revision, prior to reaching 10 EFPY, based upon the calculational methodologies as contained in the Regulatory Guide 1.99, Revision 2, Regulatory Position 1, ''Surveillance Data Not Available.'' Therefore, the proposed TS revision to the first capsule withdrawal schedule results in no impact to the calculational methodologies that will be used for the P–T curve revision that will be necessary to extend the curves beyond 10 EFPY.

The fluence data as determined from the surveillance capsule flux wires at 15 EFPY will provide an accurate indication of neutron fluence. In accordance with Regulatory Guide 1.99, Revision 2, Regulatory Position 1 methodology, data from these flux wires will permit an adjustment of TS Figure 3.4.6.1–1 in accordance with TS SR 4.4.6.1.3, if required, and will meet the requirements of 10 CFR 50, Appendix H, and ASTM E–185.

The potential for reactor vessel failure will be adequately assessed by the proposed withdrawal schedule. In addition, the results from the SSP will provide industry data that bounds the materials used in the LGS vessel, until the data from the first LGS capsule is available. The proposed changes provide the same level of confidence in the integrity of the vessel . The P–T curves are currently controlled by the TS and are determined using the conservative methodology in Regulatory Guide 1.99. Therefore, the possibility of failure of the reactor vessel is not increased. The current P–T limit curves are inherently conservative and will continue to be adhered to.

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

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

The proposed changes to the TS do not reduce the margin of safety as defined in the Bases for any TS. The proposed changes will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed changes do not represent a change in initial conditions, system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any TS. The proposed changes do not involve revision of the P–T limits, but rather a revision of the withdrawal schedule for the surveillance capsules. The current P–T limits were established based on the adjusted reference temperatures for reactor pressure vessel beltline materials calculated in accordance with the guidance stipulated in Regulatory Position 1 of Regulatory Guide 1.99, Revision 2. P–T limits will continue to be revised as necessary for changes in adjusted reference temperature due to changes in fluence according to Regulatory Position 1 until two (2) or more credible surveillance data sets becomes available. When two (2) or more credible surveillance data sets become available, P–T limits will be revised as prescribed by Regulatory Position 2 of Regulatory Guide 1.99, Revision 2, or other NRC approved guidance.

The current P–T limit curves are inherently conservative and provide sufficient margin to ensure the integrity of the reactor vessel. The changes do not adversely affect these curves. The fluence data as determined from the surveillance capsule flux wires at 15 EFPY will provide an accurate indication of neutron fluence. In accordance with Regulatory Guide 1.99, Revision 2, Regulatory Position 1 methodology, data from these flux wires will permit an adjustment of TS Figure 3.4.6.1–1 in accordance with TS SR 4.4.6.1.3, if required, and will meet the requirements of 10 CFR 50, Appendix H, and ASTM E–185.

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

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

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Project Director: Robert A. Capra.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: April 16, 1998.

Description of amendment request: This application for amendment to the Indian Point 3 Technical Specifications (TSs) proposes to modify a testing requirement for the emergency diesel generators (EDGs).

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

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response:

No. The three Emergency Diesel Generators (EDG) at Indian Point 3 are designed to provide a source of power to support a safe and orderly plant shutdown in the event that all other normal and standby sources of power are not available, such as during a postulated Loss of Offsite Power (LOOP). The probability of such events occurring is not affected by the proposed amendment. Any two of the three EDGs are capable of supplying the minimum power requirements for emergency safeguards equipment that mitigate the consequences of postulated design basis accident conditions. Periodic preventive maintenance and surveillance testing are performed to provide assurance that the operability of all three EDGs is maintained. In the event that an inoperable EDG is identified, both the existing specification and the proposed change provide for actions that verify the operability of the remaining 2 EDGs. Operability of 2 EDGs ensures that sufficient emergency power is available, if needed, to mitigate the consequences of postulated accidents. Therefore, the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously analyzed.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

No. The proposed license amendment does not involve any physical changes to plant systems or component setpoints. Also, there are no changes to the way in which systems or equipment are operated. The proposed change will continue to require that the operability of the remaining two EDGs be verified if one of the three EDGs is found to be inoperable. The proposed change to allow the use of a common cause failure evaluation, as an alternative to testing, to accomplish the operability verification can benefit overall EDG reliability by eliminating unnecessary EDG starts. Therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

No. Important performance requirements for the EDGs include electrical output capacity, elapsed time to start and reach rated output, and fuel storage supply to support a minimum period of operation. The proposed amendment does not change EDG performance requirements. The existing specification allows a period of 24 hours in which to verify the operability of the remaining 2 EDGs if one of the three EDGs is found inoperable. The proposed amendment does not change the 24-hour time limit. Operability verification, either by testing or evaluation, within 24 hours provides assurance that this source of emergency power is available if needed. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety. Also, this verification method has been approved for use with the current Standard Technical Specifications.

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: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: April 16, 1998, as supplemented August 20, 1998.

Description of amendment request: This application for amendment to Table 4.1–1 of the Indian Point 3 Technical Specifications (TSs) proposes to change surveillance frequency requirements for the various instrument channels to accommodate a 24-month operating cycle. The proposed amendment also revises Section 6 of the TSs to reflect updated analyses.

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:

Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response:

No. The proposed license amendment to extend the calibration surveillance frequency of the following instrument channels is being made to support plant operation with a 24-month fuel cycle:

(a) Pressurizer Water Level

(b) Accumulator Level and Pressure

(c) Reactor Coolant System Subcooling Margin Monitor

(d) Core Exit Thermocouples

(e) Reactor Vessel Level Indication System

Changing the calibration intervals for these instrument channels neither directly nor indirectly affects the initiation or probability of any previously analyzed accident. The changes do not affect the integrity of any of the principal barriers against radiation release (fuel cladding, reactor vessel, and containment building). The ability of the plant to mitigate the consequences of any previously analyzed accidents is not adversely affected. Evaluation of the proposed change to the surveillance interval demonstrates that licensing basis safety analyses acceptance criteria and Indian Point 3 Emergency Operating Procedure (EOP) criteria continue to be met.

Item (a) provides an input to the Reactor Protection System (RPS) to initiate a reactor trip if the measured parameters exceed specified values. Item (b) is used by control room operators to ensure that the accident mitigation capability of the accumulators is maintained within specified limits. Items (c), (d), and (e) provide post-accident information to control room operators to support recovery efforts. Item (d) is also used to monitor core performance for fuel management activities.

The proposed new surveillance frequency for these instrument channels was evaluated using the guidance of Generic Letter 91–04. The basis for the changes includes a quantitative evaluation of instrument drift. Also, loop accuracy/setpoint calculations were updated to accommodate the extended surveillance period. Analyses and evaluations completed to assess the proposed increase in the surveillance interval demonstrate that the effectiveness of these instruments in fulfilling their respective functions is maintained. Channel checks required to be performed each shift or each day, according to Technical Specifications for the subject channels, will continue to be performed to provide assurance of instrument channel operability. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of any previously analyzed accident.

Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

No. The increased calibration surveillance intervals for the above listed instrument channels were justified based on evaluation of past equipment performance and do not require any plant hardware changes or changes in normal system operation. Changing the calibration intervals for these channels neither directly nor indirectly has any means of creating the possibility of a new or different kind of accident. Certain alarm and EOP setpoint changes will be made consistent with the revised uncertainty calculations for the subject channels. These new setpoints and related operator responses support existing accident mitigation strategies and do not create the possibility of a new or different kind of accident from any previously analyzed. Therefore, there are no new failure modes introduced as a result of extending these surveillance intervals, and the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

No. Pressurizer water level instrumentation provides input to the reactor protection system and to the pressurizer water level control system. Pressurizer water level, as indicated by the selected control channel, is used to establish the initial condition pressurizer water level assumption for certain UFSAR [Updated Final Safety Analysis Report] Chapter 14 safety analyses. The proposed change to the calibration surveillance interval was evaluated using the criteria of 95% probability/95% confidence level for process sensor drift. The loop accuracy/setpoint calculations were updated for the level channels to demonstrate the acceptability of the proposed increase in the surveillance interval. There are no changes required to the limiting safety system setting (LSSS) stated in the Technical Specifications for these channels. The LSSS for high pressurizer water level will remain at [less than or equal to] 92% of span. The margin of safety between the specified LSSS value required by Technical Specifications and the safety limit used in the UFSAR Chapter 14 safety analyses is unchanged.

The instrument channels for accumulator pressure and level do not provide input to the reactor protection system or the engineered safety features system. These instruments provide alarms and indication to control room operators to maintain accumulator cover gas pressure and water volume within specified limits. They are also used for establishing initial condition accumulator pressure and level assumptions for certain UFSAR Chapter 14 safety analyses. Accordingly, the process sensor drift analysis was performed using the criteria of 95% probability/75% confidence level.

The remaining three instrument channels addressed by this proposed license change are used to provide indication of adequate core cooling following certain hypothetical accident conditions. These instrument channels are not associated with any margin of safety specified by the Technical Specifications, and they are not factors in any UFSAR Chapter 14 safety analyses. However, they are factored into the calculations of pertinent setpoints used in alarm response procedures and EOPs. The updated drift and uncertainty calculations and evaluations for these instrument channels demonstrate that applicable accuracy requirements for Indian Point 3 are satisfied with the proposed new surveillance intervals. The instrument channels will remain effective to support plant operator implementation of the Emergency Operating Procedures, which are consistent with the Westinghouse Owners' Group Emergency Response Guidelines.

Changing the calibration interval for these channels does not affect margin of safety for previously analyzed accidents. Also, the evaluation of related changes to UFSAR Chapter 14 safety analyses input assumptions has demonstrated that licensing basis safety analysis acceptance criteria and EOP criteria continue to be met, and previously existing margins based on these pertinent acceptance criteria continue to be maintained.

Therefore, the proposed amendment 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 50.92(c) are satisfied. The staff has also reviewed the licensee's proposed change to reflect updated safety analyses in Section 6 of the TSs and it appears that the three standards of 50.92(c) are satisfied for these changes as well. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: September 17, 1998.

Description of amendment request: The amendments would revise Technical Specification (TS) 3/4.8.2, ''Electrical Power Sources—Shutdown,'' for the AC distribution system and the 125-volt and 28-volt DC distribution systems. Specifically, the amendments would change the Applicability and Action Statements, if less than the complement of equipment and busses are operable, to eliminate the need to establish containment integrity and to add the action to suspend core alterations, positive reactivity additions, and movement of irradiated fuel assemblies.

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 not involve a significant increase in the probability or

consequences of an accident previously evaluated.

In Modes 1 through 4 [power operation through hot shutdown], a Design Basis Accident would cause the release of radioactive material into the containment. Release of that radioactive material to the environment is prevented during operation in Modes 1 through 4 by maintaining containment integrity. In Modes 5 and 6 [cold shutdown and refueling] the probability and consequences of this event are lower because of the reduced reactor coolant pressure and temperature limitations of these modes.

A minimum complement of electrical power sources and distribution systems is established in Modes 5 and 6 to assure that adequate electrical power is available to mitigate the consequences of a fuel handling accident. Because of the lack of containment pressurization potential during a fuel handling accident, less stringent requirements are needed to isolate containment from the outside atmosphere. These requirements are applied during refueling operations by Technical Specification 3.9.4, Refueling Operations, Containment Building Penetrations. Technical Specification 3.9.4 is applicable in Mode 6 and establishes containment closure vice containment integrity during refueling operation (core alterations and movement of irradiated fuel within containment).

In Mode 5, fuel handling is generally limited to placement of new fuel prior to core off load or movement of irradiated fuel within the spent fuel pool. Because the Spent Fuel Pool is not located within containment, establishment of either containment integrity or containment closure would not help to mitigate the consequences of a fuel handling accident in that area. Mitigation of a fuel handling accident is accomplished through Technical Specification 3.9.12, Refueling Operations, Fuel Handling Area Ventilation System, which requires that the Fuel Handling Area Ventilation system be operable whenever irradiated fuel is present in the storage pool. This insures that all radioactive material released from the rupture of an irradiated fuel assembly would be filtered through filtration equipment prior to discharge to the atmosphere.

With the number of energized A.C. or D.C. power distribution systems less than the required, sufficient power may not be available to recover from a fuel handling accident. Consequently, the Action statements require immediate suspension of all operations involving core alterations, positive reactivity changes, and movement of irradiated fuel assemblies. This precludes the possibility of a fuel handling accident and the need for containment integrity.

Based upon the above, the proposed change will not increase the probability or consequences of an accident previously analyzed.

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

The proposed changes do not require any change in the configuration or operation of the plant. Specifically, no new hardware is being added to the plant as part of the proposed change, no existing equipment is being modified, and no significant changes in operations are being introduced. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will not involve a significant reduction in a margin of safety.

The proposed change will not alter any assumptions, initial conditions, or results of any accident analyses. The proposed additional Applicability will ensure proper operation of the Fuel Handling Area Ventilation system during movement of irradiated fuel in the spent fuel pool. The proposed ACTIONS, to be taken in the event that the LCO [limiting condition for operation] is not met, will preclude the conditions that would lead to the need for establishing containment integrity. The change will, therefore, not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: September 29, 1998.

Description of amendment request: The proposed amendments would revise Technical Specification 3/4.9.4, ''Refueling Operations, Containment Building Penetrations,'' to permit the use of equivalent methods to obtain containment closure during refueling operations. Specifically, the proposed changes would allow the installation of an outage equipment door or other closure devices that are capable of providing access for temporary services needed to support maintenance activities within containment.

In addition to the above changes, the terminology for the Containment Equipment Hatch inside door used in LCO 3.9.4.a is being changed. The term ''Containment Equipment Door'' is being changed to ''Containment Equipment Hatch Inside Door'' to bring it into agreement with the terminology used in Salem design documents.

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

In Modes 1 through 4 [power operations through hot shutdown], a Design Basis Accident would cause the release of radioactive material into the containment. The release of radioactive material from the containment to the environment is prevented during operation in Modes 1 through 4 by maintaining CONTAINMENT INTEGRITY. In Mode 5 and 6 [cold shutdown and refueling] the requirements to prevent releases from the containment to the environment from postulated accidents are less stringent because of the reduced reactor coolant pressure and temperature limitations of these modes. In all cases, the containment serves as a passive barrier to mitigate the consequences of accidents analyzed. The containment is not considered to be a contributor to the probability of those accidents. Therefore, this change, which will permit the use of equivalent methods for establishing containment closure during refueling operations, will not increase the probability of an accident previously analyzed.

During refueling operations, a release of radioactive material to the containment could occur as the result of a fuel handling accident. Actions are taken to mitigate the consequences of a fuel handling accident inside containment during refueling operations through application of technical specification requirements for Refueling Cavity water level, minimum decay time prior to CORE ALTERATIONS, and Containment Building Penetrations.

Because of the lack of containment pressurization potential and the reduced source term during a fuel handling accident, less stringent requirements are needed to isolate containment from the outside atmosphere. These requirements are applied during refueling operations by Technical Specification 3.9.4, Refueling Operations, Containment Building Penetrations. Technical Specification 3.9.4 is applicable in Mode 6 and establishes containment closure vice CONTAINMENT INTEGRITY during CORE ALTERATIONS and movement of irradiated fuel within containment. Containment closure means that all potential release paths are closed or capable of being closed to provide an atmospheric pressure, ventilation barrier. Since there is no potential for containment pressurization, establishment of a pressure tight boundary is not required.

As a part of the containment closure requirements of Technical Specification 3.9.4, the Containment Equipment Hatch inside door must be installed with a minimum of four bolts. In addition, each penetration providing direct access from the containment atmosphere to the outside atmosphere must be closed by either an isolation valve, a blind flange, or a manual valve, or must be capable of being closed by an OPERABLE automatic containment isolation valve.

The proposed changes will modify Technical Specification 3/4.9.4 to permit the use of an equivalent closure device as an alternative to installation of the inner door with a minimum of four bolts to provide containment closure for the Containment Equipment Hatch. The proposed change will also modify Technical Specification 3.9.4 to permit the use of an equivalent method for containment closure for containment penetrations providing direct access from the containment to the outside atmosphere as an alternate method to closure by an isolation valve, blind flange, or manual valve. Any alternate method used will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that it is capable of providing containment closure during a fuel handling accident to prevent the release of fission product radioactivity to the environment. Because the proposed technical specifications must provide equivalent containment closure, these changes will not increase the consequences of an accident previously evaluated.

Based upon the above, the proposed changes do not increase the probability or the consequences of an accident previously evaluated.

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

The proposed changes do not require any change in the operation of the plant. The proposed changes will permit the use of an equivalent method to achieve containment closure for the Containment Equipment Hatch or for individual containment penetrations that provide direct access to the outside atmosphere. However, any equivalent method used will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the closure method meets design requirements.

Based upon the above, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will not involve a significant reduction in a margin of safety.

The proposed change will not affect the existing analysis that forms the basis for the Technical Specifications, and does not violate Technical Specification and Updated Final Safety Analysis Report (UFSAR) requirements. The proposed change will not affect any design or functional requirements of the containment, the Containment Equipment Hatch, or containment penetrations or any conditions or assumptions of the applicable safety analyses.

Based upon the above, the proposed changes will not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: 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: Robert A. Capra.

Southern California Edison Company, et al., Docket No. 50–362, San Onofre Nuclear Generating Station, Unit No. 3, San Diego County, California

Date of amendment request: September 22, 1998.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) to change the parameter used to establish and remove the bypasses for high reactor power trips. The parameter would be changed from the current ''THERMAL POWER'' to logarithmic power. This amendment was processed on San Onofre Nuclear Generating Station (SONGS) Unit 2 under emergency circumstances to allow resumption of power operations, and is being processed under normal notice circumstances on SONGS Unit 3.

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

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

The proposed change to Technical Specification (TS) 3.3.1 does not adversely impact structure, system, or component design or operation in a manner which would result in a change in the frequency of occurrence of accident initiation. The reactor trip bypass and automatic enable functions are not accident initiators. Consequently, the proposed TS change will not significantly increase the probability of accidents previously evaluated. Clarifying the input process variable of the operating bypasses and automatic bypass removals of the affected reactor trips does not alter the setpoint nor the manner of operation of the operating bypasses and automatic bypass removals. Therefore, the consequences of previously evaluated accidents remain unchanged.

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

No new or different accidents result from clarifying the input process variable of the operating bypasses and automatic bypass removals of the affected reactor trips. The results of previously performed accident analyses remain valid. Therefore, this amendment request does not create the possibility of a new or different kind of accident.

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

The proposed change does not alter the setpoint nor the manner of operation of the operating bypasses and automatic bypass removals of the affected reactor trips. The change merely replaces the identification of the input process variable with the appropriate identification of power. Therefore, this amendment request does not involve a significant reduction in any 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 requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: August 31, 1998.

Description of amendment request: The proposed amendment would revise the cold overpressure mitigation curves in Technical Specification (TS) Figure 3.4–4. This change would account for the TS maximum allowable power-operated relief valve setpoint changes associated with the new Model Delta 94 steam generator operating parameters.

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

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

The current pressurizer maximum allowable Power Operated Relief Valve (PORV) setpoints, provided by the Cold Overpressure Mitigation System (COMS) curves (Figure 3.4–4) of Technical Specification 3.4.9.3, are nonconservative for application with the new Delta 94 Replacement Steam Generators. The South Texas Project Cold Overpressure Event has been re-analyzed as a result of changed operating parameters due to installation of new Delta 94 Steam Generators. The re-analysis determined that maximum allowable PORV setpoint required decreases to ensure that the Cold Overpressure Mitigation System (COMS) continued to provide design basis low temperature overpressure protection with Delta 94 Steam Generators. New COMS curves have been developed and are to be incorporated into Technical Specification 3.4.9.3 by this change request. Since the proposed COMS curves result in maximum allowable PORV setpoint decreases to account for the changed Delta 94 Steam Generator operating parameters, these curves are more conservative than the existing COMS curves utilized for Model E Steam Generators. Therefore, application of these proposed COMS curves for a unit with Model E or Delta 94 Steam Generators ensures compliance with the original design basis of the Cold Overpressure Mitigation System for the South Texas Project.

This proposed change is based on a re-analysis which accounts for changed operating parameters associated with the Delta 94 Replacement Steam Generators. Reflecting actual operating parameters and adjusting the maximum allowable PORV setpoints, as necessary, in the conservative direction has no adverse effect on the probability or consequences of an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed PORV maximum allowable setpoint changes do no create any new operating conditions or modes. The proposed change only revises the maximum allowable PORV setpoint curves for the Cold Overpressure Mitigation System to account for the revised operating parameters associated with Delta 94 Steam Generators. The actions of this system continue to be performed in accordance with existing requirements, which are sufficient to ensure plant safety is maintained.

The proposed change is the result of a re-analysis of a previously evaluated accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed change reflects the revised operating parameters associated with the new Delta 94 Steam Generators. The revised COMS curves are the result of a re-analysis of the COMS analysis performed to ensure the margin of safety is not reduced with Delta 94 Steam Generators. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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: John N. Hannon.

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

Date of amendment request: August 31, 1998.

Description of amendment request: The proposed amendment would modify Technical Specification Surveillance Requirement 3.6.1.3.4 to permit removal of the inclined fuel transfer system primary containment blind flange while primary containment integrity is required.

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

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

The proposed change permits removal of the blind flange on the Inclined Fuel Transfer System (IFTS) when primary containment operability is required in Modes 1, 2 and 3. This will permit operation of IFTS when the plant is operating. This aspect of the containment structure does not directly interface with the reactor coolant pressure boundary. The removal of this blind flange does not involve modifications to plant systems or design parameters that could contribute to the initiation of any accidents previously evaluated. Operation of IFTS is unrelated to the operation of the reactor, and there is no aspect of IFTS operation that could lead to or contribute to the probability of occurrence of an accident previously evaluated. Removal of the blind flange and operation of IFTS does not result in changes to procedures that could impact the probability of occurrence of an accident.

With respect to consequences, the function of the containment is to mitigate the radiological consequences of a loss of coolant accident (LOCA) or other postulated events that could result in radiation release from the fuel inside containment. The pressure and temperature transient resulting from a design basis loss of coolant accident (LOCA) is considered the primary challenge to the integrity of the containment. While the proposed change does not change the plant design, it does permit alteration of the containment boundary for the IFTS penetration. Altering the containment boundary in this case (removing the blind flange) results in some IFTS components possibly seeing a containment pressure rise should a LOCA occur. The thermal and mechanical load requirements do not appreciably change as a result of such a small pressure increase (peak post-accident pressure (P_a) of 7.8 psig). The IFTS components will be more than adequate and capable of withstanding the Design Basis LOCA and associated loads prior to implementation of this amendment. Therefore, they are considered an acceptable barrier to prevent uncontrolled release of post-accident fission products for this proposed change.

The proposed change required examination of two potential leakage pathways. The larger is the transfer tube itself, the other, much smaller one, is the drain piping. It is clear that the gate valve at the bottom of the transfer tube is always water sealed and maintained so by the submergence of the water in the transfer tube and in the Fuel Handling Building Fuel Transfer Pool. The height of this water seal is greater than that necessary to prevent leakage from the bottom of the transfer tube during accidents that result in the calculated peak post-accident pressure (P_a). The potential leakage pathway from the drain piping which attaches to the transfer tube will be isolated if required, via administrative controls on the drain piping isolation valve. Additionally, the drain piping isolation valve will be added to the Primary Containment Leakage Rate Testing Program (Specification 5.5.12) to ensure that leakage past this valve will be maintained consistent with the leakage rate assumptions of the accident analysis. Due to the test methodology, the portion of the large transfer tube piping outboard of the blind flange (the portion of the tube which becomes exposed to containment air during the draining portion of the IFTS operation) will also be part of the leakage rate test boundary and will therefore also be tested with air. Therefore, no unidentified leakage paths will exist from the piping and components that are outboard of the blind flange, and the leakage rate assumptions of the accident analysis will be maintained.

Therefore, the proposed change does not result in a significant increase in the probability or the consequences of previously evaluated accidents.

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

The proposed change consists of the removal of a passive component which is not part of the primary reactor coolant pressure boundary nor involved in the operation or shutdown of the reactor. Being passive, its presence or absence does not affect any of the parameters or conditions that could contribute to the initiation of any incidents or accidents that are created from loss of coolant or positive reactivity. Re-aligning the boundary of the primary containment to include portions of the IFTS is also passive in nature and therefore has no influence on, nor does it contribute to the possibility of a new or different kind of incident, accident or malfunction from those previously analyzed. Furthermore, operation of IFTS is unrelated to the operation of the reactor and there is no mishap in the process that can lead or contribute to the possibility of losing any coolant in the reactor or introducing the chance for positive or negative reactivity or other accidents different from and not bounded by those previously evaluated.

Therefore, the proposed change does not result in creating the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed change will not involve a significant reduction in the margin of safety.

The proposed change involves the re-alignment of the primary containment boundary by removing the blind flange which is a passive component. The margin of safety that has the potential of being impacted by the proposed change involves the dose consequences of postulated accidents which are directly related to potential leakage through the primary containment boundary. The potential leakage pathways due to the proposed change have been reviewed, and leakage can only occur from the administratively controlled IFTS transfer tube drain piping. An individual will be designated to provide timely isolation of this drain piping during the durations of time when this proposed change is in effect. The conservatively calculated dose which might be received by the designated individual while isolating the drain piping is less than or equal to 1.9 rem, well within the guidelines of General Design Criterion 19. Furthermore, the drain piping isolation valve will be added into the Primary Containment Leakage Rate Testing Program (Specification 5.5.12) to ensure that leakage from the piping and components located outboard of the blind flange will be maintained consistent with the leakage rate assumptions of the accident analysis. Therefore, the dose consequences of an event would be unchanged, and the associated margin of safety would also be unchanged.

Therefore, the proposed change does 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: Perry Public Library, 3753 Main Street, Perry, OH 44081.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: September 3, 1998.

Description of amendment request: The proposed amendment would permit an Emergency Diesel Generator (EDG) Technical Specification (TS) Action Completion Time of up to 14 days for a Division 1 or 2 EDG and allow performance of the EDG 24-hour TS surveillance requirement test in modes 1 and 2.

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

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

The proposed Technical Specification changes do not significantly increase the probability of occurrence of a previously evaluated accident because the standby

Emergency Diesel Generators (EDGs), including the High Pressure Core Spray diesel generator, are not initiators of previously evaluated accidents. The EDGs mitigate the consequences of previously evaluated accidents involving a loss of offsite power. The proposed changes to the Technical Specification Action Completion Times do not affect any of the assumptions used in the deterministic or Probabilistic Safety Analysis (PSA).

The proposed Technical Specification changes will continue to ensure the EDGs perform their function when called upon. Extending the Technical Specification Completion Times to 14 days and allowing the performance of the EDG 24-hour run test in either Modes 1 or 2 does not affect the design of the EDGs, the operational characteristics of the EDGs, the interfaces between the EDGs and other plant systems, the function, or the reliability of the EDGs. Thus, the EDGs will be capable of performing their accident mitigation function and there is no impact to the radiological consequences of any accident analysis.

To fully evaluate the effect of the EDG Completion Time extension, PSA methods and deterministic analysis were utilized. The results of this analysis show no significant increase in the Core Damage Frequency. The proposed changes remain bounded by the Core Damage Frequency identified in the Individual Plant Examination.

The Configuration Risk Management Program (CRMP) is an administrative program that assesses risk based on plant status. Adding the requirement to implement the CRMP for Technical Specification 3.8.1 requires the consideration of other measures to mitigate consequences of an accident occurring while an EDG is inoperable.

The proposed change will not alter the operation of any plant equipment assumed to function in response to an analyzed event or otherwise increase its failure probability. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

This proposed change does not change the design, configuration, or method of operation of the plant. The proposed activity involves a change to the allowed plant mode for the performance of specific Technical Specification surveillance requirements. No physical or operational changes to the EDGs or supporting systems are made by this activity. Since the proposed changes do not involve a change to the plant design or operation, no new system interactions are created by this change. The proposed Technical Specification changes do not produce any parameters or conditions that could contribute to the initiation of accidents different from those already evaluated in the Updated Safety Analysis Report.

The proposed changes only address the methods used to ensure EDG reliability. Thus, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not affect the Limiting Conditions for Operation or their Bases that are used in the deterministic analysis to establish any margin of safety. PSA evaluations were used to evaluate these changes, and these evaluations determined that the changes are either risk neutral or risk beneficial. The proposed activity involves changes to certain Completion Times and to the allowed plant mode for the performance of specific Technical Specification Surveillance Requirements. The proposed change remains bounded by the existing Surveillance Requirement Completion Times and therefore has no impact to the margins of safety.

The proposed change does not involve a change to the plant design or operation, and thus does not affect the design of the EDGs, the operational characteristics of the EDGs, the interfaces between the EDGs, and other plant systems, or the function or reliability of the EDGs. Because EDG performance and reliability will continue to be ensured by the proposed Technical Specification changes, the proposed changes do not result in a 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: Perry Public Library, 3753 Main Street, Perry, OH 44081.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: September 9, 1998.

Description of amendment request: The proposed license amendment concerns hydrostatic (water) testing of containment isolation valves in the Feedwater System lines. The proposed technical specification change stipulates that water leakage from the feedwater motor-operated containment isolation valves will be added into the Primary Coolant Sources Outside of Containment Program (Technical Specification 5.5.2), and therefore the feedwater check valves do not need to be included in the hydrostatic test program addressed by Surveillance Requirement 3.6.1.3.11. The proposed testing change is based on design and licensing basis changes being implemented to improve functioning of the Feedwater Leakage Control 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) This proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

It is proposed that water leakage from the Feedwater motor-operated containment isolation valves will be added into the Primary Coolant Sources Outside Containment Program (Technical Specification 5.5.2), and therefore the Feedwater lines do not need to also be included in the hydrostatic test program addressed by Surveillance Requirement 3.6.1.3.11. The proposed testing change is based on design/licensing basis changes being implemented to improve functioning of the Feedwater Leakage Control System. The proposed design change will provide Feedwater Leakage Control System seal water directly to the bonnets and seats of the motor operated gate valves in the Feedwater lines, and allow for power to the valves to be provided from redundant power supplies.

The proposed changes do not increase the probability of occurrence of an accident previously evaluated because the Feedwater Leakage Control System is not an initiator of a previously evaluated accident. The Feedwater Leakage Control System is used to mitigate the consequences of an event that has already been initiated due to some other cause, specifically a design basis Loss of Coolant Accident (LOCA). Therefore, changes to the design and testing on the Feedwater Leakage Control System have no impact on the probability of occurrence of an accident previously evaluated. The Feedwater Leakage Control System is a manually initiated system, and the probability of an inadvertent initiation remains unchanged from that previously reviewed, so the possibility of a loss of feedwater transient is not increased.

The proposed changes do not significantly increase the radiological consequences of an accident previously evaluated, because the Feedwater lines will continue to be isolated following a LOCA either inside or outside of containment. For a line break outside of containment, the check valves will provide the necessary short-term closure function to prevent significant loss of reactor coolant inventory, as currently stated in Updated Safety Analysis Report (USAR) Section 6.2.4.2.2.1.a.1. The third (gate) valves in the Feedwater line will also be available to provide the long-term, high integrity leakage protection. The check valves Code Class 1 closure function will be verified at an appropriate frequency by performance of an exercise closed (EC) test comprised of a visual inspection of the internals of the valves, in accordance with the Inservice Testing Program. The radiological consequences of such a line break outside of containment event are not significant, as there is no postulated fuel damage.

For a line break inside of containment (a design basis LOCA event), the majority of the currently reviewed and accepted licensing basis is being maintained. Design changes are being implemented to improve the functioning of the Feedwater Leakage Control System. The redundant subsystems will be piped to the bonnets of the third, high integrity valves in the Feedwater lines (the gate valves) to provide a more rapid and effective seal on the stem, bonnet and flexible wedge seats. Water leakage from the stem, bonnets and seats of the gate valves will be addressed through controls imposed by Technical Specification 5.5.2, ‘‘Primary Coolant Sources Outside Containment.’’ The doses from such water leakage are accounted for in the radiological dose calculations. Since the leakage from the Feedwater lines is accounted for by the Primary Coolant Sources Outside Containment Program, there is no need to include the water test results of the Feedwater lines into the Surveillance Requirement 3.6.1.3.11 leak test totals.

The branch lines off of the Feedwater lines will also be addressed either through the Primary Coolant Sources Outside Containment Program (Technical Specification 5.5.2) or through additional Appendix J air leak rate test requirements (Technical Specification Surveillance Requirement 3.6.1.1.1 and Specification 5.5.12, ‘‘Primary Containment Leakage Rate Testing Program’’). The new test methods for these lines do not impact the existing radiological dose calculations, since the existing leakage limits of the leak rate test programs are not changed by the proposal.

The design changes associated with the Feedwater Leakage Control System will continue to satisfy licensing/design criteria for this piping to an equivalent degree as the current design. The minor exception is where the two Feedwater Leakage Control subsystems tie in to the bonnets of the gate valves, and this constitutes only a separation issue. Since the Feedwater Leakage Control System piping at this juncture is Code Class 2, break excluded, and protected from pipe whips and jet impingements, it is considered to be acceptable.

Addition of the provisions for an alternate power supply to be provided to the gate valves (if necessary following a LOCA event) will improve the probability of closure of these high integrity valves without creating an electrical separation concern. A separation concern will not be created since the supply circuitry from the alternate power source will be a permanent modification, and physical and electrical separation between electrical divisions will be maintained by employing two features:

1. Normally open, fused disconnect switches at both ends of the circuit, and
2. Fuses normally stored out of the circuit.

Based on the discussions above, it is concluded that neither the probability nor the consequences of previously evaluated accidents are significantly increased as a result of the proposed changes to the Technical Specifications and to the licensing bases for the Feedwater penetrations.

(2) This proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The Feedwater Leakage Control System was developed specifically to mitigate the consequences of a design basis LOCA inside the containment. The system itself and the proposed changes do not produce parameters or conditions that could contribute to the initiation of accidents different than those already evaluated in the Updated Safety Analysis Report. The proposed changes are intended to improve the functioning of the Feedwater Leakage Control System should it be called upon following a LOCA. The changes affect mitigation of that previously evaluated event.

In other plant conditions, including normal operation, the system is not activated and cannot induce events. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) This proposed amendment does not involve a significant reduction in a margin of safety.

The proposed changes only affect the methods used to ensure Feedwater Leakage Control System performance and reliability, and clarification of the licensing/design basis of the system. The new proposed Note in Surveillance Requirement 3.6.1.3.11 clarifies that the water leakage from the Feedwater lines does not need to be counted in two separate leak test programs. The Primary Coolant Sources Outside Containment Program (Technical Specification 5.5.2) will ensure that leakage from the Feedwater lines is minimized, and accounted for in an appropriate fashion in the radiological does calculations. Leak rate testing on the branch lines off of the Feedwater lines will also be controlled and limited by existing acceptance criteria for plant programs that protect the assumptions of the radiological dose calculations. Therefore, the margin of safety provided in the Perry Nuclear Power Plant dose calculations will remain unchanged.

The majority of the existing licensing basis, and therefore the margins of safety, are maintained by this proposal. The items that are changed are done so to improve the reliability of the system or for an administrative clarification. The Feedwater Leakage Control System Technical Specification itself (Technical Specification 3.6.1.8) does not need revision. The design changes will maintain the existing licensing/design criteria, with the minor exception of divisional separation at the point that the two divisions have to be piped into the bonnets of the third (gate) valve. Since the piping at this junction point is Code Class 2, break excluded, and protected from pipe whips and jet impingements, it is considered to be acceptable. It will not lead to a significant reduction in a margin of safety. The manually initiated divisional cross-tie will not create an electrical separation concern. The alternate power supply provision will be a permanent modification, and physical and electrical separation between electrical divisions will be maintained.

Based on the above discussions, the proposed license amendment is concluded to not result in 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: Perry Public Library, 3753 Main Street, Perry, OH 44081.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Stuart A. Richards.

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.

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 17, 1998.

Description of amendment request: The proposed amendments would allow a revision to the Oconee Updated Final Safety Analysis Report that addresses potential plant conditions that could occur during engineered safeguards functional tests of the emergency electrical system. These tests are planned to be performed on Unit 3 in November 1998, with Unit 3 in the cold shutdown condition, and Units 1 and 2 operating at power. If an actual loss-of-coolant accident with loss of offsite power were to occur on Unit 1 or 2, simultaneously with test initiation on Unit 3, the Emergency Power System would be placed in a condition outside the present design basis. This involves an unreviewed safety question that requires NRC approval before implementation of the tests.

Date of publication of individual notice in **Federal Register**: September 30, 1998 (63 FR 52304).

Expiration date of individual notice: October 30, 1998.

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

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: September 19, 1998.

Description of amendment request: The amendment would revise Section 5.4.8 of the Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR) such that it incorporates the use of a freeze seal as a temporary part of the reactor coolant pressure boundary.

Date of publication of individual notice in **Federal Register**: September 30, 1998 (63 FR 52307).

Expiration date of individual notice: October 30, 1998.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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

Date of amendment requests: January 29, 1997, as supplemented February 11, 12, March 7, 10, 11, 19, 20, April 29, June 30, and July 10, 1997, June 20, June 22, July 24 and September 15, 1998.

Brief description of amendment request: The proposed amendments would change the design basis of the cooling water system emergency intake line flow capacity. The licensee determined through testing that the emergency intake line flow capacity was less than the design value stated in the Updated Safety Analysis Report. The proposed changes reflect the use of operator actions to control cooling water system flow following a seismic event. The proposed changes also reclassify the intake canal for use during a seismic event, which would be an additional source of cooling water during a seismic event.

Date of publication of individual notice in **Federal Register**: October 1, 1998 (63 FR 52772).

Expiration date of individual notice: November 2, 1998.

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

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: October 14, 1997, as supplemented July 23, 1998.

Description of amendment request: The amendment would update the Technical Specifications to provide for installation of additional racks to increase spent fuel storage capacity, and to correct the maximum exposure dependent, infinite lattice multiplication factor for fuel bundles.

Date of publication of individual notice in **Federal Register**: August 24, 1998 (63 FR 45096).

Expiration date of individual notice: September 23, 1998.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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 amendment request: August 8, 1997, as supplemented by letters dated March 9, May 6, July 6, July 31, September 4, and September 11, 1998, and advanced information related to the application submitted April 17, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications to accommodate an increase in the maximum licensed thermal power level from 2558 megawatts thermal (MWt) to 2736 MWt.

Date of publication of individual notice in **Federal Register**: October 6, 1998 (63 FR 53730).

Expiration date of individual notice: November 5, 1998.

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

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: September 4, 1998.

Description of amendment request: The amendment would revise the Technical Specifications to reflect an increase in the spent fuel storage capacity.

Date of publication of individual notice in **Federal Register**: October 1, 1998. (63 FR 52774)

Expiration date of individual notice: November 2, 1998.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona

Date of application for amendment: June 13, 1995, as supplemented by letters dated August 16, 1995, June 9, 1998, and September 6, 1998.

Brief description of amendment: These amendments revise TS 3.5.1, ''Safety Injection Tanks (SITs)—Operating,'' and TS 3.5.2, ''Safety Injection Tanks—Shutdown,'' to extend the allowed outage times for the SITs.

Date of issuance: October 2, 1998.

Effective date: October 2, 1998.

Amendment No.: 118.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 25, 1995 (60 FR 54715)

The June 9, 1998, and September 6, 1998, letters provided additional clarifying information and do not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1998.

No significant hazards consideration comments received: No.

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

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: June 12, 1997, as supplemented by letter dated August 27, 1998. The August 27, 1998, supplemental letter provided clarifying information only, and did not change the initial no significant hazards consideration determination.

Brief description of amendment: This amendment changes the description of the Harris Nuclear Plant Operations organization in TS 6.0, ''Administrative Controls.''

Date of issuance: October 7, 1998.

Effective date: October 7, 1998.

Amendment No: 83.

Facility Operating License No. NPF–63. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 30, 1997 (62 FR 40847).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 7, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: December 30, 1997.

Brief description of amendments: The amendments change the Technical Specifications for the condensate storage tank (CST) level and the automatic auxiliary feedwater pump switchover from the suction of the CST to the essential service water system.

Date of issuance: October 6, 1998.

Effective date: October 6, 1998.

Amendment Nos.: 104; 104 & 96; 96.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 25, 1998. (63 FR 9596)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 6, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: May 18, 1998.

Brief description of amendments: The amendments will change several Technical Specification (TS) values to reflect design values. These TS values affect (1) 125/250 volts direct current (Vdc) electrolyte temperature; (2) control rod drive accumulator pressure; (3) standby liquid control solution temperature; (4) ultimate heat sink minimum water level; (5) shutdown suppression chamber level (Quad Cities only); and (6) a degraded voltage setpoint (Quad Cities only).

Date of issuance: October 8, 1998.

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: Dresden 169 & 164; Quad Cities 181 & 179.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 17, 1998 (63 FR 33105).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 8, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

Date of application for amendment: August 6, 1998.

Brief description of amendment: The amendment deletes Surveillance Requirement 4.8.1.1.2.i.2, regarding diesel fuel oil system pressure testing, from the Technical Specifications on the basis that the staff had previously approved alternative surveillance based on Code Case N–498–1 of the American Society of Mechanical Engineers.

Date of issuance: September 28, 1998.

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

Amendment No.: 164.

Facility Operating License No. NPF–52: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 17, 1998 (63 FR 43962).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 28, 1998.

No significant hazards consideration comments received: No.

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

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: October 22, 1996, as supplemented by letters dated March 19, July 6, and September 15, 1998.

Brief description of amendments: The amendments allow continued plant operation at elevated Containment Lower Compartment temperatures between 125 °F and 135 °F for a period not to exceed 72 cumulative hours per calendar year.

Date of issuance: September 28, 1998.

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

Amendment Nos.: Unit 1–183; Unit 2–165.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 12, 1997 (62 FR 6574).

The March 19, July 6, and September 15, 1998, submittals provided clarifying information that did not change the scope of the October 22, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 28, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application of amendment: July 21, 1998.

Brief description of amendment: The amendment permits an alternative to the requirement to perform Control Rod Drive scram time testing with the reactor pressurized prior to resuming power operation. The change permits: (1) scram time testing with the reactor depressurized prior to resuming operation, and (2) a second scram time test with the reactor pressure above 800 psig, prior to exceeding 40% reactor power.

Date of Issuance: October 1, 1998.

Effective date: October 21, 1998, to be implemented within 30 days.

Amendment No.: 198.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43204).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 1, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of application for amendment: March 23, 1998, as supplemented June 30, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.1.2, to incorporate new pressure/temperature limits for reactor vessel pressurization heatup, cooldown, and inservice leak and hydrostatic test.

Date of issuance: October 5, 1998.

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

Amendment No.: 208.

Facility Operating License No. DPR–50: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 22, 1998 (63 FR 19970). The June 30, 1998, submittal provided additional information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 5, 1998.

No significant hazards consideration comments received: No.

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

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: April 15, 1998.

Brief description of amendment: The amendment revises the Technical Specifications by updating the existing pressure-temperature curves with new curves with values from 18 to 32 effective full power years. Applicable surveillance requirements are also revised to reflect operation with the new curves.

Date of issuance: October 1, 1998.

Effective date: October 1, 1998.

Amendment No.: 224.

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

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25110).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, IA 52401.

Illinois Power Company, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: May 20, 1998, as supplemented July 17 and August 6, 1998.

Brief description of amendment: The amendment provides for automatic operation of a new emergency reserve auxiliary transformer to provide power to the plant 4.16-kV buses from the offsite 138-kV transmission network.

Date of issuance: October 1, 1998.

Effective date: October 1, 1998.

Amendment No.: 116.

Facility Operating License No. NPF–62: The amendment authorized revision of the Updated Safety Analysis Report.

Date of initial notice in **Federal Register**: June 4, 1998 (63 FR 30519).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

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 23, 1997, as supplemented by letter dated September 11, 1998. The September 11, 1998, letter provided the typed TS pages that did not change the Nuclear Regulatory Commission staff's proposed no significant hazards consideration determination.

Brief description of amendments: The proposed amendments would revise the Technical Specifications TSs to exclude the Main Steam Isolation Valves leakage from the total Type B and Type C local leak rate test results.

Date of issuance: October 1, 1998.

Effective date: The amendments are effective as of the date of issuance, and are to be implemented within 30 days from the date of their issuance.

Amendments Nos.: 223 and 227.

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

Date of initial notice in **Federal Register**: July 2, 1998 (62 FR 35852).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 1998.

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.

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 1, 1998, as supplemented by letter dated September 11, 1998.

Brief description of amendments: These amendments revise the technical specifications to delete the requirements for functional testing of safety relief valves during each unit startup.

Date of issuance: October 5, 1998.

Effective date: As of the date of issuance and is to be implemented, Unit 2, prior to October 1998 refueling outage and Unit 3, prior to October 1999 refueling outage.

Amendments Nos.: 224 and 228.

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

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40559).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 5, 1998.

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.

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: March 20, 1998, as supplemented by letters dated June 26, August 11, and September 14, 1998. The August 11 an September 14 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

Brief description of amendments: These amendments would revise the Technical Specifications to permit incorporation of end-of-cycle recirculation pump trip systems.

Date of issuance: October 5, 1998.

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

Amendments Nos.: 225 and 229.

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

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40558).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 5, 1998.

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.

Philadelphia Electric Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: March 24, 1997, as supplemented September 4, 1998.

Brief description of amendments: These amendments approve the deletion of the Drywell and Suppression Chamber Purge System operational time limit, removal of a footnote regarding 1-inch and 2-inch valves, and the addition of a surveillance requirement ensuring the purge system large supply and exhaust valves are closed as required.

Date of issuance: October 1, 1998.

Effective date: Units 1 and 2, As of date of issuance, to be implemented within 30 days.

Amendment Nos.: 130 and 91.

Facility Operating License Nos. NPF–39 and NPF–85: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 4, 1997 (62 FR 30643).

The September 4, 1998, letter 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 October 1, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

Power Authority of the State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County , New York

Date of application for amendment: November 13, 1997.

Brief description of amendment: The amendment changes the Technical Specifications by increasing the minimum test frequency for main turbine stop valves.

Date of issuance: October 5, 1998.

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

Amendment No.: 182.

Facility Operating License No. DPR–64: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 15, 1998 (63 FR 38203).

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

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

Date of application for amendment: May 13, 1998.

Brief description of amendment: This amendment revises Technical Specification (TS) 3/4.10.8, ''Inservice Leak and Hydrostatic Testing,'' to delete the requirement for an operable High Drywell Pressure trip function. Specifically, TS 3.10.8.a is being revised to remove the reference to the Secondary Containment Isolation Actuation Instrumentation trip function 2.b.

Date of issuance: October 1, 1998.

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

Amendment No.: 112.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 1, 1998 (63 FR 35994).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: December 31, 1997, as supplemented June 30, August 6, August 18, and August 27, 1998.

Brief description of amendments: The amendments revised TS 2.1 (Safety Limits), 2.2 (Limiting Safety System Settings), and 3/4.2.5 (Departure from Nucleate Boiling Parameters) by including alternate operating criteria to allow continued plant operation with a reduced measured reactor coolant system flow rate, if necessary.

Date of issuance: September 29, 1998.

Effective date: September 29, 1998.

Amendment Nos.: Unit 1—Amendment No. 97; Unit 2—Amendment No. 84.

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

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4325).

The additional information contained in the supplemental letters dated June 30, August 6, August 18, and August 27, 1998, were clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 29, 1998.

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.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio.

Date of application for amendment: April 18, 1997, as supplemented by letters dated October 10, 1997, and February 27 and September 8, 1998.

Brief description of amendment: This amendment revises TS Section 3/4.7.6, ''Plant Systems—Control Room Emergency Ventilation System,'' and the associated bases. Action statements have been added related to the availability of the station vent normal range radiation monitoring instrumentation. The bases have been modified consistent with these changes.

Date of issuance: October 5, 1998.

Effective date: October 5, 1998.

Amendment No.: 227.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 4, 1997 (62 FR 30646). The supplemental information submitted by letters dated October 10, 1997, and September 8, 1998, did not affect the proposed no significant hazards consideration. However, the supplemental letter dated February 27, 1998, included a new analysis of the issue of no significant hazards consideration. Based on this, the Commission issued a new proposed finding that the amendment involves no significant hazards consideration (63 FR 25117). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 5, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: August 8, 1997, as supplemented by letters dated December 16, 1997, January 20, 1998, March 4, 1998, March 17, 1998, June 29, 1998, and July 28, 1998.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.7–2 to specify that the lift setting tolerance for the main steam line safety valves is +3/−1 percent as-found and +/−1 percent as-left. The amendment also revised TS Table 2.2–1 to reduce the sensor error for the pressurizer pressure-high trip.

Date of issuance: October 2, 1998.

Effective date: October 2, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 128.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66144).

The December 16, 1997, January 20, 1998, March 4, 1998, March 17, 1998, June 29, 1998, and July 28, 1998, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Elmer Ellis Library, University of Missouri, Columbia, Missouri 65201–5149.

Wisconsin Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: May 14, 1998, supplemented July 3, August 27, and October 1, 1998.

Brief description of amendment: This amendment redefines the pressure boundary for Westinghouse mechanical hybrid expansion joints (HEJs) in sleeved steam generator tubes. TS 4.2 b, ''Steam Generator Tubes,'' is changed to incorporate a length criterion to allow tubes with degraded HEJ sleeves to remain in service if a minimum length of the HEJ is free of flaws.

Date of issuance: October 2, 1998.

Effective date: October 2, 1998.

Amendment No.: 139.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 3, 1998 (63 FR 30269).

The July 3, August 27, and October 1, 1998 submittals provided clarifying information within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001

Dated at Rockville, Maryland, this 14th day of October 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–28069 Filed 10–20–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Consolidated Guidance About Materials Licenses: Program-Specific Guidance About Exempt Distribution Licenses,'' Dated September 1998

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is announcing the completion and availability of NUREG–1556, Vol. 8, ''Consolidated Guidance about Materials Licenses: Program-Specific Guidance About Exempt Distribution Licenses,'' dated September 1998.

ADDRESSES: Copies of NUREG–1556, Vol. 8, may be obtained by writing to the Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, D.C. 20402–9328. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161. A copy of the document is also available for inspection and/or copying for a fee in the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, D.C. 20555–0001.

FOR FURTHER INFORMATION, CONTACT: Anthony Kirkwood, Mail Stop TWFN 8–F–5, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Telephone: 301–415–6140.

SUPPLEMENTARY INFORMATION:

On April 7, 1997 (62 FR 16630), NRC announced the availability of draft NUREG–1562, ''Standard Review Plan for Applications for Licenses to Distribute Byproduct Material to Persons Exempt from the Requirements for an NRC License,'' dated January 1997, and requested comments on it. The final version of NUREG–1562 will be published as NUREG–1556, Vol. 8, ''Consolidated Guidance about Materials Licenses: Program-Specific Guidance about Exempt Distribution Licenses,'' dated September 1998. In finalizing the NUREG report, the staff considered all the comments, including constructive suggestions, to improve the document.

This report is intended for use by applicants, licensees, and NRC staff, and will also be available to Agreement States. It combines, updates, and supersedes the guidance found in Draft NUREG–1562, ''Standard Review Plan for Applications for Licenses to Distribute Byproduct Material to Persons Exempt from the Requirements for an NRC License.'' When published, this final report should be used in applications for exempt distribution. NRC staff will use this final report in reviewing these applications.

Electronic Access

NUREG–1556, Volume 8, will be available electronically, approximately 1 month after the date of this notice, by visiting NRC's Home Page (http://www.nrc.gov) and choosing ''Nuclear Materials,'' and then ''NUREG–1556, Volume 8.''

Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Act of 1996, NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of the Office of Management and Budget.

Dated at Rockville, Maryland, this 14th day of October, 1998.

For the Nuclear Regulatory Commission.

Josephine M. Piccone,

Acting Director, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 98–28190 Filed 10–20–98; 8:45 am]

BILLING CODE 7590–01–P

POSTAL SERVICE BOARD OF GOVERNORS

Sunshine Act Meeting

TIMES AND DATES: 1:00 p.m., Monday, November 2, 1998; 8:30 a.m., Tuesday, November 3, 1998.

PLACE: Potomac, Maryland, at the William F. Bolger Center for Leadership Development, 9600 Newbridge Drive, Main Building in Room 200.

STATUS: November 2 (Closed); November 3 (Open).

MATTERS TO BE CONSIDERED:

Monday, November 2—1:00 p.m. (Closed)

1. International Mail Rates.
2. Compensation Issues.

Tuesday, November 3—8:30 a.m. (Open)

1. Minutes of the Previous Meeting, October 5–6, 1998.
2. Remarks of the Postmaster General/Chief Executive Officer.
3. Quarterly Report on Service Performance.
4. Capital Investments.
   a. Stamford, Connecticut, Springdale Station.
   b. Tray Management System Phase II—Additional Funding.
5. Briefing on the Diversity Study.
6. Tentative Agenda for the December 7–8, 1998, meeting in Washington, D.C.

CONTACT PERSON FOR MORE INFORMATION: Thomas J. Koerber, Secretary of the Board, U.S. Postal Service, 475 L'Enfant Plaza, SW, Washington, DC 20260–1000. Telephone (202) 268–4800.

Thomas J. Koerber,

Secretary.

[FR Doc. 98–28407 Filed 10–19–98; 8:45 am]

BILLING CODE 7710–12–M