Information regarding the time to be set aside for this purpose may be obtained by contacting Dr. Sher Bahadur prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Dr. Sher Bahadur if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Dr. Sher Bahadur (telephone 301–415–0138), between 7:30 a.m. and 4:15 p.m., EST.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the Internet at *http://www.nrc.gov/ACRSACNW.*

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., EST, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: January 15, 2002.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 02–1489 Filed 1–18–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on February 6, 2002, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows: Wednesday, February 6, 2002—1:30 p.m. until the conclusion of business.

The Subcommittee will discuss proposed ACRS activities and related matters. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff person named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting the cognizant ACRS staff person, Sam Duraiswamy (telephone: 301/415–7364) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any changes in schedule, etc., that may have occurred.

Dated: January 14, 2002.

Sher Bahadur,

Associate Director for Technical, Support, ACRS/ACNW.

[FR Doc. 02–1490 Filed 1–18–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 28, 2001 through January 10, 2002. The last biweekly notice was published on January 8, 2002 (67 FR 924).

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 Administrative 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's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By February 21, 2002, 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 NRC's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* 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: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, 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 PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendments request: November 9, 2001.

Description of amendments request: The amendments would revise Technical Specification (TS) 5.6.5b to add topical report CENPD–404–P–A, ''Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs,'' to the list of analytical methods used to determine core operating limits.

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 allows the use of methods required for the implementation of ZIRLO™ clad fuel rods in PVNGS [Palo Verde Nuclear Generating Station], Units 1, 2, and 3. The use of this methodology will not increase the probability of an accident because the plant systems will not be operated outside of design limits, no different equipment will be operated, and system interfaces will not change.

As ZIRLO™ material is introduced to the reactor, transition cores will exist in which ZIRLO'' and Zircaloy-4 clad fuel assemblies are co-resident. Fuel assemblies clad with each material will be evaluated based on the approved topical reports.

The use of this additional methodology will not increase the consequences of an accident because Limiting Conditions of Operation (LCOs) will continue to restrict operation to within the regions that provide acceptable results, and Reactor Protection System (RPS) trip setpoints will restrict plant transients so that the consequences of accidents will be acceptable. In addition, the consequences of the accidents will be calculated using NRC accepted methodologies.

The transition cores that will exist as ZIRLO'' clad fuel is introduced to the reactor will not increase the consequences of an accident. Operation within the LCOs and RPS setpoints will continue to restrict plant transients so that the consequences of accidents will be acceptable.

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 change does not add any new equipment, modify any interfaces with any existing equipment, alter the equipment's function, or change the method of operating the equipment. The proposed change does not alter plant conditions in a manner that could affect other plant components. The proposed change does not cause any existing equipment to become an accident initiator. The ZIRLO™ clad fuel rod design does not introduce features that could initiate an accident.

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

Safety Limits ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during steady state operation, normal operational transients and anticipated operational occurrences. All fuel limits and design criteria shall be met based on the approved methodologies defined in the topical reports. The RPS in combination with the LCOs will continue to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a violation of the Safety Limits. Therefore, the proposed changes will have no impact on the margins as defined in the Technical Specification bases.

The safety analyses determine the LCO settings and RPS setpoints that establish the initial conditions and trip setpoints, which ensure that the Design Basis Events (Postulated Accidents and Anticipated Operational Occurrences) analyzed in the Updated Final Safety Analysis Report (UFSAR) produce acceptable results. In addition, all fuel limits and design criteria shall be satisfied. The Design Basis Events that are impacted by the implementation of ZIRLO'' cladding will be analyzed using the NRC accepted methodology described in CENPD–404–P–A.

The change in the fuel rod cladding material and the use of the ECCS [emergency core cooling system] performance evaluation models, CENPD–132, Supplement 4–P, ''Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model'' and CENPD–137, Supplement 2–P, ''Calculative Methods for the CE Small Break LOCA Evaluation Model'' will not involve a reduction in the margin of safety because LCOs and Limiting Safety System Settings (LSSS) will be adjusted, if necessary, to maintain acceptable results for the impacted Design Basis Events.

Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three 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.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: December 13, 2001.

Description of amendments request: The amendments would add (1) the phrase ''or if open, capable of being closed'' to Limiting Condition for Operation 3.9.3, ''Containment Penetration,'' and (2) Surveillance Requirement (SR) 3.9.3.3 on verifying the capability to close the equipment hatch, if open. For refueling operations, the amendments would allow the equipment hatch to be open during core alterations and movement of irradiated fuel assemblies inside containment.

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 amendment[s] to Technical Specification (TS) 3.9.3 ''Containment Penetrations,'' would allow the equipment hatch to remain open, but capable of being closed, during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The position of the equipment hatch (open or closed) is not an initiator of any accident.

The fuel handing accident (FHA) contained in the *Updated Final Safety Analysis Report,* Revision 11, currently assumes that the entire airborne radioactivity reaching the containment is released to the outside environment. This results in a maximum offsite dose of 74.7 rem to the thyroid and 0.39 rem to the whole body. The calculated control room dose of 11.5 rem thyroid and 0.13 whole body are within the acceptance criteria specified in General Design Criteria 19 ''Control Room.''

Therefore, the proposed amendment request 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 amendment to TS 3.9.3 ''Containment Penetrations,'' allowing the equipment hatch to be open and capable of being closed does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, the proposed amendment request 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 amendment to TS 3.9.3 ''Containment Penetrations,'' allowing the equipment hatch to be open and capable of being closed remains bounded by previously determined radiological dose consequences for a FHA inside containment. The previously analyzed dose consequences were determined to be within the limits of 10 CFR [part] 100, ''Reactor Site Criteria,'' and they meet the acceptance criteria of NUREG–0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants Section 15.7.4 ''Radiological Consequences of Fuel Handling Accidents.'' Therefore, the proposed amendment request does not involve a significant reduction in a margin of safety. Additionally, a new surveillance will be added to verify the capability to close the equipment hatch, if open and CORE ALTERATIONS or movement of irradiated fuel assemblies are in progress within containment, at a frequency of seven days.

Based on the above, APS [the licensee] concludes that the activities associated with the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of ''no significant hazards consideration'' is justified.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three 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.

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

NRC Section Chief: Stephen Dembek.

Dominion Nuclear Connecticut Inc., et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No 3, New London County, Connecticut

Date of amendment request: October 1, 2001.

Description of amendment request: The proposed amendment would modify the Millstone Nuclear Power Station, Unit No. 3 (MP3) Technical Specifications (TSs) to increase the emergency diesel generator (EDG) allowed outage time (AOT), to perform a verification of the offsite circuits within 1 hour prior to or after entering the condition of either an inoperable offsite source or inoperable EDG, to revise the requirements for the pressurizer heaters and the pressurizer power operated relief and block valves, and to improve the format of the electrical power sources action requirements. The Bases of the affected TSs will be modified to address the proposed changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis, which is based on the representation made by the licensee in the October 1, 2001, application, is presented below:

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

The proposed TS changes to increase the EDG AOT, to perform a verification of the offsite circuits within 1 hour prior to or after entering the condition of either an inoperable offsite source or inoperable EDG, to revise the requirements for the pressurizer heaters and the pressurizer power operated relief and block valves, and to improve the format of the electrical power sources action requirements are not accident initiators nor will they impact the consequences of any previously evaluated accidents. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes to the TSs do not impact any system or component that could cause an accident nor do the proposed changes alter the plant configuration or require any unusual operator actions or alter the way any structure, system, or component functions. 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 TS changes to revise the requirements for the pressurizer heaters and the pressurizer power operated relief and block valves, the TS change to allow verification of offsite circuits within 1 hour prior to or after entering the condition of an inoperable offsite source or inoperable EDG, and the changes to improve the format of the electrical power sources action requirements do not change the TS-required safety limits or safety system settings; therefore, these additional changes will not result in a significant reduction in a margin of safety. The proposed TS changes to increase the EDG AOT do not affect any assumptions or inputs to the safety analyses. Unavailability of a single EDG due to maintenance or repair activities does not reduce the number of EDGs below the minimum required to mitigate all DBAs. Therefore, the proposed change will not result in a significant reduction in a margin of safety.

Based on this analysis, 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.

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina and Docket Nos. 50–413 and 50–414 Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: August 6, 2001.

Description of amendment request: The proposed amendments would decrease the McGuire Units 1 & 2 and Catawba Unit 1 Overtemperature Delta Temperature (OTΔT) Allowable Values and the McGuire Units 1 & 2 and Catawba Units 1 & 2 Overpower Delta Temperature (OPΔT) Allowable Values. OTΔT and OPΔT are trip functions provided in the reactor trip system to protect against departure from nucleate boiling and to ensure fuel integrity under all overpower conditions. The licensee states that due to changes in reload reactor core designs since the 1.0°F hot leg streaming uncertainty value was determined, it is now necessary to increase the uncertainty value to 1.21°F. Associated changes in the TS Table 3.3.1–1 OTΔT and OPΔT allowable values have been proposed. The licensee states that the decreases are in the conservative direction and will not adversely affect the steady-state or transient analyses documented in the Updated Final Safety Analysis Reports. In addition, the licensee has proposed two minor editorial changes for Catawba Units 1 & 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

First Standard

Would implementation of the changes proposed in this LAR involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This license amendment request [LAR] proposes to decrease the McGuire Units 1 & 2 and Catawba Unit 1 Overtemperature Delta Temperature (OTΔT) Allowable Values and the McGuire Units 1 & 2 and Catawba Units 1 and 2 Overpower Delta Temperature (OPΔT) Allowable Values. This decrease is in the conservative direction and will not adversely affect the steady-state or transient analyses documented in the Updated Final Safety Analysis Report. These changes have no impact on accident probabilities or consequences.

The proposed changes to the Catawba Table of Contents and Bases are solely administrative in nature and have no impact on any accidents.

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

Second Standard

Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any previously evaluated?

No. The proposed changes contained in this LAR only correct administrative errors and add conservative operability requirements which are consistent with the plants' existing licensing bases. No new or different kinds of accidents are being created.

3. Involve a significant reduction in margin of safety.

Third Standard

Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. These barriers are unaffected by the changes proposed in this LAR. Consequently, no margin of safety will be significantly impacted by this LAR.

Conclusion

Based upon the preceding evaluation, performed pursuant to 10CFR50.92, Duke Energy Corporation has concluded that implementation of this LAR at McGuire and Catawba Nuclear Station will not involve a significant hazards consideration. The changes proposed in this LAR make a conservative decrease in the McGuire Units 1 & 2 and Catawba Unit 1OTΔT Allowable Values and the McGuire Units 1 & 2 and Catawba Units 1 and 2 OPΔT Allowable Values and correct unrelated administrative errors. Following implementation of these proposed changes, McGuire and Catawba will continue to be operated in a conservative manner.

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.

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: December 20, 2001.

Description of amendment request: The proposed amendments would change the McGuire Technical Specifications (TS) to eliminate the revision number and dates from the list of topical reports that contain the analytical methods used to determine the core operating limits. This proposed change is consistent with the NRC approved Industry Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler TSTF–363, ''Revise Topical Report References in ITS 5.6.5 COLR''. Implementation of the changes proposed in this license amendment request will have no adverse impact on Duke's practices for controlling the methodologies used to develop the core operating limits for McGuire. The complete citations (i.e., report number, title, revision number, report date or NRC safety evaluation date, and any supplements) for each of the topical reports listed in TS 5.6.5 will be displayed as applicable in each station's Core Operating Limits Report (COLR). NRC review and approval of new or revised topical reports will continue to be obtained in the same manner. Changes to the COLRs will be controlled by 10 CFR 50.59.

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

Duke Energy Corporation has made the determination that this license amendment request (LAR) for McGuire Technical Specifications (TS) involves No Significant Hazards. This determination was made through the application of the standards established by 10CFR50.92. The three standards are discussed below.

1. Would implementation of the changes proposed in this LAR involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This LAR makes an administrative change to TS 5.6.5.b, Core Operating Limits Report (COLR), affecting a list of documents that are separately reviewed and approved by the NRC. The changes proposed to TS 5.6.5.b have no substantive impact on the McGuire licensing bases. Only NRC-approved methodologies will be used to generate the core operating limits. Based on these considerations, it has been determined that the proposed changes have no impact on any accident probabilities or consequences.

2. Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This LAR makes administrative changes that have no impact on any accident analyses.

3. Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. The analytical methodologies used to generate the core operating limits are unchanged by this LAR. As such, this LAR has no affect on margins of safety. Future changes to these methodologies will remain subject to NRC review and approval. Therefore, this proposed amendment does not involve a 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: December 20, 2001.

Description of amendment request: The proposed amendments would make changes in the Technical Specifications (TS) Bases Control Program to reflect changes in the NRC's regulations in 10 CFR 50.59 as noticed in the **Federal Register** on October 4, 1999. The proposed changes in the license amendment request are consistent with an NRC approved Technical Specifications Task Force Standard TS Traveler (TSTF–364).

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. Would implementation of the changes proposed in this LAR [license amendment request] involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This LAR makes an administrative change to the Technical Specifications [TS] made necessary as part of Duke's implementation of revised NRC regulations. The changes proposed to these TS have no substantive impact on the McGuire licensing bases, nor Duke's ability to conservatively evaluate changes to these licensing bases. Therefore, the proposed changes have no impact on any accident probabilities or consequences.

2. Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This LAR makes administrative changes that have no impact on any accident analyses.

3. Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. The proposed changes are administrative, an implementation of the revised 10CFR50.59 regulation. Implementation of the revised 10CFR50.59 regulation provides the necessary regulatory requirements to ensure that nuclear plants' margin of safety is preserved.

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.

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard J. Laufer, Acting.

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: October 16, 2001.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to incorporate changes resulting from the use of an alternate source term and the implementation of several plant modifications. The proposed TS changes include the following:

- The Penetration Room Ventilation System (PRVS) will be removed from TS because the PRVS will not be credited in licensing analyses that determine Control Room and off-site doses.
- The Spent Fuel Pool Ventilation System (SFPVS) will be removed from TS because the SFPVS will not be credited in licensing analyses that determine Control Room and off-site doses.
- During certain refueling operations, the containment air locks and/or the equipment hatch and penetrations providing direct access from the containment atmosphere to the outside atmosphere will be permitted to be unisolated under administrative controls. Additionally, the requirement to maintain an operable automatic isolation capability for the Reactor Building Purge system during refueling will be removed from TS.
- The allowable value for the Reactor Building leakage rate will be lowered from 0.25 w%/day to 0.20 w%/day.
- The requirement to measure Reactor Building leakage in excess of 50% of L_a to the penetration room will be removed from TS.
- The Ventilation Filter Testing Program (VFTP) will be revised to remove all references to the PRVS and SFPVS and their testing requirements.
- The VFTP acceptance criterion for the Control Room Ventilation System Booster Fan trains will be revised to require ≥ 97.5% radioactive methyl iodide removal.

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

The AST [Alternative Source Term] and those plant systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. The AST does not affect the design or operations of the facility. Rather, the AST is used to evaluate the consequences of a postulated accident. The implementation of the AST has been evaluated in the revisions to the analysis of the design basis accidents for Oconee Nuclear Station. Based on the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these events meet the acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183. Therefore, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The AST and those plant systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS changes and modifications do not significantly affect the mitigative function of these systems. Consequently, these systems do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The implementation of the AST, proposed changes to the TS and the implementation of the proposed modifications have been evaluated in the revisions to the analysis of the consequences of the design basis accidents for the Oconee Nuclear Station. Based on the results of these analyses, it has been demonstrated that with the requested changes the dose consequences of these events meet the acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183. Thus, the proposed amendment will 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.

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard J. Laufer, Acting.

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: December 20, 2001.

Description of amendment request: The proposed amendments would revise the licensing basis associated with the failure of non-Category I (non-seismic) piping.

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

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

No. The License Amendment Request (LAR) proposes to change the licensing basis for non-Category I (non-seismic) piping to assume a through-wall crack as the postulated piping failure. The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not affect any Chapter 15 accident analyses. Duke evaluated the effects of flooding caused by a leak from a crack size calculated using the SRP [Standard Review Plan] guidelines in the 16-inch HPSW [High Pressure Service Water] header. This evaluation concluded that for the bounding case, the effects of flooding can be mitigated without adversely affecting safety-related equipment. At least an hour is available from detection for operator action to isolate the leak. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

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

No. The License Amendment Request (LAR) changes the licensing basis associated with non-seismic moderate energy line breaks. The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Therefore, the possibility of a new or different kind of accident from any kind of accident previously evaluated is not created.

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

No. The License Amendment Request (LAR) changes the licensing basis associated with non-seismic moderate energy line breaks. The impact of flooding from a seismically induced crack in non-seismic moderate energy piping has been evaluated. Adequate time exists for operator action to isolate flooding sources prior to adversely affecting safety-related equipment required for safe shutdown. As such, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard J. Laufer, Acting.

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: December 20, 2001.

Description of amendment request: The proposed amendments would make changes in the Technical Specifications (TS) to eliminate the use of the term ''unreviewed safety question.'' The change is proposed by the licensee to reflect changes in the NRC's regulations in 10 CFR 50.59 as noticed in the **Federal Register** on October 4, 1999. The proposed changes in the license amendment request are consistent with an NRC approved Technical Specifications Task Force Standard TS Traveler (TSTF–364).

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. Would implementation of the changes proposed in this LAR [license amendment request] involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This LAR makes an administrative change to the Technical Specifications [TS] made necessary as part of Duke's implementation of revised NRC regulations. The changes proposed to these TS have no substantive impact on the Oconee licensing bases, nor Duke's ability to conservatively evaluate changes to these licensing bases. Therefore, the proposed changes have no impact on any accident probabilities or consequences.

2. Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This LAR makes administrative changes that have no impact on any accident analyses.

3. Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. The proposed changes are administrative, an implementation of the revised 10 CFR 50.59 regulation. Implementation of the revised 10 CFR 50.59 regulation provides the necessary regulatory requirements to ensure that nuclear plants' margin of safety is preserved.

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.

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard J. Laufer, Acting.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick (JAF) Nuclear Power Plant, Oswego County, New York

Date of amendment request: November 2, 2001.

Description of amendment request: The proposed change to the JAF Nuclear Power Plant Technical Specifications establishes a combined leakage rate limit for the sum of the four main steam line leakage rates that is equal to four times the current main steam line valve (MSIV) leakage rate limit.

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:

Operation of the JAF plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR [Code of Federal Regulations] 50.92 since it would not: Involve an increase in the probability or consequences of an accident previously evaluated.

The proposed amendment does not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the FitzPatrick Updated Final Safety Analysis Report (UFSAR).

The proposed amendment results in no change in radiological consequences of the design basis LOCA [loss-of-coolant accident] as currently analyzed for the FitzPatrick Plant. These analyses were calculated assuming a combined total MSIV leakage at accident pressure for determining acceptance to the regulatory limits for the offsite, control room, and Technical Support Center (TSC) radiation doses as contained in 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19 [General Design Criteria]. The proposed change does not compromise existing radiological equipment qualification, since the combined total MSIV leakage rate has been factored into existing equipment qualification analyses for 10 CFR 50.49.

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

The proposed change does not modify the MSIVs or any other plant system or structure associated with this amendment and therefore, will not affect their capability to perform their design function. The combined total main steam line leakage rate is included in the current radiological analyses for the assessment of radiation exposure following an accident.

This proposal changes the allowable leakage rate from a per valve limit to a total combined leakage rate limit for all four main steam lines but does not change the cumulative limit. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.

Involve a significant reduction in a margin of safety.

The leakage rate limit specified for the MSIVs is used to quantify the maximum amount of bypass leakage assumed in the LOCA radiological analysis. Results of the analysis are evaluated against the dose guidelines contained in GDC 19 and 10 CFR 100. The margin of safety in this context is considered to be the difference between the calculated dose exposures and the guidelines provided by the GDC 19 and 10 CFR 100. Therefore, since the proposed combined total main steam line leakage rate limit is unchanged from the assumed maximum leakage rate for MSIVs, for the purpose of calculating potential radiation dose, the margin of safety is not affected because the postulated radiation doses remain the same.

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.

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: L. Raghavan, Acting.

Exelon Generation Company, LLC (Exelon), Docket No. 50–352, Limerick Generating Station (LGS), Unit 1, Montgomery County, Pennsylvania

Date of amendment request: December 21, 2001.

Description of amendment request: Exelon proposed changes that would revise Technical Specification (TS) 2.1 to incorporate revised Safety Limit Minimum Critical Power Ratios (SLMCPRs) due to the cycle-specific analysis performed by Global Nuclear Fuel for LGS Unit 1, Cycle 10, which will include the use of the GE–14 fuel product line.

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

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

The derivation of the cycle-specific SLMCPRs for incorporation into the TSs, and its use to determine cycle-specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR-II), and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000, which incorporates Amendment No. 25. Amendment No. 25 provides the methodology for determining the cycle-specific MCPR safety limits that replaces the former generic fuel type dependent values. Amendment No. 25 was approved by the NRC (Nuclear Regulatory Commission) in a March 11, 1999, safety evaluation.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling. The probability of fuel damage will not increase as a result of this change. Likewise, the consequences of accidents previously evaluated are not affected by the revised SLMCPRs values. Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR is a TS numerical value, calculated to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core if the limit is not violated. SLMCPRs are based on a calculation using an NRC-approved methodology discussed in NEDE–24011–P–A–14 (GESTAR-II), and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000. The SLMCPR is not an accident initiator, and its revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The new SLMCPRs are calculated using NRC-approved methodology discussed in NEDE–24011–P–A–14 (GESTAR-II), and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000. This methodology uses the same standards and margins that were used in the former generic fuel type methodology. Therefore, the proposed TS change will not involve a significant reduction in a margin of safety previously approved by the NRC.

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.

Attorney for licensee: Mr. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Clifford.

National Aeronautics and Space Administration (NASA), Docket No. 50–30, Plum Brook Reactor Facility (PBRF), Sandusky, Ohio

Date of amendment request: December 20, 1999, as supplemented on March 26, November 19, and December 20, 2001.

Description of amendment request: The proposed amendment would allow decommissioning of the Plum Brook Test Reactor Facility.

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 approval of the PBRF Decommissioning Plan involve a significant increase in the probability or consequences of an accident previously evaluated?

All nuclear fuel has been removed from the PBRF site. Radioactive inventories at the PBRF are very small compared to those in operating reactors (both power and non-power) and in various kinds of fuel cycle facilities subject to NRC regulation. Analyses indicate that decommissioning activities would not involve a significant increase in the probability or consequences of an accident previously evaluated in the current Final Hazards Summary for the NASA Plum Brook Reactor Facility.

Summary: NASA considers that the approval of the Decommissioning Plan does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed approval of the PBRF Decommissioning Plan create the possibility of a new or different kind of accident from any accident previously evaluated?

The current Final Hazards Summary for the NASA Plum Brook Reactor Facility evaluated those cause-and-effect accidents related to external events and loss/failure of reactor support systems that would result in the dispersal of fission products and radioactive materials to the environment. Due to the combined absence of fuel at the PBRF site and the non-operational condition of reactor support systems, NASA has determined that decommissioning activities, as described in the Decommissioning Plan, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Summary: NASA considers that the approval of the Decommissioning Plan does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Current Technical Specifications adequately restrain the scope and nature of decommissioning activities to loose equipment removal and preparations for dismantlement. Approval of the proposed Decommissioning Plan provides for additional controls prior to commencement of dismantlement activities, thereby achieving a greater margin of safety.

Summary: NASA considers that the approval of the Decommissioning Plan does not involve a significant reduction in a margin of safety.

Based on the above evaluations, NASA concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding by the NRC of no significant hazards consideration is justified.

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.

Attorney for the Licensee: J. William Sikora, Esquire, 21000 Brookpark Road, Mail Stop 500–118, Cleveland, OH 44135.

NRC Section Chief: Patrick M. Madden.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York

Date of amendment request: October 5, 2001; as revised on January 4, 2002. This notice supersedes a previous notice (66 FR 55020) published on October 31, 2001, which was based upon the licensee's application dated October 5, 2001.

Description of amendment request: The licensee proposed to amend the Technical Specifications (TSs) to change the licensing basis requirement for establishing containment hydrogen monitoring ''within 30 minutes'' to ''within 3 hours'' of initiating emergency core cooling following a loss-of-coolant accident (LOCA). The January 4, 2002, revision reduces the proposed delay from 3 hours to 90 minutes.

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 operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Updated Safety Analysis Report (USAR) Chapter 15 accident analyses do not require or take credit for hydrogen monitoring to be established shortly after a loss of coolant accident (LOCA). Post-LOCA hydrogen production occurs over a long period of time, and an extension from ''30 minutes'' to ''90 minutes'' for establishing hydrogen monitoring will have a positive impact on the ability of the operators to concentrate on their more immediate actions while having no negative impact on containment integrity or the long-term assessment efforts. Therefore, the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) The operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Control room operators use the containment hydrogen monitors following a LOCA to establish hydrogen control measures should it become necessary. The proposed license amendment would not eliminate the requirement to establish hydrogen monitoring, but would allow it to be delayed until those actions required to mitigate the accident and verify proper operation of essential safety equipment have been completed. The proposed extension maintains the requirement to establish hydrogen monitoring well before calculated conditions inside the containment indicate any need to initiate hydrogen control measures. Therefore, the proposed license amendment will not create a new or different kind of accident from any accident previously evaluated.

(3) The operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The need to establish hydrogen control measures will not be present within the first 90 minutes following a LOCA since there will not be significant hydrogen accumulation. By extending the time allowed to establish containment hydrogen monitoring, the operators can remain focused on the actions necessary to mitigate the accident before directing their attention to hydrogen control measures and other long-term actions. The proposed extension maintains the requirement to establish hydrogen monitoring well before calculated conditions inside the containment indicate any need to initiate hydrogen control measures. Therefore, the proposed license amendment 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.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington DC 20005–3502.

NRC Section Chief: L. Raghavan, Acting.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: December 21, 2001.

Description of amendment request: The proposed changes would revise Technical Specification (TS) Surveillance Requirements (SR) 4.3.1.2 and 4.3.2.2 to allow verification in place of demonstration of response time associated with certain pressure sensors, differential pressure sensors, process protection racks, nuclear instrumentation, and logic systems.

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

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

The proposed changes to TS 3/4.3.1 and TS 3/4.3.2 do not result in a condition where the design, material, and construction standards that were applicable prior to the proposed changes are altered. The same Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) instrumentation is being used; the response time allocations/modeling assumptions in the Seabrook Station UFSAR [updated final safety analysis report] analyses are still the same; only the method of verifying time response is changed. The proposed change will not modify any system interface and will not increase the probability or consequences of an accident previously evaluated since these events are independent of this change.

The proposed changes do not affect the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Seabrook Station UFSAR. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures.

Therefore, it is concluded that these proposed revisions to TS 3/4.3.1 and TS 3/4.3.2 do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed changes to TS 3/4.3.1 and TS 3/4.3.2 do not alter the performance of the pressure and differential pressure sensors used in the plant protection systems, nor do the proposed changes alter the performance of the Process Protection racks, Nuclear Instrumentation, and Logic Systems used in the plant protection systems. The sensors will still have their response time verified by test before placing the sensor in operational service and after any maintenance that could affect response time; and the plant protection systems will still have response time verified by test before being placed in operational service.

For the pressure and differential pressure sensors; and for the Process Protection racks, the Nuclear Instrumentation, and the Logic Systems; changing the method of periodically verifying instrument response from time response testing to calibration and channel checks (assuring equipment operability) will not create any new accident initiators or scenarios.

The periodic calibration of the pressure and differential pressure sensors will detect significant degradation in the sensor response characteristic.

The periodic calibration of the Process Protection racks, the Nuclear Instrumentation, and the Logic Systems will continue to be used to detect significant degradation that could cause the response time characteristic to exceed the total allowance. The total time response allowance for each function bounds degradation that cannot be detected by the periodic surveillance.

Thus, these proposed revisions to TS 3/4.3.1 and TS 3/4.3.2 do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to TS 3/4.3.1 and TS 3/4.3.2 do not affect the total system response time assumed in the Seabrook Station UFSAR analyses. The periodic system response time verification method for the pressure and differential pressure transmitters; and the periodic system response time verification method for the Process Protection racks, the Nuclear Instrumentation, and the Logic Systems, is modified to allow use of actual test data or engineering data. The method of verification will continue to provide assurance that the total system response is within that defined in Seabrook Station UFSAR analyses.

For the pressure and differential pressure sensors, calibration tests will detect degradation, which might significantly affect sensor response time.

For the Process Protection racks, the Nuclear Instrumentation, and the Logic Systems calibration tests will continue to be performed which would detect significant degradation which might cause the response time to exceed the total allowance. The total time response allowance for each function bounds degradation that cannot be detected by the periodic surveillance.

Thus, it is concluded that these proposed revisions to TS 3/4.3.1 and TS 3/4.3.2 do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 14, 2001.

Description of amendment request: The proposed amendment will remove requirements for having the equipment hatch closed with four (4) bolts and one door of the personnel access lock (PAL) closed during core alterations and refueling operations for the Fort Calhoun Station (FCS). The technical specification (TS) for other containment penetrations will be modified to delete the requirement to be closed by an operable ventilation isolation actuation signal during core alterations and refueling operations. The proposed amendment will modify requirements for radiation monitors during core alterations and refueling operations. The TS Bases that are affected by the changes described above will be modified.

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 changes to FCS TS modify requirements to have containment closure in place during core alterations and refueling operations in containment. These TS changes do not impact operation of other equipment or systems important to safety. The proposed TS changes reflect the parameters used in the radiological consequence calculations described in Section 5.0 of this license amendment request.

The proposed change to TS 2.8.2(1) will be to delete the requirement for having equipment hatch closed and held in place by at least four (4) bolts and the requirement to have at least one door in the PAL closed. The requirements for containment penetration isolation via an operable VIAS [ventilation isolation actuation signal] have been deleted with these proposed changes. Administrative controls will be put in place instead for ''defense in depth'' action in regards to containment penetrations. These administrative controls include:

a. The Equipment Hatch Enclosure (Room 66) doors or the equipment hatch and one door in the PAL shall be capable of being closed in less than one hour of a FHA [fuel handling accident].

b. The Equipment Hatch Enclosure (Room 66) doors or the equipment hatch and one door in the PAL shall not be obstructed unless capability for rapid removal of obstructions is provided (such as quick disconnects for hoses).

c. Penetrations providing direct access from the containment atmosphere to the outside atmosphere shall be capable of being closed on one side in less than one hour of a FHA.

d. An individual or individuals shall be designated and available during core alterations and refueling operations, capable of closing the Equipment Hatch Enclosure (Room 66) doors or the equipment hatch, one door in the PAL, and penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

In addition, allowance will be granted to have penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during core alterations and refueling operations. These proposed changes are based on a re-analysis that was performed with respect to radiological consequences. The FHA re-analysis (Reference 10.1 [in the December 14, 2001, submittal]) was performed in accordance with current accepted methodology, and consequences were expressed in TEDE [total effective dose equivalent] dose.

The proposed change to TS 2.8.2(3) will delete the requirement for two gaseous radiation monitors being operable and supplied by independent power supplies. Instead, only one gaseous radiation monitor is required to be operable. VIAS actuation upon radiation monitor alert is not credited in the FHA re-analysis. VIAS actuation for containment purge or other penetration isolation is not credited.

The current methodology as described in 10 CFR 50.67 specifies dose acceptance criteria in terms of TEDE dose. The revised FHA analysis results as discussed in Section 5.0 meet the applicable TEDE dose acceptance criteria (specified also in RG [regulatory guide] 1.183) for AST [alternative source term]. The most current FHA analysis does not credit containment integrity and, hence, is conservative in that aspect. These administrative controls proposed as stated above ensure that in the event of a FHA in containment (even though the containment fission product control function is not required to meet dose consequence criteria) that the Equipment Hatch Enclosure (Room 66) doors or the equipment hatch, one PAL door, and other pathways can be promptly closed.

Currently the equipment hatch is closed with four (4) bolts, at least one PAL door closed, and other penetrations either are closed or capable of being closed on VIAS during core alterations and refueling operations to prevent the escape of radioactive material in the event of a FHA in containment. Whether the equipment hatch or other penetrations are open or closed during core alterations and refueling operations has no effect on the probability of any accident previously evaluated.

Based on the TS changes approved in Reference 10.1, the changes being proposed in this amendment request will not affect assumptions contained in other plant safety analyses (Updated Safety Analysis Report) or the physical design of the plant, nor do they affect other TS that preserve safety assumptions.

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

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

The current FHA analysis (Reference 10.1) assumes that all the iodine and noble gases become airborne, escape, and reach the site boundary and low population zone with no credit for filtration, containment closure, or deposition. Since the proposed changes do not involve the addition or modification of equipment nor alter the design of plant systems, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes proposed do not change how design basis accident (DBA) events were postulated nor do the changes themselves initiate a new kind of accident or failure mode with a unique set of conditions (proposed administrative controls). The FHA analysis documented in Reference 10.1 was performed consistent with 10 CFR 50.67 and RG 1.183. Not crediting filtration systems for EAB/LPZ dose consequences and only crediting natural forces is conservative from the aspect of dose consequences.

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

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

The implementation of the proposed changes does not reduce the margin of safety as defined in the alternate source term design basis site boundary and control room dose analyses (Reference 10.1). The radiological analyses results, with the proposed changes, remain within the regulatory acceptance criteria (10 CFR 50.67) utilizing the TEDE dose acceptance criteria directed in RG 1.183. These criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting these limits demonstrates adequate protection of public health and safety. An acceptable margin of safety is inherent in these licensing limits.

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

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 14, 2001.

Description of amendment request: The licensee proposes to (1) revise Technical Specifications 3.7(2)d and 3.7(4) to allow the tests to be performed on a refueling frequency outside of a refueling outage, and (2) correct the docket concerning inconsistencies in the 1973 Fort Calhoun Station (FCS) Safety Evaluation Report (SER) associated with the 13.8 kV transmission line capability.

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 changes to Technical Specifications Sections 3.7(2)d and 3.7(4) only provide greater flexibility in the time of testing. The periodicity remains the same, i.e., refueling frequency. There are no physical alterations proposed or being made to the D.C. emergency transfer switches or the 13.8 kV–480 V service. The proposed changes continue to address and comply with the regulatory requirements as described in Fort Calhoun Station Responses to 70 Criteria, Reference 9.2. The proposed changes will continue to assure that the D.C. emergency transfer switches and the 13.8 kV–480 V service will perform their design function. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not result in any physical alterations to the D.C. emergency transfer switches or 13.8 kV–480 V service, or any plant configuration, systems, equipment, or operational characteristics. There will be no change in operating modes or safety limits. With the proposed changes, the technical specifications retain requirements for operability and functionality on a refueling frequency. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes provide flexibility in the time of performance of the required surveillance tests. The proposed changes will not alter any physical or operational characteristics of the D.C. emergency transfer switches or the 13.8 kV–480 V service. The proposed surveillance requirements will continue to assure that the design functions are met. Therefore, the proposed 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.

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 14, 2001.

Description of amendment request: The licensee has proposed to change Technical Specification (TS) 2.10.4, to decrease the minimum required reactor coolant system (RCS) flow rate from 206,000 gallons per minute (gpm) to 202,500 gpm.

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 amendment to the RCS flow rate is the same as the indicated RCS flow rate prior to the TS Amendment 193 (Reference 10.1 [in the December 14, 2001, submittal]). The plant was operated with the same RCS flow rate as the proposed value prior to Amendment 193. Chapter 14 events and design basis accidents were analyzed with the RCS flow rate of 202,500 gpm using NRC approved methodology.

In 1999 Fort Calhoun Station was granted TS Amendment 193 to increase the minimum indicated RCS flow rate to 206,000 gpm as a result of the removal of the steam generator orifice plates. Transient and thermal hydraulic analyses were performed using the amended RCS flow rate to verify that the minimum departure from nucleate boiling ratio (MDNBR) does not fall below the limiting value that supports the DNB specified acceptable fuel design limits.

The FRA–ANP analysis confirms that the proposed reduction in RCS flow rate does not degrade the margin to the mechanical fuel design limits and that the fuel design criteria continue to be met.

In view of the above confirmation, 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 change to the RCS flow rate is not new since the plant was operating with the same value prior to TS Amendment 193. The proposed revision does not change any equipment required to mitigate the consequences of an accident. OPPD will continue to analyze all applicable USAR Chapter 14 events and design basis accidents as part of the reload analyses to establish the safety margin to the mechanical fuel design limits and confirm that all the fuel design criteria continue to be met. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The decreased RCS flow rate has been analyzed for thermal hydraulic effects on the reactor core. The analysis has confirmed that the proposed amendment does not degrade the margin to the mechanical fuel design limits and meets the fuel design criteria. The RCS flow rate surveillance requirements will continue to assure that the design functions are met. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 14, 2001.

Description of amendment request: The proposed amendment deletes technical specification (TS) Figures 2–1A (Reactor Coolant System (RCS) Pressure—Temperature Limits for Heatup) and 2–1B (RCS Pressure—Temperature Limits for Cooldown) and replaces them with the single TS Figure 2–1. Additionally, the licensee proposes to change the lowest service temperature from 182°F to 164°F to be in compliance with Reference 4, American Society of Mechanical Engineers (ASME) Section III, NB–2332 and the basis for the minimum boltup temperature to be in compliance with Reference 5, ASME Section XI, Appendix G. The Basis section for Technical Specification 2.1.2 is being updated to reflect the use of ASME Code Case N–640 and Westinghouse Electric Company/Combustion Engineering's (W/CE) pressure temperature (P–T) limit curve methodology as applicable. Finally, based on the replacement of Figures 2–1A and 2–1B with the single Figure 2–1, the following TS are required to be changed: 2.1.1(8), 2.1.2, 2.1.2(1), 2.1.2(2), 2.1.2(6), 2.1.2(6)(a), 2.1.2(6)(c), 2.1.2(6)(d), and 2.1.6(4) as they reference the deleted curves.

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 changes will not increase the probability or consequence of any accident for the following reasons:

(1) TS Figure 2–1 is proposed to incorporate the use of ASME Code Case N–640, which has been approved by the NRC as being acceptable for the development of P–T curves. Additionally, it is being updated for operation to higher neutron fluence values for use in the ART [adjacent reference temperature] calculations.

(2) Reducing the lowest service temperature is in compliance with Reference 10.9, Section III, NB–2332.

(3) The shift in the basis for minimum boltup temperature is in compliance with 10 CFR 50 Appendix G.

(4) Updating the fluence and EFPY [effective full power years] applicability is in compliance with Regulatory Guide 1.99, Revision 2.

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

The proposed revision does not change any equipment required to mitigate the consequences of an accident. The continued use of the same Technical Specification administrative controls prevents the possibility of a new or different kind of accident. Since the proposed changes do not involve the addition or modification of equipment nor alter the design of plant systems, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes proposed do not change how design basis accident events are postulated nor do the changes themselves initiate a new kind of accident or failure mode with a unique set of conditions (proposed administrative controls). Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed TS Figure 2–1 does not constitute a significant reduction in the margin of safety due to the following:

(1) The current LTOP [low temperature overpressure protection] analysis setpoints are bounding and applicable to this TS Figure.

(2) The use of ASME Code Case N–640 has been approved by the NRC as acceptable for the development of P–T limit curves. W/CE's P–T limit curve methodology has been approved for the development of P–T curves.

(3) The reduction in lowest service temperature is in compliance with Reference 10.9, Section III, NB–2332.

(4) The shift in the basis of the minimum boltup temperature from NDTT to RT_{NDT} is in compliance with Reference 10.4, Section XI, Appendix G.

(5) Updating the fluence and EFPY applicability of the TS Figure 2–1 to maintain validity is in compliance with Regulatory Guide 1.99, Revision 2.

The P–T curve results, with the proposed changes, remain within the regulatory acceptance criteria utilizing W/CE methodology and ASME Code Case N–640. These criteria, 10 CFR 50.36(c)(2), have been developed for application to analyses performed for long term operation of reactor vessels. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting these limits demonstrates adequate protection of public health and safety. An acceptable margin of safety is inherent in these licensing limits. Therefore, the proposed 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.

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 14, 2001.

Description of amendment request: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * *up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to SR 3.0.3 ''A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated December 14, 2001.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: Richard J. Laufer, Acting.

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 2, 2001.

Description of amendment request: Proposed amendments would revise Technical Specification 3/4.6.1.6, ''Containment Structural Integrity,'' and replace it with reference to containment Post-Tensioning System Surveillance Program.

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:

Pursuant to 10 CFR 50.91, this analysis provides a determination that the proposed changes to the Technical Specifications do not involve any significant hazards consideration as defined in 10 CFR 50.92.

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

Response: No.

The proposed changes revise the surveillance requirements for the containment post-tensioning inservice inspection program as required by 10 CFR 50.55a(b)(2)(vi) and 10 CFR 50.55a(b)(2)(viii). The revised requirements do not affect the function of the containment post-tensioning system components. The post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors.

The proposed changes do not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. The function of the containment post-tensioning system components are not altered by this change. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

Criterion 3: Does the proposed change involve a significant reduction in the margin of safety?

Response: No.

The proposed change does not impact the margin of safety included in the design pressure compared to the peak calculated pressure because the proposed activity does not alter, in any way, the available force provided by the tendons. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

Based on the evaluation provided above, the proposed changes do not involve a significant hazards consideration under 10 CFR 50.92(c), and will not have a significant effect on the safe operation of the plant. Therefore, there is reasonable assurance that operation of the South Texas Project in accordance with the proposed revised Technical Specifications will not endanger the public health and 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.

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

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: October 22, 2001.

Description of amendments request: The proposed amendments would revise the Technical Specification Limiting Condition for Operation for Containment Penetrations to allow the equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies within containment.

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:

STPNOC [South Texas Project Nuclear Operating Company] has evaluated whether the proposed amendment involves a significant hazards consideration by focusing on the three standards set forth in 10 CFR 50.92 as discussed below:

(1) Will operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes will allow the equipment hatch to be open during core alterations and movement of irradiated fuel assemblies inside containment. The status of the equipment hatch during refueling operations has no affect on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Since the consequences of a fuel handling accident inside containment with an open equipment hatch are bounded by the current analysis described in the UFSAR [Updated Final Safety Analysis Report] and the probability of an accident is not affected by the status of the equipment hatch, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Will operation of the facility in accordance with the proposed amendment create the possibility of a new different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No new equipment will be added and no new limiting single failures will be created. The plant will continue to be operated within the envelope of the existing safety analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident previously evaluated.

(3) Will operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The previously determined radiological dose consequences for a fuel handling accident inside containment with the personnel airlock doors open remain bounding for the proposed changes. These previously determined dose consequences were determined to be well within the limits of 10 CFR 100 and they meet the acceptance criteria of SRP section 15.7.4 and GDC 19. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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 3, 2001.

Description of amendments request: The proposed amendments would revise the Technical Specifications (TSs) for the Auxiliary Feedwater (AFW) System to provide consistent allowed outage times (AOT) and required actions for any inoperable motor driven AFW pump(s). The AOT for one inoperable motor drive AFW pump is also proposed to be extended from 72 hours to 28 days based on a risk-informed approach.

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:

STPNOC [South Texas Project Nuclear Operating Company] has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92 as discussed below:

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

Response: No.

The proposed TS change reflects the STP four-train AFWS design in the required actions and AOTs. No actual plant equipment or accident analyses will be affected by the proposed change. Therefore, the proposed AOT change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The administrative change of deleting the words ''At least'' clarifies that there are only four AFW pumps in the design. The administrative change involves no increase in the probability or consequences of an accident.

If all four AFW trains are inoperable in Mode 1, 2, or 3, the unit is in a seriously degraded condition with only limited means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to operable status. Required Action (d) is modified by adding a sentence indicating that all required mode changes or power reductions are suspended until one AFW train is restored to operable status. This statement reflects the same sentence for the case of all AFW trains being inoperable in NUREG–1431, TS 3.7.5. In this case, LCO [limiting condition for operation] 3.0.3 is not applicable because it could force the unit into a less safe condition. Therefore, the addition of the sentence to Action (d) involves no increase in the probability or consequences of an accident.

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

Response: No.

The proposed TS change reflects the STP four-train AFWS design in the required actions and AOTs. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, the proposed AOT change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The administrative change of deleting the words ''At least'' clarifies that there are only four AFW pumps in the design. The change does not create the possibility of any accident.

Required Action (d) is modified by adding a sentence indicating that all required mode changes or power reductions are suspended until one AFW train is restored to operable status. This statement reflects the same sentence for the case of all AFW trains being inoperable in NUREG–1431, TS 3.7.5. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition. Therefore, the addition of the sentence to Action (d) does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation dose to the public.

The proposed TS change reflects the STP four-train AFWS design in the required actions and AOTs. No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed change will not relax any criteria used to establish safety limits, will not relax any safety systems settings, and will not relax the bases for any limiting conditions of operation. Therefore, the proposed AOT change does not involve a significant reduction in a margin of safety.

The administrative change of deleting the words ''At least'' clarifies that there are only four AFW pumps in the design. The change does not involve any reduction in a margin of safety.

Required Action (d) is modified by adding a sentence indicating that all required mode changes or power reductions are suspended until one AFW train is restored to operable status. This statement reflects the same sentence for the case of all AFW trains being inoperable in NUREG–1431, TS 3.7.5. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition. Therefore, the addition of the sentence to Action (d) does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment involves no significant hazards consideration under the standards set forth in 10 CFR 50.92, and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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.

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

TXU Generation Company LP, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: December 18, 2001.

Brief description of amendments: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated December 18, 2001.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of application request: December 6, 2001.

Description of amendment request: The amendment would revise the following Technical Specifications (TSs): (1) TS 3.3.6, ''Containment Purge Isolation Instrumentation;'' (2) TS 3.3.7, ''Control Room Emergency Ventilation

System (CREVS) Instrumentation;'' and (3) TS 3.9.4, ''Containment Penetrations.'' The revisions to TS 3.3.6 would alter Condition C of the Actions for the Limiting Condition for Operation (LCO), and delete footnotes (a) and (b) from applicable Modes for the automatic actuation logic and actuation relays function and the containment purge exhaust radiation gaseous function in Table 3.3.6–1. The revisions to TS 3.3.7 would add (1) Surveillance Requirement (SR) 3.3.7.6, (2) footnote (c) to Table 3.3.7–1, (3) the fuel building exhaust radiation gaseous function to the table, and (4) footnote (c), 2 trains, and SR 3.3.7.6 to the applicable Modes, required channels, and surveillance requirements columns for the automatic actuation logic and actuation relays and control room radiation control room air intakes functions in Table 3.3.7–1. The revisions to TS 3.9.4 are to add the phrase ''or if open, capable of being closed'' to item a on the equipment hatch in LCO 3.9.4, delete the word ''closed'' from item b on the emergency air lock in LCO 3.9.4, add SR 3.9.4.2 on verifying the capability to install the equipment hatch when it is open, and renumber the existing SR 3.9.4.2 to SR 3.9.4.3. The revisions to the TSs are to allow the equipment hatch and the emergency air lock to be open in refueling outages during core alterations and/or movement of irradiated fuel within containment. The revisions to TSs 3.3.6 and 3.3.7 are to eliminate the requirement for automatic actuation of containment purge isolation during core alterations and/or during movement of irradiated fuel to allow the containment purge system to remain in operation during refueling when the equipment hatch is open, and to add a new surveillance to response time test the channels for the control room radiation monitor detectors, respectively.

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

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes will allow the equipment hatch [or the emergency air lock] to be open during CORE ALTERATIONS and movement of irradiated fuel assemblies inside containment. The status of the equipment hatch or the emergency air lock during refueling operations has no affect on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Since the consequences of a fuel handling accident inside containment with an open equipment hatch [or open emergency air lock] are bounded by the current analysis described in the FSAR [Final Safety Analysis Report] and the probability of an accident is not affected by the status of the equipment hatch [or the emergency air lock], the proposed change[s do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No new equipment will be added and no new limiting single failures will be created. The plant will continue to be operated within the envelope of the existing safety analysis.

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

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

Response: No.

The previously determined radiological dose consequences for a fuel handling accident inside containment with the [equipment hatch or emergency] air lock doors open remain bounding for the proposed changes. Those previously determined dose consequences were determined to be well within the limits of 10 CFR 100 and they meet the acceptance criteria of SRP [Standard Review Plan] section 15.7.4 and GDC 19 [for exposure of control room operators].

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

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

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: September 21, 2001.

Brief description of amendments: Will revise the Technical Specifications to allow Sequoyah to insert tritium-producing burnable absorber rods into the reactor core.

*Date of publication of individual notice in the **Federal Register:*** December 17, 2001 (66 FR 65000).

Expiration date of individual notice: January 16, 2002.

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

Date of application for amendments: April 20, 2001.

Brief description of amendments: Will revise the Final Safety Analysis Report to reflect a change in the spent fuel pool cooling analysis methodology.

*Date of publication of individual notice in the **Federal Register:*** December 17, 2001 (66 FR 64998).

Expiration date of individual notice: January 16, 2002.

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

Date of application for amendments: August 20, 2001.

Brief description of amendments: Will revise the Technical Specifications to allow Watts Bar to insert tritium-producing burnable absorber rods into the reactor core.

*Date of publication of individual notice in the **Federal Register:*** December 17, 2001 (66 FR 65005).

Expiration date of individual notice: January 16, 2002.

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: May 31, 2001, as supplemented August 1, 2001, and September 26, 2001.

Brief description of amendment: The amendment approves a change to the Technical Specifications and Bases associated with the operability of A.C. electrical power sources to increase the allowed outage time (AOT) for one inoperable emergency diesel generator (EDG) from 72 hours to 14 days. This change to the AOT allows the performance of various EDG maintenance and repair activities during plant operation.

Date of issuance: January 4, 2002.

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

Amendment No.: 261.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in *Federal Register:* August 8, 2001 (66 FR 41614).

The August 1, 2001, and September 26, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

*Date of initial notice in **Federal Register**:* August 8, 2001 (66 FR 41614). The August 1, 2001, and September 26, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 4, 2002.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: July 31, 2001.

Brief description of amendment: The amendment deletes Technical Specifications Section 6.18, ''PASS [Post Accident Sampling System]/Sampling and Analysis of Plant Effluents,'' for Millstone Nuclear Power Station, Unit No. 2 and thereby eliminates the requirements to have and maintain the post-accident sampling program.

Date of issuance: January 8, 2002.

Effective date: As of the date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment No.: 262.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* October 31, 2001 (66 FR 55009).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 8, 2002.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: July 31, 2001.

Brief description of amendment: The amendment deletes Technical Specifications Section 6.8.4.d, ''Post Accident Sampling,'' for Millstone Nuclear Power Station, Unit No. 3 and thereby eliminates the requirements to have and maintain the post-accident sampling program.

Date of issuance: January 8, 2002.

Effective date: As of the date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment No.: 201.

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

*Date of initial notice in **Federal Register**:* October 31, 2001 (66 FR 55011).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 8, 2002.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut Inc., et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: April 23, 2001, as supplemented by letters dated June 29, July 19, and November 13, 2001.

Brief Description of amendment: The amendment changes Technical Specification (TS) 3.4.1.6, ''Reactor Coolant System—Isolated Loop Startup'' which includes revisions to the limiting condition for operation. Some of the changes to TS 3.4.1.6 affect restrictions that were included as part of the original Millstone Unit 3 licensing basis allowing power operation with one isolated reactor coolant system loop.

Date of issuance: January 9, 2002.

Effective date: As of the date of issuance and shall be implemented within 90 days from the date of issuance.

Amendment No.: 202.

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

*Date of initial notice in **Federal Register**:* July 11, 2001 (66 FR 36338). The letters dated June 29, July 19, and November 13, 2001, provided clarifying information and did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the application as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 9, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 13, 2001.

Brief description of amendment: The change relaxes the allowable cooldown rate in the Reactor Coolant System (RCS) Technical Specifications (TS) 3.4.8.1, ''Pressure/Temperature Limits.'' Specifically, the change eliminates the limitation of a 10 °F per hour cooldown rate when the RCS temperature is below 135 °F. The proposed limitations permit a 100 oF per hour cooldown rate to continue down to an RCS temperature of 110 °F, at which point the rate is reduced to 30 °F per hour.

Date of issuance: January 8, 2002.

Effective date: As of the date of issuance and shall be implemented 60 days from the date of issuance.

Amendment No.: 177.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* May 2, 2001 (66 FR 22029).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 8, 2002.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, 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: July 9, 2001.

Brief description of amendments: The proposed amendments would incorporate TS changes that are being made to provide consistency with the changes to 10 CFR 50.59, ''Changes, tests, and experiments,'' as published in the **Federal Register** (64 FR 53582), dated October 4, 1999. Specifically, the changes replace the terms ''safety evaluation'' with ''10 CFR 50.59 evaluation'' and ''unreviewed safety question'' with ''requires NRC approval pursuant to 10 CFR 50.59.''

Date of issuance: December 28, 2001.

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

Amendment Nos.: 125 and 120.

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****:* August 22, 2001 (66 FR 44170).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 28, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, 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: September 21, 2001.

Brief description of amendments: The amendments delete Technical Specification 5.5.3, ''Post Accident Sampling,'' and thereby eliminate the requirement to have and maintain the Post Accident Sampling System at the Braidwood Station, Unit Nos. 1 and 2, and Byron Station, Units Nos. 1 and 2.

Date of issuance: December 27, 2001.

Effective date: As of the date of issuance and shall be implemented within 365 days.

Amendment Nos.: 126 and 121.

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****:* October 31, 2001 (66 FR 55018).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 27, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: June 15, 2001, as supplemented by letter dated November 12, 2001.

Brief description of amendments: The proposed amendments would allow use of ATRIUM 10 fuel from Framatome Advanced Nuclear Fuel, Inc.

Date of issuance: December 27, 2001.

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

Amendment Nos.: 152 and 138.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* August 8, 2001 (66 FR 41618).

The November 12, 2001, submittal was clarifying in nature and did not change the scope of the original notice or proposed no significant hazards finding dated August 8, 2001 (66 FR 41618). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 27, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: June 14, 2000, as supplemented December 19, 2001.

Brief description of amendment: The amendment changes the operating license to reflect a change in the name of IES Utilities, Inc., a co-owner of the Duane Arnold Energy Center and licensee, to Interstate Power and Light Company.

Date of issuance: January 2, 2002.

Effective date: As of January 1, 2002, and shall be implemented within 30 days.

Amendment No.: 244.

Facility Operating License No. DPR–49: The amendment revised the operating license.

Date of initial notice in ***Federal Register***: July 26, 2000 (65 FR 46009).

The December 19, 2001, supplemental letter provided notification that (1) the required regulatory approvals for the merger had been received and (2) the projected schedule for the merger was January 1, 2002. The supplemental letter did not change the staff's initial proposed no significant hazards consideration determination or expand the application beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 2, 2002.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: January 8, 2001, as supplemented on February 6, December 7, and December 27, 2001.

Brief description of amendment: The amendment revises Technical Specification (TS) 4.5.1.b.1 to change the minimum acceptable Core Spray subsystem flow from 6,350 gallons per minute (gpm) to 6,150 gpm.

Date of issuance: January 7, 2002.

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

Amendment No.: 136.

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

*Date of initial notice in **Federal Register**:* January 22, 2001 (66 FR 6701).

The letters dated February 6, December 7, and December 27, 2001, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 7, 2002.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: April 16, 2001, as supplemented on July 5, 2001.

Brief description of amendments: The amendments revise Technical Specifications (TSs) requirements associated with the operation and surveillance testing of the 28 Volt D.C. (VDC) Batteries. The revised Limiting Condition for Operation (LCO) and Surveillance Requirements (SRs) are now more consistent with the 125 VDC Battery System LCO and SRs as well as similar to standard TSs provided by NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 1, dated April 1995.

Date of issuance: January 4, 2002.

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

Amendment Nos.: 249 and 229.

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

*Date of initial notice in **Federal Register**:* November 14, 2001 (66 FR 57124).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 4, 2002.

No significant hazards consideration comments received: No.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: June 19, 2001, as supplemented by letters dated August 15, August 31, November 20, and December 17, 2001.

Brief description of amendments: The amendments modified Facility Operating License Nos. NPF–87 and NPF–89 to reflect the direct transfer of control of TXU Electric Company's operating authority and 100-percent ownership interest in the Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, to a newly formed generating company: TXU Generation Company LP.

Date of issuance: January 1, 2002.

Effective date: As of the date of issuance and shall be implemented within 7 days from the date of issuance.

Amendment Nos.: 90 and 90.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Facility Operating Licenses.

*Date of initial notice in **Federal Register***: August 20, 2001 (66 FR 43594).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 21, 2001.

Dated at Rockville, Maryland, this 11th day of January 2002.

For the Nuclear Regulatory Commission,

John A. Zwolinski,

Director, Division of Licensing, Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 02–1210 Filed 1–18–02; 8:45 am]

BILLING CODE 7590–01–P

OVERSEAS PRIVATE INVESTMENT CORPORATION

Sunshine Act; Meeting

Time and Date: Thursday, January 31, 2002, 1:00 PM (Open Portion) 1:30 PM (Closed Portion).

Place: Offices of the Corporation, Twelfth Floor Board Room, 1100 New York Avenue, NW., Washington, DC

Status: Meeting Open to the Public from 1 p.m. to 1:30 p.m. closed portion will commence at 1:30 p.m. (approx.).

Matters to be Considered:

1. President's Report
2. Appointment: Daniel A. Nichols
3. Meeting Schedule Through September 2002

Further Matters to be Considered: (Closed to the Public 1:30 PM)

1. Finance Project in Indonesia
2. Finance Project in Pakistan
3. Pending Major Projects
4. Reports

Contact Person for Information: Information on the meeting may be obtained from Connie M. Downs at (202) 336–8438.

Dated: January 17, 2002.

Connie M. Downs,

OPIC Corporate Secretary.

[FR Doc. 02–1677 Filed 1–17–02; 4:02 pm]

BILLING CODE 3210–01–M

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

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

Extension:

Form CB, OMB Control No. 3235–0518, SEC File No. 270–457

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

Form CB is a tender offer statement filed in connection with a tender offer for a foreign private issuer. This form is used to report an issuer tender offer conducted in compliance with Exchange Act Rule 13e–4(h)(8) and a third-party tender offer conducted in compliance with Exchange Act Rule 14d–1(c). It also is used by a subject company pursuant to Exchange Act Rule 14e–2(d). This information is made available to the public. Information provided on Form CB is mandatory. Approximately 200 issuers file Form CB annually and it takes approximately .5 hours per response for a total of 100 annual burden hours. Finally, persons who respond to collection contained in Form CB are not required to respond unless the form displays a currently valid control number.

Written comments regarding the above information should be directed to the following persons: (i) Desk Officer for the Securities and Exchange Commission, Office of Information and Regulatory Affairs, Office of Management and Budget, Room 10102, New Executive Office Building, Washington, DC 20503; and (ii) Michael E. Bartell, Associate Executive Director, Office of Information Technology, Securities and Exchange Commission, 450 Fifth Street, NW, Washington, DC 20549. Comments must be submitted to OMB within 30 days of this notice.

Dated: January 11, 2002.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 02–1424 Filed 1–18–02; 8:45 am]

BILLING CODE 8010–01–M