amendment application are available electronically through ADAMS and can be accessed through the Public Electronic Reading Room (PERR) link <*http://www.nrc.gov/NRC/ADAMS/index.html*> at the NRC Homepage.

A request for a hearing or petition for leave to intervene should be filed within 30 days after publication of this notice in the **Federal Register**, if possible. Any request for hearing or petition for leave to intervene shall be served by the requestor or petitioner upon the applicant, the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington DC 20555; the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555; and the Executive Secretary, U.S. Department of State, Washington, DC 20520.

The information concerning the amendment application follows.

NRC IMPORT LICENSE AMENDMENT APPLICATION

Name of applicant; Date of application; Date received; Application No.	Description of material			Country of origin
	Material type	Total qty	End use	
Starmet CMI; September 28, 2000; October 4, 2000; IW008/01.	Depleted uranium swart/turnings; DU solid cylindrical pieces and.	Increase from 80,000 kgs to 250,000 kgs DU.	DU will be recycled	United Kingdom.
	Contaminated mineral oil ..	Increase from 45,000 liters to 240,000 liters mineral oil.	Oil will be processed and reused.	United Kingdom.

Dated this 25th day of October 2000 at Rockville, Maryland.

For the Nuclear Regulatory Commission,

Ronald D. Hauber,

Deputy Director, Office of International Programs.

[FR Doc. 00–28033 Filed 10–31–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–271]

Vermont Yankee Nuclear Power Corporation; Correction

The October 18, 2000, **Federal Register** contained a ''Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing,'' (65 FR 62393) for the Vermont Yankee Nuclear Power Station. This notice offered an opportunity for comment or hearing requests.

Inadvertently, this was the second offering of such opportunity as a notice had already been published in the September 27, 2000, **Federal Register** (65 FR 68111). The 30-day comment/hearing request deadline is October 27, 2000, at 4:15 p.m. as stated in the September 27, 2000, **Federal Register**.

Dated at Rockville, Maryland this 26th day of October 2000.

For the Nuclear Regulatory Commission.

Richard P. Croteau, Sr.,

Project Manager, Section 2, Project Directorate I, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 00–28034 Filed 10–31–00; 8:45 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 9, 2000, through October 20, 2000. The last biweekly notice was published on October 18, 2000.

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 Review and Directives Branch, Division of Administration Services, Office of Administration, U.S. Nuclear Regulatory

Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC through September 22, 2000. The NRC has relocated its Public Document Room to the NRC's headquarters building. Effective September 26, 2000, documents may be examined at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By December 1, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of amendment request: October 4, 2000.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to support the replacement of the current Westinghouse Model D4 steam generators (SGs) with Westinghouse Model Delta 75 replacement steam generators (RSGs), which is planned to occur during the fall 2001 refueling outage. The proposed changes to the TS are required in part as a result of the physical differences between the currently installed Model D4 SGs and the Model Delta 75 RSGs. The Model Delta 75 RSGs have thermally treated alloy 690 tube material, which has been proven through laboratory testing and operational experience to provide increased corrosion resistance compared to the Inconel 600 tube material in the Model D4 SGs. The licensee has completed a comprehensive engineering review program in support of the SG replacement. This program evaluated the difference between the current Model D4 SGs and the Model Delta 75 RSGs in conjunction with restoration of the original nominal reactor coolant average temperature (Tavg) of 588.8°F. The supporting analyses and evaluations were performed to support operations with the RSGs at the current licensed core power of 2775 MWt and also, where possible, at an uprated core power level of 2900 MWt. This license amendment application, however, does not include a request for a power uprate.

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 does not affect accident initiators. The change in RCS [reactor coolant system] volume, an input in certain accident analyses, is acceptable since applicable acceptance criteria continue to be met. The change to the calculated residual heat removal (RHR) cooldown time does not adversely impact safe shutdown and meets Standard Review Plan (SRP) requirements. The analysis results for peak accident pressure (P_a remain below the containment design pressure limit. Revised Technical Specification limits, such as SG minimum water levels, have been analytically determined based on the new SG design features to preserve analysis assumptions. Since the proposed change has no effect on accident initiators, the probability of design basis events is not impacted by these analyses.

The leak rate assumed in accident analyses is conservative and independent of the value of P_a. Since the peak calculated accident pressure remains below the design value, the containment function as a barrier to the release of radioactive material is not adversely affected. Dose consequences have been analyzed with respect to the proposed change and applicable acceptance criteria continue to be met.

HNP [Harris Nuclear Plant] structures, systems and components, as well as the RSGs, are designed to operate at the original T_{avg} value of 588.8°F. The comprehensive engineering review performed to support SG replacement includes evaluation and analysis results for transients and design basis accidents that meet applicable acceptance criteria. The proposed DNB [departure from nucleate boiling] parameter values ensure the core and reactor coolant system do not exceed their safety limits. The trip setpoints initiate safeguards actions that mitigate postulated accident consequences. Since plant systems will function as designed and performance requirements are verified to be acceptable with the proposed changes, there are no impacts to accident initiators or precursors. Therefore, these changes do not involve a significant increase in the probability of an accident previously evaluated.

The RSGs are designed to minimize potential for the types of tube degradation experienced with the OSGs [original steam generators]. The changes to reactor core safety limits, RTS/ESFAS [reactor trip system/engineered safety feature actuation system] values and T_{avg} limit support transient and accident analysis assumptions relating to fuel clad failure. Since SG tube and fuel clad integrity will continue to be adequately maintained, these barriers to release of radioactive material will perform their required function. All dose consequences continue to meet acceptance criteria. As a result, the changes do not involve a significant increase in consequences of previously evaluated accidents.

The proposed change reduces Technical Specification limits on reactor coolant activity to ensure that the dose consequences of the proposed new SGTR [steam generator tube rupture] analysis meet guidelines. The Technical Specification limits are not accident initiators and the proposed change provides additional conservatism in the limiting initial condition for RCS activity level. The change in these limits functions only to mitigate consequences of RCS liquid and steam release events and does not impact methods for plant operation. Since there are no adverse impacts to initiators of accidents previously evaluated, there is no significant increase in the probability of such accidents.

The proposed change to the basis for thyroid dose conversion factors is an accepted update to that used in the analysis of record. Dose consequences for all analyses, including SGTR, remain within allowable guidelines. Although there are increases to previously reported dose consequences, the increase is not significant since the results remain under the SRP acceptance criteria.

Reducing the required time at hot standby to maintain adequate CST [condensate storage tank] minimum volume ensures accident mitigation capability and is not involved with accident initiation. Conservative reactor trip setpoints during power operation with inoperable main steam line safety valves ensure accident analysis assumptions for maximum secondary system pressure. The proposed setpoint changes are also unrelated to initiation of previously evaluated accidents.

The removal of F* plugging criteria is consistent with NUREG–0452, ''Standard Technical Specifications.'' The remaining Technical Specification plugging criterion is based on a minimum wall thickness due to wastage as determined by ASME [American Society of Mechanical Engineers] Section XI. The proposed change is conservative because operation with F* degraded tubes is eliminated.

The potential for a tube rupture is expected to be acceptably low based on the qualification analysis and testing for the RSGs. The proposed program for periodic in-service inspection monitors the integrity of the SG tubing to provide reasonable assurance that there is sufficient time to take proper and timely corrective action if any tube degradation is detected. The deleted OSG tube inspection requirements are not applicable to the RSG due to design differences.

SG tube inspections are not accident initiators and the function of the inspection program is to identify and eliminate, when appropriate, accident precursor conditions. Because of this, the probability of postulated accidents (*e.g.,* SG tube rupture) is not increased.

Since the proposed changes are either conservative (reduced RTS setpoints, elimination of F* tube plugging criteria) or related to mitigation of postulated accidents (hot standby time prior to cooldown), there is no significant increase in dose consequences.

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 adversely impact methods of plant operation and new methods of plant systems operation are not created. The impacts of SG replacement on NSSS [nuclear steam supply system] and other plant systems, including the change in RCS volume and other design differences, have been evaluated and design and performance requirements continue to be met. Accordingly, no new accident scenarios, failure mechanisms or limiting single failures are introduced.

Changing P_a to reflect the results of current analyses while maintaining its value below the containment design pressure will not introduce significant or adverse changes to the plant design basis that would create a new or different kind of accident.

Since HNP structures, systems and components are designed for operation at the proposed T_{avg}, returning to this value following SG replacement does not introduce a new method of plant operation. The proposed reactor core safety limits, RTS/ESFAS and T_{avg} limit changes do not introduce new failure mechanisms or limiting single failures since the values have been shown to preserve transient and accident analysis assumptions. The acceptable evaluation results demonstrate that the changes do not adversely affect or challenge the performance or integrity of safety related systems. Hence, no new accident scenarios are created.

The proposed change to specific activity limits does not modify plant systems, structures, or components or impact methods of plant operation. The SGTR analysis that provides the basis for the change demonstrates adequate margin to overfill based on revised emergency operating procedures [EOPs]. Operator actions and response times have been demonstrated to be acceptable and to satisfy analysis assumptions in simulator exercises. The operator actions do not represent different types of actions from those currently included in the EOPs, and are not subject to errors that would create new failure mechanisms.

No new types of failures and no new or different accident initiators are created. As a result, the change will not introduce significant or adverse changes to the plant design basis that could lead to the creation of a new or different kind of accident.

The revised thermal power restrictions imposed by [the] Technical Specification[s] when there are inoperable main steam line safety valves are conservative and consistent with the Technical Specification Bases methodology. The reduction in hot standby time to six hours continues to satisfy requirements for safe shutdown. No new failure mechanisms or limiting single failures are introduced and there are no adverse effects or challenges to the performance or integrity of safety related systems.

Removing application of F* repair criteria and sleeving methodologies from [the] Technical Specification[s] with the RSGs does not introduce significant or adverse changes to the plant design basis that could lead to a new or different kind of accident being created. These requirements were needed to address design and material conditions of the original HNP SGs that are not applicable to the RSGs. This change does not alter the objective of SG surveillance activities—maintaining the structural integrity of this portion of the reactor coolant system. The surveillance activities are performed during outages and the proposed surveillance program is consistent with regulatory guidance. No new failures are created.

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.

Analyses supporting the change reflect the RSG design, including changes to primary and secondary volume and minimum water level for adequate heat sink. Mass and energy and containment response analyses model ESF [engineered safety features] systems and safety analysis limits. The analysis results demonstrate that applicable acceptance criteria, including DNB, RCS pressure, peak cladding temperature, containment pressure and temperature, and dose, continue to be met.

The calculated containment peak accident pressures result in a value of P_a that remains below the containment design pressure. The margin of safety during design basis events is preserved if P_a does not exceed the design pressure. In addition, performance of the containment leakage rate testing program ensures the validity of the assumptions used in the accident analyses.

All evaluations and analyses performed to support SG replacement reflect the proposed values. The results demonstrate that applicable acceptance criteria (including DNB, RCS pressure and temperature, LOCA [loss-of-coolant accident] peak clad temperature, containment pressure and temperature, and dose) continue to be met. Calculations confirm that the proposed RTS/ESFAS values provide margin to the safety analysis limits, thereby ensuring that the margin of safety inherent in the safety analysis limit itself is preserved.

The specific activity limit has no effect on safety limits or limiting safety system settings. The only impact of the specific activity limits is on dose consequences for certain design basis accidents such as SGTR. The proposed change is in the conservative direction for dose. The SGTR analysis is the most limiting of these accident dose evaluations. The proposed new analysis demonstrates that the SGTR consequences remain within allowable SRP guideline limits, which establish the margin of safety. Since the analysis results remain below the SRP guidelines, the margin of safety is not reduced.

The thermal power restrictions imposed by [the] Technical Specification[s] when there are inoperable main steam line safety valves [ensure] sufficient relieving capacity during all postulated transients, thereby maintaining margin of safety for secondary system pressure.

The proposed in-service inspection program for the RSGs monitors the integrity of the SG tubing and is consistent with NUREG–0452. Since the F* repair criteria and sleeving methods specifically addressed OSG degradation issues, their removal does not involve a reduction in a margin of safety. Analyses demonstrate that the RSG tubing retains adequate structural and leakage integrity using the proposed plugging criteria during normal, transient, and postulated accident conditions. The RSG tubing design and proposed plugging criteria are consistent with applicable ASME requirements and do not allow for operation with indications identified by F* criteria.

The proposed plugging limit maintains the margin of safety by providing for leakage detection and shutdown in the event of an unexpected tube leak while minimizing the potential for excessive leakage or tube burst in the event of [an] MSLB [main steam line break] or LOCA.

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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: September 20, 2000.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 5.5.7.d to decrease the maximum allowed pressure drops across control room emergency filtration (CREF) make-up and recirculation train filters and charcoal adsorbers. Specifically, the pressure drops would be changed from ''6.0 inches WG [water gauge]'' to ''3.0 inches WG'' for the CREF makeup train, and from ''8.0 inches WG'' to ''4.2 inches WG'' for the CREF recirculation train. Additionally, the words ''(CREF only)'' would be removed to clarify that standby gas treatment system (SGTS) prefilter is included in the Ventilation Filter Testing Program (VFTP).

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 revise the pressure drop acceptance criteria of TS Section 5.5.7.d from ''6.0 inches WG'' to ''3.0 inches WG'' for the CREF makeup train, and from ''8.0 inches WG'' to ''4.2 inches WG'' for the CREF recirculation train. The change in pressure drop reflects the impact of reduced fan speed on system characteristics. The removal of ''(CREF only)'' is editorial and clarifies the inclusion of the SGTS prefilters in the VFTP. These changes assure that the Surveillance procedure appropriately demonstrate[s] the ability of the CREF trains to perform its specified function.

No changes in either system design or operating strategies will be made as a result of these changes. Thus, no opportunity exists to increase the probability or consequences of a previously analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of a previously evaluated accident.

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 involve a change to the plant operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to the initiation of any accidents. This change involves the pressure drop across the filters and adsorbers of the CREF make-up and recirculation trains. The surveillance requirements for performing the actual test have not changed. The VFTP tests are performed such that the pressure drop across the combined HEPA [high efficiency particulate adsorber] filters, the prefilters, and the charcoal adsorbers is less than the value specified in the acceptance criteria in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510–1980 at the system flowrate of plus or minus 10 percent. The change involves restrictive changes in test acceptance criteria and test methodology and no change in system operation.

No new accident scenarios are created by the lower value of filter and adsorber pressure drop. No safety-related equipment or safety functions are altered as a result of this change. Additionally, the removal of ''(CREF only)'' is editorial and clarifies the inclusion of the SGTS prefilters in the VFTP. Therefore, the changes do not create the possibility of a new or different kind of accident or malfunction from those previously analyzed.

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

The proposed change in combination with existing restrictions within the TS provides assurance that there are no credible mechanisms to prevent the CCHVAC [control center heating, ventilation, and air conditioning] system from performing its specified function. The maximum allowable differential pressure is more restrictive and corresponds to the reduced fan speed across the HEPA filters, prefilters, and charcoal adsorbers as a result of this change. There will be no changes in system operating strategies because of the inclusion of the make-up moisture separator/prefilter and test acceptance criteria for the CREF trains. Additionally, the removal of ''(CREF only)'' is editorial and clarifies the inclusion of the SGTS prefilters in the VFTP. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: Claudia M. Craig.

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: June 13, 2000.

Description of amendment request: The proposed amendments would make administrative changes to the McGuire Nuclear Station, Units 1 and 2, Facility Operating Licenses (FOLs). Specifically, they would: (1) Delete existing License Conditions which have been met by completed licensee's actions or are imposed by other regulatory requirements, or (2) make other miscellaneous changes to the McGuire Nuclear Station, Units 1 and 2, FOLs.

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

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

No. The proposed amendment to the FOLs are either administrative, eliminate duplication of other regulatory requirements or delete License Conditions fully met by Duke Energy Corporation. No actual plant equipment, operating practices, or accident analyses are affected by this proposed amendment. Therefore, the proposed amendment has no impact on the possibility of any type of accident: new, different, or previously evaluated.

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

No. The proposed amendment to the FOLs [are] either administrative, eliminate duplication of other regulatory requirements or delete License Conditions fully met by Duke Energy Corporation. No actual plant equipment, operating practices, or accident analyses are affected by this proposed amendment and no failure modes not bounded by previously evaluated accidents are created. Therefore, the proposed amendment has no impact on the possibility of any type of accident: new, different, or previously evaluated.

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

No. Margin of safety is associated with confidence in the ability of the fission product barriers (*i.e.,* fuel and fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed amendment to the FOLs are either administrative, eliminate duplication of other regulatory requirements or delete License Conditions fully met by Duke Energy Corporation. Therefore, no reduction in any existing margin of safety is involved.

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 L. Emch, Jr.

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: August 22, 2000.

Description of amendment request: The proposed amendments would revise the Technical Specifications and associated Bases to limit the amount of time a Refueling Water Storage Tank level channel can be in a trip condition.

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

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

No. The proposed change modifies the allowed outage time that a channel of the Refueling Water Storage Tank (RWST) can be in the tripped condition from an indefinite period of time to a more conservative maximum of 48 hours. The Engineered Safety Features Actuation System (ESFAS) is an accident mitigating system, and not an accident initiator. Therefore, the proposed change will have no impact on any accident probabilities. Accident consequences will not be affected, as no changes are being made to the plant involving a reduction in reliability or effectiveness of the Emergency Core Cooling System (ECCS). Consequently, any previous evaluations associated with accidents will not be affected by this change.

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

No. The proposed change does not modify the design or configuration of the plant. The proposed change provides a more conservative time limit for a channel to be in the tripped condition. No physical changes are being made to plant systems, structures or components nor will the proposed change reduce the ability of any of the safety related equipment required for accident mitigation. Consequently, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change 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 accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed change provides a more restrictive time limit for a channel of the RWST to be in a tripped condition than is currently allowed by the ITS. The performance of the fission product barriers will not be degraded by the proposed changes. Consequently, plant safety analyses will not be affected.

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 L. Emch, Jr.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: September 1, 2000.

Description of amendment request: The proposed amendment would make a change to Pilgrim Technical Specification Table 4.6–3. The proposed change substitutes ''21 (approx)'' under the column ''Effective Full Power Years (EFPY)'' for the current ''18 (approx).'' This proposed change would defer the second reactor vessel surveillance capsule pull to allow Pilgrim to pursue participation in the Boiling Water Reactor Integrated Surveillance Program Plan.

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

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

Pressure-temperature (P/T) limits (Pilgrim Technical Specifications Figures 3.6.1, 3.6.2, and 3.6.3) are imposed on the reactor coolant system to ensure that adequate safety margins against nonductile or rapidly propagating failure exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P/T limits are related to the nil-ductility reference temperature, RTndt, as described in American Society of Mechanical Engineers (ASME) Section III, Appendix G. Changes in the fracture toughness properties of reactor pressure vessel (RPV) beltline materials, resulting from the neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10 CFR Part 50, Appendix H. The effect of neutron fluence on the shift in the nil-ductility reference temperature of pressure vessel steel is predicted by methods given in Regulatory Guide (RG) 1.99, Revision 2.

The proposed change is a revision of the second surveillance capsule withdrawal time, identified in Technical Specification Table 4.6–3, from approximately 18 effective full power years (EFPY) to approximately 21 EFPY. This change will not affect P/T limits as given in Pilgrim Technical Specifications Figures 3.6.1, 3.6.2, and 3.6.3. This change is not related to any accidents previously evaluated. This change will not affect any plant safety limits or limiting conditions of operation. The proposed change will not affect reactor pressure vessel performance because no physical changes are involved and Pilgrim vessel P/T limits will remain in accordance with RG 1.99, Revision 2 requirements. The proposed change will not cause the reactor pressure vessel or interfacing systems to be operated outside of their design or testing limits. Also, the proposed change will not alter any assumptions previously made in evaluating the radiological consequences of accidents.

Therefore, the probability or consequences of accidents previously evaluated will not be increased by the proposed change.

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

The proposed change revises the second RPV material surveillance capsule withdrawal time in Pilgrim Technical Specification Table 4.6–3 from approximately 18 EFPY to approximately 21 EFPY. This proposed change does not involve a modification of the design of plant structures, systems, or components (SSCs). The proposed change will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. The proposed change will not degrade the reliability of SSCs important to safety because equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be downgraded, the frequency of operation of equipment important to safety will not be increased, and increased or more severe testing of equipment important to safety will not be imposed. No new accident types or failure modes will be introduced as a result of the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from that previously evaluated.

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

Operation in accordance with the existing P–T curves ensures reactor vessel and cooling system integrity. The capsule pull is a surveillance technique that provides data for modification of the curves. The margins will not change without new data from the capsule pull. The NRC-approved methods used to develop the temperatures associated with the current curves reside in Pilgrim's TSs and have been determined to be conservative. The proposed change will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed change does not represent a change in initial conditions, or in a system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. Therefore, the proposed changes do not involve a significant reduction in any margins of safety.

Based on the staff's analysis, 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: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts 02360–5599

NRC Section Chief: James W. Clifford.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: August 7, 2000.

Description of amendment request: The proposed changes would revise Technical Specification (TS) Bases 3/4.3.1 and 3/4.3.2 to clarify the actions that must be performed when Steam and Feedwater Rupture Control System (SFRCS) components and SFRCS-actuated components are inoperable. Specifically, the proposed changes would provide guidance on which TS actions are applicable for SFRCS-actuated components. Also, the proposed changes would introduce new TS 3/4.7.1.8 which would provide appropriate requirements for the Main Feedwater Control Valves and the Startup Feedwater Control Valves. Additionally, the proposed changes would introduce new TS 3/4.7.1.9 which would provide requirements for the Turbine Stop Valves.

Basis for proposed no significant hazards consideration determination:

As required by 10 CFR 50.91(a), the licensees have provided their analysis of the issue of no significant hazards consideration, which is presented below:

The Davis-Besse Nuclear Power Station (DBNPS) has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. The amendment application proposes to revise Technical Specification (TS) Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, to clarify Steam and Feedwater Rupture Control System (SFRCS) requirements; to add new TS 3/4.7.1.8, Main Feedwater Control Valves (MFCVs) and Startup Feedwater Control Valves (SFCVs); to add new TS 3/4.7.1.9, Turbine Stop Valves (TSVs); and to make associated changes to the Bases and TS Index.

The proposed change would provide clear guidance for the actions required when SFRCS logic and SFRCS-actuated components are inoperable. The new TSs for MFCVs, SFCVs, and TSVs provide appropriate operational requirements for these SFRCS-actuated components. The proposed change will not change any system hardware or testing requirements. Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report (USAR). The proposed changes to the TS Bases and Index are consistent with the changes described above.

The proposed changes are consistent with the intent of NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' Revision 1, April 1995. The proposed change does not involve a change to any plant hardware and does not affect the probability of any equipment malfunction or accident-initiating event.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation, or radiological releases are not affected by the proposed changes. The proposed changes ensure that excessive unavailability of SFRCS-actuated equipment which is used to mitigate accident consequences does not occur. The reliability of the plant hardware and the ability of the SFRCS-actuated equipment to perform its safety function are not affected. Existing system and component redundancy is not affected by the proposed changes. The existing system and component operation is not affected by the proposed changes, and the assumptions used in evaluating the radiological consequences in the DBNPS USAR are not invalidated. Therefore, for each postulated accident the consequences remain bounded by the consequences from the previously evaluated accidents.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these proposed changes do not involve any physical changes to systems or components, nor do they alter the manner in which the systems or components are operated.

3. Not involve a significant reduction in a margin of safety because, for the proposed changes, there are no new or significant changes to the initial conditions contributing to accident severity or consequences. Accordingly, there are no significant reductions in a margin of safety.

On the basis of the above, the DBNPS has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: September 19, 2000.

Description of amendment request: The proposed amendment revises the Standby Liquid Control (SLC) boron solution requirements in TS Figure 3.1.7–1 to ensure a minimum boron concentration of 660 parts per million (ppm) in the reactor.

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 current TS SRs ensure acceptable SLC boron solution volume and concentration values which produce a minimum boron concentration of 600 ppm in the reactor. This proposed amendment revises the boron solution requirements to ensure a minimum boron concentration of 660 ppm in the reactor. A minimum boron concentration of 660 ppm in the reactor is sufficient to initiate and maintain reactor subcriticality as the nuclear system cools. The DAEC is currently pursuing the use of GE–14 fuel, power uprate to 1912 MWth and extended cycle length. Increasing the minimum boron concentration to 660 ppm in the reactor will provide adequate shutdown margin for these future core designs. A General Electric analysis, using the approved methods described in Revision 14 of General Electric Standard Application for Reactor Fuel (GESTAR II), NEDE 24011–P–A, provides the basis for the increase in minimum boron concentration. This analysis assumes DAEC operation with an equilibrium core of GE–14 fuel, operating at 1912 MWth with 24 month cycles and bounds all planned future core designs.

The concentration requirement of the SLC boron solution will continue to comply with the requirements of 10 CFR 50.62(c)(4).

The proposed editorial change to the title of TS Figure 3.1.7–1 provides clarity only and does not alter performance of TS SR 3.1.7.1 and SR 3.1.7.5.

Therefore, this 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.

Ensuring a minimum boron concentration of 660 ppm in the reactor will result in an increase in the acceptable minimum volume of boron solution in the SLC tank. The tank can accept the increased volume of boron solution. The tank low and high level alarms will be altered; these alarms alert the operator to take action in the event boron solution volume is decreasing or increasing, respectively. Increasing the volume in the SLC tank and increases to the low and high level alarm setpoints will have no effect on operator actions to maintain system operability. Revising the SLC boron solution requirements to ensure a minimum boron concentration of 660 ppm in the reactor cannot create a new or different kind of accident.

The proposed editorial change to the title of TS Figure 3.1.7–1 provides consistent wording between the Figure, TS SR 3.1.7.1 and the TS BASES to avoid potential inconsistencies when determining the tank volume acceptability.

Therefore this 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 a margin of safety.

DAEC TS SRs ensure SLC boron solution volume and concentration values which produce a minimum boron concentration of 600 ppm in the reactor. This proposed amendment revises SLC boron solution requirements to ensure a minimum boron concentration of 660 ppm in the reactor. A boron concentration of 660 ppm in the reactor is sufficient to initiate and maintain reactor subcriticality as the nuclear system cools. The DAEC is currently pursuing the use of GE–14 fuel, power uprate to 1912 MWth and extended cycle length. A General Electric (GE) analysis, using the approved methods described in Revision 14 of General Electric Standard Application for Reactor Fuel (GESTAR II), NEDE 24011–P–A, provides the basis for the increase in minimum boron concentration. GE's analysis demonstrated that at a minimum boron concentration of 600 ppm in the reactor, the SLC shutdown margin was marginal with respect to the TS requirement for shutdown margin. At a minimum boron concentration of 660 ppm in the reactor, sufficient margin is maintained such that TS Section 3.1.1 will be met under all anticipated conditions. The GE analysis assumes DAEC operation with an equilibrium core of GE–14 fuel, operating at 1912 MWth with 24 month cycles and bounds all planned future core designs.

The concentration requirement of the SLC boron solution will continue to comply with the requirements of 10 CFR 50.62(c)(4).

The SLC system is designed to inject a quantity of boron that produces a minimum boron concentration of 600 ppm in the reactor. An additional 25% of that quantity of boron is also injected to compensate for imperfect mixing, leakage and volume in other small piping connected to the reactor. This margin will be maintained such that an additional 25% of the quantity of boron required to achieve a minimum boron concentration of 660 ppm in the reactor will also be injected. Therefore, this proposed amendment will not involve a significant reduction in a margin of safety.

The proposed editorial change to the title of TS Figure 3.1.7–1 provides consistent wording between the Figure, TS SR 3.1.7.1 and the TS BASES to avoid potential inconsistencies when determining tank volume acceptability.

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: Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Claudia M. Craig.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: September 26, 2000.

Description of amendment request: The licensee proposed to amend the unit's Technical Specifications (TSs), Sections 3.3.0, 3.3.2 thru 3.3.5 and 3.3.7 to allow performance of reactor vessel hydrostatic or leakage tests, and scram time and excess flow check valve tests, when the reactor system temperature is above 215 degrees Fahrenheit and the reactor is not critical, without having to maintain primary containment integrity. These proposed changes are consistent with the guidelines of NUREG–1433, Revision 1, ''Standard Technical Specifications, General Electric Plants, BWR/4.'' Concurrently, the licensee proposed to amend TS Sections 3.4.0 thru 3.4.5 to require reactor building (secondary containment) integrity when the reactor system temperature is above 215 degrees Fahrenheit. This will ensure that, should accidents occur during the above tests, the reactor building will perform its function to contain radiological releases. The licensee also proposed to impose a dose-equivalent iodine level of 1.5 microcuries per gram (as opposed to 9.47 microcuries per gram allowed during plant operation) during the above conditions to limit radiological consequences of possible accidents to limits bound by the previously analyzed main steam line break accident.

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

The operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes will result in not requiring containment integrity, but requiring reactor building integrity (secondary containment) during performance of the specified tests. These are changes in configuration (no hardware design change is involved) to permit testing of various systems or components under different conditions. The changed conditions are not considered precursor of accidents. Accordingly, the probability of an accident previously evaluated is not increased.

Accidents could occur during a test. As part of the proposed TS change, the licensee proposed to limit the dose-equivalent iodine level to ensure that the consequences of possible accidents will continue to be bounded by a previously analyzed accident, the main steam line break accident. Thus, there will not be any increase in the consequences of an accident previously evaluated.

The operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No hardware design change is involved with the proposed TS changes. Other than the changes in the configuration of the containment, the reactor building, and the permissible dose-equivalent iodine level, there are no other procedural changes. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed TS changes will not adversely affect the performance characteristics and intended functions of systems and components covered by the above-listed TS sections. The tests, not affected by the proposed amendment, will ensure the ability of the subject systems and components to perform their intended function. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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: Marsha Gamberoni.

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

Date of amendment request: August 25, 2000.

Description of amendment request: Millstone Unit No. 1 (MP1) is being decommissioned. To support this activity, several modifications are required to modify/eliminate MP1 systems that support the operation of structures, systems, and components that are shared or common to Millstone Unit No. 2 (MP2) and Millstone Unit No. 3 (MP3). One of the separation projects entails the replacement of the existing MP1 to MP2 4160-volt cross-tie with a new MP3 to MP2 4160-volt cross-tie. Northeast Nuclear Energy Company (NNECO) has proposed a one time extension to the allowed outage time (AOT) for Action a.2 of Technical Specification (TS) 3.8.1.1, ''Electrical Power Systems—A. C. Sources—Operating'' in order to complete this modification. The proposed change would extend the AOT from 72 hours to 14 days, provided the MP3 station blackout (SBO) diesel generator (DG) is available to supply MP2; otherwise the AOT would only be extended to 7 days.

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

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

The offsite circuits supply power to equipment required to support the safe shutdown and post-accident operations of MP2. The preferred off-site power supply is from the 345 kV [kilovolt] switchyard, through the reserve station service transformer (RSST). The alternate (delayed) source of offsite power is the 4.16 kV cross-tie from MP1 via bus 14H.

To ensure that the probability of a complete loss of offsite power is not significantly increased, the MP1 4.16 kV cross-tie will only be removed from service when the weather conditions and forecast are favorable. Additionally, during the time that the alternate offsite source is inoperable, actions will be taken to protect the operable offsite circuit (i.e., no work will be conducted that could challenge the operability of the offsite circuit).

Although the offsite circuits provide power to components that help mitigate the consequences of accidents previously evaluated, the extension in the AOT does not affect any of the assumptions used in the deterministic evaluations of these accidents. Thus, this change will not increase the consequences of any accident previously analyzed.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed change is an extension to a TS AOT. It does not alter the physical design, configuration, or method of operation of the plant. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously analyzed.

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

During the implementation of the modification to provide MP2 with a 4.16 kV cross-tie with MP3, NNECO will:

a. Appropriately consider the 7 day and 14 day weather forecasts prior to removing the MP2 4.16 kV cross-tie with MP1 from service to minimize the potential for loss of offsite power due to severe weather or salt spray.

b. Protect the equipment redundant to the systems removed from service or whose power supply is affected by the modification. This includes limiting work on the 345 kV lines, the switchyard, the RSST, the diesel generators, the service water system, the high pressure safety injection system, and the reactor building closed cooling water system. This restriction will ensure that MP2 will remain capable of mitigating any potential design basis accident during the implementation of the modification.

c. Within 7 days of entering Action a. of TS 3.8.1.1, establish the capability to supply MP2 with power from the MP3 SBO DG via operator actions within one hour of an event resulting in a loss of the remaining offsite source of power. The capability to utilize the MP3 SBO DG is a contingency measure (i.e., will be able to serve as a temporary diesel).

Additionally, NNECO has evaluated the dominant sequences affecting plant risk using probabilistic safety analysis techniques. The analysis determined that the Delta Core Damage Probability associated with the extended allowed outage time was small.

There will be no significant reduction in a margin of safety because the increased risk is acceptable, focus on maintaining the operability of the redundant equipment and systems will be increased, the probable weather conditions will be appropriately considered, and the capability to supply MP2 with power from the MP3 SBO DG via operator action will be established within 7 days of entering Action a. of TS 3.8.1.1. The capability to utilize the MP3 SBO DG is a contingency measure (i.e., will be able to serve as a temporary diesel).

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

NRC Section Chief: James W. Clifford.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of amendment request: August 25, 2000.

Description of amendment request: Millstone Unit No. 1 (MP1) is being decommissioned. To support this activity, several modifications are required to modify/eliminate MP1 systems that support the operation of structures, systems, and components that are shared or common to Millstone Unit No. 2 (MP2) and Millstone Unit No. 3 (MP3). One of the separation projects entails the replacement of the existing MP1 to MP2 4160-volt cross-tie with a new MP3 to MP2 4160-volt cross-tie. Northeast Nuclear Energy Company (licensee) has evaluated this proposed new cross-tie utilizing the criteria of 10 CFR 50.59. The modification involves four unreviewed safety questions (USQs). One USQ pertains to MP2 and three USQs pertain to MP3. The licensee is requesting approval of the USQs.

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:

Basis for No Significant Hazards Consideration—Millstone Unit No. 2 Unreviewed Safety Question

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

The test that will be performed to verify the ability of the cross-tie to supply approximately 3 MVA of power from MP3 to MP2 will require parallel operation of the MP3 SBO [station blackout] DG [diesel generator] with the MP2 4160 Volt system. To ensure that the MP3 SBO DG and associated connection will be isolated immediately should any abnormal event or failure occur at MP2, a special Class 1E instantaneous overcurrent relaying scheme will be used. This will provide additional assurance that if a fault occurs in the MP3 SBO DG or the associated cross-tie cabling, it will be isolated immediately and should not affect operability of the MP2 safety related 4160 Volt system. In addition, if a loss of normal power occurs at MP2 during the test, the undervoltage condition will be detected by the credited undervoltage scheme, and MP2 will respond as designed to the loss of power.

The performance of this test does not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The response of the plant and the operators following a design basis event will not be significantly different. Therefore, the proposed activity does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The installation of the Class 1E instantaneous overcurrent relay scheme to be used during performance of this test will provide additional assurance that the MP2 safety related/busses will remain operable during the test to parallel the MP3 SBO DG to the MP2 4160 Volt system. The failure of the temporary overcurrent protection scheme would have the same effect as failure of the existing overcurrent protection scheme. Therefore, the proposed activity does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The installation of the Class 1E instantaneous overcurrent relay scheme to be used during performance of this test will provide additional assurance that the MP2 safety related busses will remain operable during the test to parallel the MP3 SBO DG to the MP2 4160 Volt system. This will provide additional assurance that if a fault occurs in the MP3 SBO DG or the associated cross-tie cabling, it will be isolated immediately and should not affect operability of the MP2 safety related 4160 Volt system. In addition, if a loss of normal power occurs at MP2 during the test, the undervoltage condition will be detected by the credited undervoltage scheme, and MP2 will respond as designed to the loss of power.

The proposed activity will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

Basis for No Significant Hazards Consideration—Millstone Unit No. 3 Unreviewed Safety Questions

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

MP3 USQ 1

The MP3 to MP2 4160 Volt cross-tie will impose up to an additional 3 MVA load on MP3. As a result, the minimum acceptable switchyard voltage for MP3 will increase from 334 kV to 337 kV. The analysis performed to support the new design shows that, with a minimum voltage of 337 kV in the switchyard, MP3 will remain connected to offsite power. The minimum voltage protection assures that acceptable starting and running voltages are present for the safety systems that may be required to operate should a loss of coolant accident (limiting design basis event) occur at MP3 while MP2 is being supplied with the safe shutdown load of 3 MVA from either the MP3 RSST [reserve station service transformer] or the MP3 NSST [normal station service transformer]. Thus, the MP3 offsite connection will not be tripped when offsite power is available.

The extra loading imposed by the MP2 safe shutdown loads on the MP3 power supplies is within the MP3 capability to simultaneously supply the worst case MP3 normal and accident loads. The ability of MP3 to mitigate design basis accidents and events is not adversely affected by the new design.

This activity does not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. Also, the response of the plant and the operators following a design basis event will not be significantly different. Therefore, this activity does not involve a significant increase in the probability or consequences of an accident previously evaluated.

MP3 USQ2

The proposed design change will allow the MP3 RSST (or the MP3 NSST, if the MP3 RSST is unavailable) to provide the alternate offsite source for MP2 to meet GDC [General Gesign Criterion] 17 requirements. If the MP3 RSST is not available, MP2 will need to credit the MP3 NSST as the alternate offsite power supply for GDC 17 compliance. The connection for the MP3 NSST is between breakers 13T and 14T. This connection point only provides 1 breaker separation (13T) between the MP2 RSST and the MP3 NSST. If breaker 13T is closed, this arrangement does not provide adequate separation between the two offsite sources as required by GDC 17. Opening breaker 13T when MP3 is shutdown will provide the required separation for MP2, but it will reduce the reliability of the offsite supply for MP3. Therefore, opening the 13T breaker to allow MP2 to meet GDC 17 requirements increases the probability of a loss of offsite power at MP3 when shutdown.

A loss of offsite power at MP3 due to a fault in the offsite distribution network is a low probability event. When combined with the expected frequency of removing the MP3 RSST from service of once per MP3 refueling outage, the probability of a loss of offsite power at MP3 when shutdown as a result of breaker 13T being open is low. In addition, the MP3 shutdown risk program will evaluate the impact of removing the MP3 RSST from service, and plan accordingly. This will ensure the MP3 RSST will be removed from service when the shutdown risk is determined to be acceptably low. Also, at least one MP3 EDG will be available, as required by Technical Specifications, to supply power to necessary loads if offsite power is lost.

This activity does not significantly alter the manner in which the plant is operated. There will be no adverse effect on accident mitigation equipment.

Also, the response of the plant and the operators following a design basis event will not be significantly different. Therefore, this activity does not involve a significant increase in the probability or consequences of an accident previously evaluated.

MP3 USQ 3

The connection of an additional 3 MVA load for MP2 onto the MP3 electrical distribution system will increase the short circuit fault levels on the MP3 electrical distribution system. If worst case conditions are established, the additional contribution from the MP2 loads would increase the fault level to above the switchgear ratings. The worst case short circuit conditions occur with MP3 at full power, the switchyard at its maximum voltage, an MP3 EDG paralleled to the bus for surveillance testing, and the MP3 to MP2 cross-tie supplying 3 MVA of power to MP2.

Although the fault levels could exceed the MP3 switchgear and circuit breaker ratings, it is not expected that the worst case conditions will be frequently established. For the majority of scenarios, this situation can be avoided because the new design allows the 3 MVA of power for MP2 to be supplied by either MP3 bus 34A or 34B. Procedural guidance will be provided to select the MP3 bus (34A or 34B) to supply power to MP2 that powers the opposite train from the train associated with the MP3 EDG to be tested. Even if the worst case plant configuration cannot be avoided, there is a low probability of event occurrence given the low probability of a bus fault coincident with the short time that the MP3 EDG would be paralleled to the system. In addition, the event would only affect one MP3 train of safety related equipment. The other train will be available to function as assumed to mitigate any accidents that may occur.

The use of the MP3 to MP2 cross-tie does not significantly alter the manner in which the plant is operated. Considering the low likelihood of a fault occurring concurrently with worst case conditions, and that only one MP3 train of safety related equipment should be affected, sufficient accident mitigation equipment will be available to mitigate a design basis accident. In addition, the response of the plant and the operators following a design basis event will not be significantly different. Therefore, this activity does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

MP3 USQ 1

The ability to supply 3 MVA of power from MP3 to MP2 will reduce the margin between the minimum switchyard voltage and the minimum acceptable switchyard voltage. However, the extra loading imposed by the MP2 safe shutdown loads on the MP3 power supplies is within the MP3 capability to simultaneously supply the worst case MP3 normal and accident loads. As a result, the response of the plant and the operators following an accident will not be significantly different. There will be no adverse effect on accident mitigation equipment and no new failure modes will be introduced. Therefore, the activity will not create the possibility of a new or different kind of accident from any previously evaluated.

MP3 USQ 2

Maintaining breaker 13T and its associated disconnect switches open to ensure separation and independence of the two MP2 offsite sources is a change to the normal configuration of MP3. The new configuration involves the opening of a breaker. It does not involve any other physical changes to the plant or the operating methodology. This activity does reduce the diversity of offsite power supplies for the MP3 NSST, which increases the potential for a loss of offsite power at MP3. However, the plant response to a loss of offsite power because breaker 13T is open will not be significantly different from any previously analyzed loss of offsite power. Therefore, opening breaker 13T and its associated disconnect switches will not create the possibility of a new or different kind of accident from any previously evaluated.

MP3 USQ 3

The connection of an additional 3 MVA load for MP2 onto the MP3 electrical distribution system will increase the short circuit fault levels on the MP3 electrical distribution system. In the worst case, the short circuit fault could lead to a loss of one MP3 train of safety related equipment. However, the safety related equipment in the other train would remain capable of mitigating the event. The loss of a train of safety-related equipment is an analyzed event. Therefore, this activity will not introduce the possibility of a new or different kind of accident than any previously evaluated.

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

MP3 USQ 1

The MP3 to MP2 4160 Volt cross-tie will impose up to an additional 3 MVA load on MP3. As a result, the minimum acceptable switchyard voltage for MP3 will increase from 334 kV to 337 kV. The margin between the minimum switchyard voltage (345 kV) and the minimum acceptable switchyard voltage will decrease from 11 kV to 8 kV. This is an improvement for MP2. However, it is a reduction in the operating margin for MP3.

The analysis performed to support the new design shows that, with a minimum voltage of 337 kV in the switchyard, MP3 will remain connected to offsite power. The minimum voltage protection assures that acceptable starting and running voltages are present for the safety systems that may be required to operate should a loss of coolant accident (limiting design basis event) occur at MP3 while MP2 is being supplied with the safe shutdown load of 3 MVA from either the MP3 RSST or the MP3 NSST. Thus, the MP3 offsite connection will not be tripped when offsite power is available.

The extra loading imposed by the MP2 safe shutdown loads on the MP3 power supplies is within the MP3 capability to simultaneously supply the worst case MP3 normal and accident loads. As a result, the response of the plant and the operators following an accident will not be significantly different. There will be no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, this activity does not involve a significant reduction in a margin of safety.

MP3 USQ 2

The proposed design change will allow the MP3 RSST (or the MP3 NSST, if the MP3 RSST is unavailable) to provide the alternate offsite source for MP2 to meet GDC 17 requirements. If the MP3 RSST is not available, MP2 will need to credit the MP3 NSST as the alternate offsite power supply for GDC 17 compliance. This will require the 13T breaker to be open to meet GDC 17 requirements. As a result, the probability of a loss of offsite power at MP3 will increase.

A loss of offsite power at MP3 due to a fault in the offsite distribution network is a low probability event. When combined with the expected frequency of removing the MP3 RSST from service of once per MP3 refueling outage, the probability of a loss of offsite power at MP3 when shut down as a result of breaker 13T being open is low. In addition, the MP3 shutdown risk program will evaluate the impact of removing the MP3 RSST from service, and plan accordingly. This will ensure the MP3 RSST will be removed from service when the shutdown risk is determined to be acceptably low. Also, at least one MP3 EDG will be available, as required by Technical Specifications, to supply power to necessary loads if offsite power is lost.

The plant response to the design basis accidents will not change with breaker 13T open, and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, this activity does not involve a significant reduction in a margin of safety.

MP3 USQ 3

The connection of an additional 3 MVA load for MP2 onto the MP3 electrical distribution system will increase the short circuit fault levels on the MP3 electrical distribution system. If worst case conditions are established, the additional contribution from the MP2 loads would increase the fault level to above the switchgear ratings. The worst case short circuit conditions occur with MP3 at full power, the switchyard at its maximum voltage, an MP3 EDG paralleled to the bus for surveillance testing, and the MP3 to MP2 cross-tie supplying 3 MVA of power to MP2.

Although the fault levels could exceed the MP3 switchgear and circuit breaker ratings, it is not expected that the worst case conditions will be frequently established. For the majority of scenarios, this situation can be avoided because the new design allows the 3 MVA of power for MP2 to be supplied by either MP3 bus 34A or 34B. Procedural guidance will be provided to select the MP3 bus (34A or 34B) to supply power to MP2 that powers the opposite train from the train associated with the MP3 EDG to be tested. Even if the worst case plant configuration cannot be avoided, there is a low probability of event occurrence given the low probability of a bus fault coincident with the short time that the MP3 EDG would be paralleled to the system. In addition, the event would only affect one MP3 train of safety related equipment. The other train will be available to function as assumed to mitigate any accidents that may occur.

The response of the plant and the operators following a design basis event will not be significantly different when using the MP3 to MP2 4160 Volt cross-tie. The accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, this activity 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut

NRC Section Chief: James W. Clifford.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: November 18, 1999, as supplemented August 7, 2000.

Description of amendment request: The proposed amendment to the Kewaunee Nuclear Power Plant Technical Specifications requests approval to increase the allowable number of spent fuel assemblies stored in the spent fuel pools.

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. In the analysis of the safety issues concerning the expanded pool canal storage capacity, the following previously postulated accident scenarios have been considered:

a. A spent fuel assembly drop in the Spent Fuel Pool
b. Loss of Spent Fuel Pool cooling flow
c. A seismic event

The probability that any of the accidents in the above list can occur is not significantly increased by the modification. The probabilities of a seismic event or loss of SFP cooling flow are not influenced by the proposed changes. The probability of an accidental fuel assembly drop is primarily influenced by the methods used to lift and move the fuel. The method of handling fuel during the loading of the canal racks will be the same as current fuel handling methods, since the same equipment (*i.e.*, Spent Fuel Handling Crane) and the same procedural guidance will be used. Since the methods used to move fuel during normal operations remain nearly the same as those used previously, there is no significant increase in the probability of an accident.

Accordingly, the proposed modification does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the SFP have been re-evaluated for the proposed change and were determined to be bounded by the existing analysis. The results show that the postulated accident of a fuel assembly striking the top of the storage racks will not distort the racks sufficiently to impair their functionality. The resulting structural damage to a falling assembly and/or a stored assembly has been determined to remain unchanged. The minimum subcriticality margin, K_{eff} less than or equal to 0.95, will be maintained. The structural damage to the SFP structure, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks remains unchanged. The resulting structural damage to these items subsequent to this event is not influenced by the proposed changes.

The consequences of a loss of SFP cooling have been evaluated and found to have no increase. The concern with this accident is a reduction of SFP water inventory from bulk pool boiling resulting in uncovering fuel assemblies. This situation would lead to fuel failure and subsequent significant increase in offsite dose. Loss of SFP cooling at Kewaunee is mitigated by ensuring that a sufficient time period exists between the loss of forced cooling and uncovering fuel. This period of time is compared against a reasonable period to re-establish cooling or supply an alternative water source (such as service water, makeup water, or fire protection system water). This evaluation included the determination of the time to boil. The time to boil represents the onset of loss of pool water inventory and is commonly used as a gage for establishing the comparison of consequences before and after a refueling project. The heat up rate in the SFP is a nearly linear function of the fuel decay heat load. The fuel decay heat load will increase subsequent to the proposed changes because of the increase in the number of stored assemblies. The heat up rate established for the limiting heat load conditions prior to the addition of canal racking was 10.8°F per hour. This would result in the pool temperature increasing from the maximum design temperature of 150°F to boiling in a period of 5.7 hours. The heat up rate established for the limiting heat load conditions subsequent to the proposed changes has been determined as 7.5°F per hour. This would result in the pool temperature increasing from the maximum design temperature of 150°F to boiling in a period of 8.4 hours.

This time to boil comparison was made for limiting heat load conditions. However, the end of this period of time does not represent the onset of any significant increase in offsite doses. As stated above, this consequence would result after fuel is uncovered through unchecked boiling and the resulting water level drop of approximately 25 feet to the top of the fuel storage racks. This depth is conservative, since the top of active fuel is below this level. Subsequent to the proposed changes under limiting heat loads the maximum boil-off rate will decrease to 40.9 gpm. This will result in the time lapse between loss of SFP cooling and the uncovering of the racks to increase to 48.5 hours.

As stated above, subsequent to racking the canal, the time to boil after loss of forced cooling in the most severe scenario was 8.4 hours. The ensuing rate of evaporative loss would not result in the fuel being uncovered until after an additional 40.1 hours. The margin concerning time to boil will increase due to the proposed modification. This increased margin is due to the conservatism of the assumptions previously used to determine the heat load for this condition (*i.e.* the assumption that all fuel assemblies in the core were transferred to the pool instantaneously). Also, another factor is that the required incore hold time prior to fuel movement will increase from 100 hours to 148 hours. Therefore, the calculated time to boil in this most severe scenario will increase subsequent to the proposed modification. In the unlikely event that all pool cooling is lost, sufficient time will continue to be available subsequent to the proposed changes for the operators to provide alternate means of cooling (*i.e.*, service water, makeup water, or fire protection system water) before fuel is uncovered. Therefore, the proposed changes represent no increase in the consequences of a loss of pool cooling.

The consequences of a design basis seismic event are not increased. The consequences of this accident are evaluated on the basis of subsequent fuel damage or compromise of the fuel storage or building configurations leading to radiological or criticality concerns. The canal racks have been analyzed and found to properly function during seismic motion. The stored fuel has been determined to remain intact and the storage racks maintain the fuel and fixed poison configurations subsequent to a seismic event. The structural capability of the pool and liner will not be exceeded under the appropriate combinations of dead weight, thermal, and seismic loads. The SFP structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks, stored fuel, and pool coolant. Thus, the consequences of a seismic event are not increased.

Therefore, it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

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

During and following project implementation, the spent fuel must be safely stored, meeting the requirements 10 CFR 20 and 10 CFR 100. The installation activities as well as the use of the canal racks must be evaluated for the possibility of creating a new or different kind of accident from any previously analyzed.

This evaluation was done by reviewing the differences between the analyses and assumptions associated with the proposed storage configuration (existing racks plus canal racks) with those of the existing storage configuration. The accident scenarios associated with the existing racks were then reviewed to determine if the possibility of a new type of accident would be introduced by the implementation of the proposed modification. Also, the installation activities were evaluated.

Due to the proposed changes, the following events were considered as the only events which might represent a new or different kind of accident:

a. An accidental drop of a rack module into the pool during construction activity.

b. Fuel assembly mispositioning accident in the canal.

A construction accident resulting in a rack drop is an unlikely event. A rack-lifting rig will be used to lift and suspend the racks using the existing Fuel Handling Building crane. The crane, hoists and lifting rig have been or will be designed, tested and inspected to ensure that they will properly lift and transport the new racks. The postulated rack drop event is commonly referred to as a ''heavy load drop'' over the pools. Heavy loads will not be allowed to travel over any racks containing fuel assemblies. The area of concern represented by this event is that the pool structure will be compromised leading to loss of moderator/coolant. The question of a new or different type of event is answered by determining whether heavy load drops of this type have been considered previously. The last phase of the re-racking of the SFPs at KNP was completed in 1987. The rack modules transported during installation of the pool re-racking project were significantly larger than those associated with the proposed canal rack project. Also, the KNP Technical Specifications state that ''Placement of additional fuel storage racks is permitted, however, these racks must not traverse directly above spent fuel stored in the pools''. All movements of heavy loads will be strictly controlled by procedure, will follow approved safe load paths, and will be performed by qualified personnel. Therefore, the rack drop does not represent a new or different kind of accident.

Fuel assembly mispositioning in the canal is an unlikely event since assembly placement will be administratively controlled. The administrative controls will include positive, visual identification of each fuel assembly to be stored in the canal racks. The mispositioning event for the canal racks represents a change from the previously analyzed condition, since Kewaunee currently has no restriction on fuel storage.

However, the mispositioning event in the canal does not represent a new or different kind of accident. A fuel assembly mispositioning event has been previously analyzed for the existing racks and was found to be acceptable. The new mispositioning event for the canal racks was evaluated using similar techniques with similar acceptance criteria and was also shown to be acceptable. Therefore, since both events represent the mispositioning of a fuel assembly, the mispositioning of an assembly in the canal racks is not considered to represent a new or different kind of accident.

Under the extremely unlikely conditions of a mispositioning event involving the placement of a fresh (un-burned) fuel assembly in the canal racks, the boron in the pool water will maintain the required subcriticality margin. Only a small fraction of the minimum pool boron concentration required by the KNPP Technical Specifications would be required for this condition. Under the postulated accident condition of a total loss of boron from the pool water, proper subcriticality margin would exist. The multiple accident scenario of the extremely unlikely mispositioning event combined with a postulated total loss of pool boron is a non-credible condition. Therefore, although credit is taken for a small amount of boron in the pool water for one accident condition, a decrease of pool boron concentration is not a new or different kind of accident from any previously analyzed.

The proposed change does not involve the modification of any equipment credited in the mitigation of the design basis accidents. The proposed change meets all applicable requirements for the safe storage of spent fuel at KNP. Therefore, the potential for a new or previously unanalyzed accident is not created by the implementation of this modification.

The refueling canal does not have floor drains. Therefore, draining the refueling canal through a floor drain is not a postulated event.

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

The function of the SFP is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with the SFP.

WPSC has addressed the safety issues related to the expanded pool storage capacity in the following areas:

a. Material, mechanical and structural considerations
b. Nuclear criticality
c. Thermal-hydraulic and pool cooling

The mechanical, material, and structural designs of the new racks have been reviewed in accordance with the applicable provisions of the NRC Guidance entitled, ''Review and Acceptance of Spent Fuel Storage and Handling Applications''. The rack materials used are compatible with the spent fuel assemblies and the SFP environment. The design of the new racks preserves the proper margin of safety during abnormal loads such as a dropped assembly and tensile loads from a stuck assembly. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration.

The methodology used in the criticality analysis of the expanded SFP meets the appropriate NRC guidelines and ANSI standards. The margin of safety for subcriticality is maintained by having the neutron multiplication factor equal to, or less than, 0.95 under all accident conditions, including uncertainties. This criterion is the same as that used previously to establish criticality safety evaluation acceptance and remains satisfied for all analyzed accidents. Therefore, the accepted margin of safety remains the same.

The thermal-hydraulic and cooling evaluation of the pool demonstrated that the pool can be maintained below the specified thermal limits under all required design conditions. The pool temperature will not exceed 140°F during the failure of a cooling pump under normal offload conditions. The pool temperature will remain below the maximum design temperature of 150°F under maximum heat load conditions. The maximum local water temperature in the hot channel will remain below the boiling point. The fuel will not undergo any significant heat up after an accidental drop of a fuel assembly on top of the rack blocking the flow path. Following a loss of cooling to the pool sufficient time exists (48.5 hours for the limiting heat load) for the operators to intervene and line up alternate cooling paths and the means of inventory makeup before the onset of pool boiling. The thermal limit specified for the evaluations performed to support the proposed change is the same as was used in the previous evaluation. Therefore, the accepted margin of safety remains the same.

Thus, it is concluded that the changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: Claudia M. Craig.

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station (VCSNS), Unit No. 1., Fairfield County, South Carolina

Date of amendment request: September 14, 2000.

Description of amendment request: The proposed changes will delete Technical Specification (TS) Section 6.2.3 and revise TS Sections 6.3.1 and 6.5.2.8 to remove the references to the Independent Safety Engineering Group (ISEG).

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:

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the significant Hazards Criteria of 10 CFR 50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

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

The proposed amendment is a programmatic and administrative change to delete the ISEG. The requirement for an ISEG is located within the Administrative Section of TS (Section 6.0). The ISEG functions in an oversight role and, as such, performs no actions that would affect any precursors to any accident previously evaluated. This change does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. However, the independent oversight functions and qualification requirements stated in TS are being performed by qualified personnel in other departments of the VCSNS organization. This assures that the functions and qualification requirements delineated for the ISEG are being met. Because the design of the facility and system operating parameters are not being changed, and the oversight functions are performed by other departments, the proposed amendment does not involve an increase in the probability or consequences of any accident previously evaluated.

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

The proposed amendment to delete the ISEG is a programmatic and administrative change. There are no physical alterations to safety-related systems and no changes to the functions of any safety-related systems. The independent oversight functions assigned to the ISEG are addressed by other departments within the VCSNS organization. Personnel in those departments meet the qualifications required of the ISEG. This assures that the oversight functions and qualification requirements delineated for the ISEG are being met. Oversight functions that are performed do not in themselves lead to activities that are considered as accident precursors or initiators. Because this change does not alter the design of the facility and system operating parameters are not being changed, and the oversight functions and qualification requirements of the ISEG are being met by other departments, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment is a programmatic and administrative change to delete the ISEG. This proposed change involves utilizing appropriately trained and qualified personnel in other departments to perform the oversight functions previously assigned to the ISEG. No physical alterations to safety-related systems and no changes to the functions of any safety-related systems are being made. The use of personnel in other departments to perform the oversight functions of the ISEG will provide assurance that plant operations continue to be conducted in a safe manner. These oversight functions will be performed by personnel that maintain their independence from line operations. Because the design of the facility and system operating parameters are not being changed, and the oversight functions of the ISEG are appropriately addressed by other departments and will continue to be performed, the proposed amendment does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: September 6, 2000.

Brief description of amendments: The proposed amendment would change Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, Technical Specification (TS) 5.5.9, ''Steam Generator Tube Surveillance Program,'' to permit installation of a laser welded tube sleeve as an alternative to plugging defective steam generator tubes. TS 5.5.10, ''Steam Generator Tube Inspection Report,'' would be revised to address reporting requirements for repaired tubes. Also, an editorial correction is proposed to Table 5.5–2.

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

Response: No.

The tubesheet and/or tube support plate intersection laser welded sleeve configurations [were] designed and analyzed in accordance with the requirements of the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code]. Fatigue and stress analyses of the sleeved tube assemblies produced acceptable results. Additionally, mechanical testing for the full length tubesheet sleeves has shown that the structural strength of Alloy 690 sleeves under normal, faulted, and upset conditions is within acceptable limits. Leakage testing for these same ¾ inch tube sleeves has demonstrated that primary to secondary leakage is not expected during any plant conditions. Similar results are anticipated for the lower joints of elevated tubesheet sleeves. Confirmatory mechanical and leak testing will be conducted supporting the installation of elevated tubesheet sleeves at CPSES, Unit 1.

The hypothetical consequences of failure of a sleeve would be bounded by the current steam generator tube rupture analysis included in the Comanche Peak Steam Electric Station (CPSES) Final Safety Analysis Report (FSAR). Due to the slight reduction diameter caused by the sleeve wall thickness, it is expected that primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system. Combinations of tubesheet sleeves and tube support plate sleeves would reduce the primary fluid flow through the sleeved tube assembly due to the series of diameter reductions the fluid would have to pass on its way to the break area. The overall effect would be reduced steam generator tube rupture release rates. The proposed Technical Specification change to support the installation of Alloy 690 laser welded sleeves does not adversely impact any other previously evaluated design basis accident or the results of (LOCA) [loss-of-coolant accident] and non-LOCA accident analyses for the current Technical Specification minimum RCS [reactor coolant system] flow rate.

Conformance of the sleeve design with the applicable sections of the ASME Code and the successful completion of the leakage and mechanical tests (for the lower sleeve joint for the elevated tubesheet sleeves (ETS), support the conclusion that the installation of laser welded tube sleeves will not increase the probability or consequences of an accident previously evaluated. Depending upon the break location for a postulated steam generator tube rupture event, implementation of tube sleeving could act to reduce the radiological consequences to the public due to reduced flow rate through a sleeved tube compared [to] a non-sleeved tube based on the restriction afforded by the sleeve wall thickness.

The editorial correction [to] Technical Specification (TS) Table 5.5–2 is typographical in nature and does not require additional evaluation. Therefore, the proposed changes 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.

Implementation of the laser welded sleeving (LWS) will not introduce significant or adverse changes to the plant design basis. Stress and fatigue analysis of the repair has shown the ASME Code minimum stress values are not exceeded. Implementation of laser welded sleeving restores the overall tube bundle structural and leakage integrity to a level consistent to that of the originally supplied tubing during all plant conditions. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. Finally, through the results obtained from the extensive testing and qualification program, the possibility of a common-mode failure, such as multiple simultaneous steam generator tube failures, is not credible. Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The editorial correction [to] TS Table 5.5–2 is typographical in nature and does not require additional evaluation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The laser welded sleeving repair of degraded steam generated tubes as identified in References 1 and 2 was shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The design of the full length tubesheet sleeve lower joints for the ¾ inch tube sleeves (roll-first installation sequence) were verified by testing to preclude pullout and primary-to-secondary leakage during normal and postulated accident conditions. The qualification of the lower joint of the TSS [tube support sleeve], ETS and the full length tubesheet sleeves (FLTS) (roll-last installation sequence) will be confirmed at the time of the sleeving outage. Since the installed sleeve represents a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with the sleeves installed. The portions of the installed sleeves assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the recommendations of Regulatory Guide 1.83, Rev. 1. The portion of the tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The areas of the sleeved tube assembly which require inspection are defined in WCAP–13698, Rev. 3.

EPRI [Electric Power Research Institute] qualified eddy current techniques will be used for the detection of tube degradation in ¾ inch laser welded sleeved tubes. Alternate inspection techniques may be used as they become available, as long as it can be demonstrated that the technique used provides the same degree or greater degree of inspection rigor.

The effect of sleeving on the design transients and accident analyses were reviewed and found to remain valid up to the level of steam generator tube plugging consistent with the minimum reactor flow rate as specified in Technical Specification 3.4.1. Continued compliance with the RCS flow limits of Technical Specification 3.4.1 is assured through precision flow measurements.

Because all relevant safety analyses were reviewed and found to remain valid, and because the appropriate design margins are maintained through compliance with the relevant ASME Code requirements, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

The editorial correction [to] TS Table 5.5–2 is typographical in nature and does not require additional evaluation. The confirming modifications to the reporting requirements of TS 5.6.10 are administrative only. Therefore, these 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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 15, 2000.

Brief description of amendments: The proposed change replaces the general references currently provided in Technical Specification 5.6.6 for determining the reactor coolant system pressure and temperature limits with the requirement that the Pressure/Temperature Limits and Low Temperature Overpressure Protection System Setpoints shall not be revised without prior U.S. Nuclear Regulatory Commission approval.

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 addition of this requirement to the Technical Specifications is administrative and has no impact on accident initiation or mitigation. Therefore, the proposed changes 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 addition of this requirement to the Technical Specifications is administrative and cannot initiate an accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The addition of this requirement to the Technical Specifications is administrative and has no impact on accident initiation or mitigation. Therefore the proposed change does not involve a reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: September 22, 2000.

Description of amendment request: The proposed changes would remove obsolete license conditions from the Operating Licenses (OLs) and implement associated changes to the Technical Specifications (TS) and Bases. The proposed changes include removal of license conditions associated with completed facility modifications (including the Steam Generator (SG) Repair Program, as well as support modifications related to Leak-Before-Break Technology); removal of superseded license conditions (addressing security); relocation of secondary water chemistry monitoring program requirements into the TS; removal of expired license conditions and TS (addressing service water piping restoration); and editorial changes.

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

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed administrative change to the Operating Licenses, DPR–32 and DPR–37, and the associated Technical Specifications for Surry Units 1 and 2 and determined that a significant hazards consideration is not involved. The proposed administrative change to the Surry Operating Licenses and associated Technical Specifications makes minor editorial corrections, relocates one license condition to Appendix A of the license, and removes outdated, superceded or otherwise non-applicable license conditions and Technical Specifications requirements and provides a license document that is directly applicable to the current plant licensing and design bases. There is no safety significance associated with this proposed change since the change does not alter any currently applicable Operating License requirements. Accordingly, the current Surry licensing and design bases are unchanged. In support of this conclusion, the following evaluation is provided.

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

The proposed change is administrative (and in part editorial) in nature and neither station operations nor design are affected by the change. The removal of license conditions and associated Technical Specifications regarding superceded (OL Sections 3.H and 3.L) or expired (OL Section 3.0, TS Table 3.7–2, and TS 3.14) requirements has no impact on plant operations since the requirements no longer have a legitimate means of being applied. The relocation within the Operating License of the requirement to have a secondary water chemistry monitoring program (OL Section 3.K) to new Section 6.4.P of the Technical Specifications does not alter the program or its implementation. The impact of the replacement SGs at Surry and the re-design of the SG and RCP [reactor coolant pump] supports on the previously evaluated accidents was performed, approved and documented by the issuance of the license amendments for OL Sections 3. G and 3.M. Removal of these license conditions which refer to completed work and a design and licensing bases that is documented in the UFSAR [updated final safety analysis report] also does not alter station operation or the design of the affected components. The proposed change is within the current design and licensing bases of the facility. This change does not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. This change does not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed change to the Surry Operating Licenses and Technical Specifications does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

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

The proposed change is administrative (and in part editorial) in nature. The license conditions and Technical Specifications that are being removed or relocated by this proposed change do not impact station operations or station equipment in any manner. The proposed change does not involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients that has not been previously analyzed. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed change to the Surry Operating Licenses and Technical Specifications does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Criterion 3—The proposed license amendment does not involve a significant reduction in a margin of safety.

The proposed change is administrative (and in part editorial) in nature and neither station operations nor design are affected by the change. Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. Since station operations are not affected by the proposed change, the change does not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed change to the Surry Operating Licenses and Technical Specifications does not involve a reduction in any margin of safety described in the bases of the Technical Specifications.

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

Attorney for licensee: Donald P. Irwin, Esq., Hinton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch.

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.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment requests: September 22, 2000 (PCN–520).

Brief Description of amendment requests: The proposed amendments would revise the San Onofre Units 2 and 3 technical specifications (TSs) applicable in shutdown MODES relating to positive reactivity additions. For a summary of specific proposed TS changes, see Tables 1 and 2 of the licensee's application dated September 22, 2000. The licensee's proposal generally conforms to industry Technical Specification Task Force (TSTF) TSTF–286, Revision 2.

*Date of publication of individual notice in **Federal Register:*** October 13, 2000 (65 FR 60984).

Expiration date of individual notice: November 13, 2000.

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 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

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

Date of application for amendment: July 14, 2000.

Brief description of amendment: The amendment slightly reduces the required minimum reactor cavity water level.

Date of issuance: October 12, 2000.

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

Amendment No.: 133.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 23, 2000 (65 FR 51347).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 2000.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: November 30, 1999, as supplemented August 11, and September 14, 2000.

Brief description of amendment: The amendment revised the test standard for laboratory testing of activated charcoal to test in accordance with the ASTM (American Society for Testing and Materials) D3803–1989 standard in response to Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of issuance: October 18, 2000.

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

Amendment No.: 226.

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

Date of initial notice in ***Federal Register:*** March 8, 2000 (65 FR 12287).

The August 11, and September 14, 2000, letters provided clarifying information that did not change the initial no significant hazards consideration determination and did not expand the scope of the original notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 18, 2000.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: May 31, 2000.

Brief description of amendments: The amendments deleted the requirements related to the shorting links from Technical Specification (TS) Sections 3/4.3.1, ''Reactor Protection System Instrumentation;'' 3/4.9.2, ''Refueling Operations Instrumentation;'' and 3/4.10.3, ''Shutdown Margin Demonstrations;'' and increased the required signal-to-noise ratio for the source range monitor in TS Sections 3/4.9.2 and 3/4.3.7.6, ''Source Range Monitors.''

Date of issuance: October 10, 2000.

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

Amendment Nos.: 142 and 128.

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

Date of initial notice in ***Federal Register:*** August 9, 2000 (65 FR 48745).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 10, 2000.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: May 1, 2000, as supplemented by letter dated August 11, 2000.

Brief description of amendments: The amendments revise Technical Specification 3/4.8.1, ''A. C. Sources—Operating,'' to permit functional testing of the emergency diesel generators to be performed during power operation.

Date of issuance: October 16, 2000.

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

Amendment Nos.: 143 and 129.

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

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37423).

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 16, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 30, 1999; as supplemented on August 3, 2000.

Brief description of amendments: The amendments revised the technical specifications to: (1) Remove the Main Steamline Radiation Monitor (MSLRM) scram and main steamline isolation functions, and (2) add a new requirement for the MSLRM mechanical vacuum pump trip function.

Date of issuance: October 13, 2000.

Effective date: Immediately, to be implemented within 120 days.

Amendment Nos.: 196 and 192.

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

Date of initial notice in ***Federal Register:*** August 9, 2000 (65 FR 48747). The August 3, 2000, supplement provided additional information that did not change the scope of the original proposed no significant hazards findings.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 13, 2000.

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Inc., Docket No. 50–003, Indian Point Nuclear Generating Station, Unit 1

Date of amendment request: February 14, 2000.

Brief description of amendment: The amendment change to Indian Point Nuclear Generating Station, Unit 1, revised Technical Specification Sections 2.10.2, 3.1.2, 3.2.1, 4.1.8.1, and 4.1.8.1.b. Specifically, Sections 3.1.2, 3.2.1, and 4.1.8.1.b, make organizational title changes that are administrative in nature and reflect a streamlining of the Consolidated Edison Company of New York, Inc.''s management structure, Section 4.1.8.1 is changed to reflect the renumbering of 10 CFR Part 20, and a footnote was moved from Section 2.11 to Section 2.10.2 to improve the clarity of the Technical Specification since it pertains to text in subsection 2.10.2.4.

Date of issuance: October 12, 2000.

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

Amendment No: 48.

Facility Operating License No. DPR–5: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17912).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 2000.

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: November 22, 1999.

Brief description of amendment: The amendment authorized a change to the Pilgrim Nuclear Power Station Updated Final Safety Analysis Report (UFSAR). The change involves the use of containment overpressure to ensure sufficient net positive suction head for the emergency core cooling system pumps following a loss-of-coolant accident.

Date of issuance: October 6, 2000.

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

Amendment No.: 185.

Facility Operating License No. DPR–35: Amendment revised the UFSAR.

Date of initial notice in ***Federal Register:*** March 8, 2000 (65 FR 12290).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 2000.

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: June 16, 1999, as supplemented on May 4 and July 10, 2000.

Brief description of amendment: The requested changes incorporate Technical Specification (TS) changes to comply with the operating requirements derived from GE Report, NEDO–21231, ''Banked Position Withdrawal Sequence (BPWS)'', dated January 1977, as referenced in NEDE–24011–P–A. The TS changes incorporate Specifications and Actions based upon the plant-specific CRDA and BPWS for 20% rated thermal power (RTP) and 280 cal/gram peak fuel enthalpy. The TS changes also include changes to the control rod worth limits to resolve Licensee Event Report (LER) 98–006–00, dated April 30, 1998, and its supplemental LER 98–006–01, dated August 27, 1988.

Date of issuance: October 16, 2000.

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

Amendment No.: 186.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51350).

The May 4 and July 10, 2000, letters provided clarifying information that did not change the 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 amendment is contained in a Safety Evaluation dated October 16, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 18, 1999, as supplemented by letters dated June 29, July 19, and August 9, 2000.

Brief description of amendment: The amendment revised Technical Specification 4.4.5, ''Steam Generators,'' to note that the requirements for inservice inspection do not apply during the steam generator replacement outage (2R14), to revise the requirement for tube inspection to mean an inspection from tube end (cold leg side) to tube end (hot leg side), to delete inspection requirements associated with steam generator tube sleeving and repair limits, to revise the preservice inspection requirements on when the hydrostatic test and the eddy current inspection of the tubes would be performed, and to revise the reporting frequency of the results of steam generator tube inspections to within 12 months following completion of the inservice inspection. Related changes to the Bases were also made.

Date of issuance: October 4, 2000.

Effective date: As of the date of issuance to be implemented prior to startup from the 2R14 refueling outage.

Amendment No.: 223.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 23, 2000 (65 FR 9005). The application was renoticed on August 23, 2000 (65 FR 51353).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of application for amendment: September 1, 2000.

Brief description of amendment: The amendment revised certain 18-month surveillance requirements in the technical specifications by eliminating the condition that testing be conducted during shutdown, or during cold shutdown or refueling mode. The affected systems are emergency core cooling system, containment depressurization and cooling system, chemical addition system, and containment isolation valve system.

Date of issuance: October 13, 2000.

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

Amendment No: 118.

Facility Operating License No. NPF–73. Amendment revised the technical specifications.

Date of initial notice in ***Federal Register:*** September 12, 2000 (65 FR 55056).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 13, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of application for amendment: November 1, 1999, and as supplemented by letter dated May 22, 2000.

Brief description of amendment: Consistent with the guidance of Generic Letter 99–02, ''Laboratory Testing of Nuclear Grade Activated Charcoal,'' this amendment modifies existing Technical Specification 5.5.7, ''Ventilation Filter Testing Program (VFTP),'' to reference ASTM D3803—1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.'' The amendment also incorporates the suggested safety factor for charcoal filter efficiency regarding methyl iodide penetration.

Date of issuance: October 12, 2000.

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

Amendment No.: 117.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70088).

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 2000.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: May 30, 2000.

Brief description of amendments: The amendments would make changes to several Technical Specifications to reflect implementation of the revised 10 CFR Part 20, ''Standards for Protection Against Radiation.''

Date of issuance: October 10, 2000.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 245 and 226.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 9, 2000 (65 FR 48752).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 10, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 28, 2000.

Description of amendment request: The amendment implements the Relaxed Axial Offset Control (RAOC) operating strategy in support of the use of upgraded Westinghouse fuel with Intermediate Flow Mixers.

Date of issuance: October 6, 2000.

Effective date: As of its date of issuance, and shall be implemented at commencement of Cycle 8 operation (scheduled for November 2000).

Amendment No.: 76.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 31, 2000 (65 FR 34747).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 19, 2000.

Brief description of amendment: This amendment modifies Technical Specification (TS) 3.7.1.5, ''Plant System—Main Steam Line Isolation Valves.'' Specifically, the change removes the requirement to perform partial stroke testing of the main steam line isolation valves during power operation, modifies the TS wording for clarity, combines two surveillance requirements into one, and modifies the associated TS Bases for consistency.

Date of issuance: October 19, 2000.

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

Amendment No.: 185.

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

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51360).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 19, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 15, 1999, as supplemented August 24, 2000.

Brief description of amendments: The amendments removed TS Table 3.6.3–1 ''Primary Containment Isolation Valves,'' and references to the TS Table from the TSs and relocated the information from the TS Table to the Technical Requirements Manual. In addition, an administrative change was made which deleted references to TS Tables 3.6.5.2.1–1 and 3.6.5.2.2–1.

Date of issuance: October 18, 2000.

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

Amendment Nos.: 146 and 107.

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

Date of initial notice in ***Federal Register:*** March 8, 2000 (65 FR 12294).

The August 24, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** Notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 18, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 3, 2000.

Brief description of amendment: The amendment revises Technical Specification Table 3.2–7 by changing the reactor water level setpoint for the anticipated transient without scram, the recirculation pump trip function, and the alternate rod insertion function.

Date of issuance: October 10, 2000.

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

Amendment No.: 264.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15383).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 20, 1999, as supplemented February 4, 2000.

Brief description of amendment: The amendment changes the Main Steam Isolation Valve closure scram trip level setting from ≤10 percent to ≤15 percent valve closure.

Date of issuance: October 10, 2000.

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

Amendment No.: 265.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 9, 2000 (65 FR 6410).

The February 4, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination and did not expand the amendment beyond the scope of the original notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 21, 2000

Brief description of amendment: The amendment revises the Technical Specifications to remove the Control Room Emergency Air Treatment System Actuation Instrumentation operability during Modes 5 and 6 except during core alterations and fuel movement based on the control room dose calculations.

Date of issuance: October 10, 2000.

Effective date: October 10, 2000.

Amendment No.: 78.

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

Date of initial notice in ***Federal Register:*** August 9, 2000 (65 FR 48757).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 2000.

No significant hazards consideration comments received: No.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of application for amendments: February 23, 2000, as supplemented July 7, 2000.

Brief description of amendments: The amendments revise the licensing basis for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 regarding the methodology for measuring the reactivity worth of control element assembly (CEA) groups during low power physics testing following a refueling. The amendments allow measuring the worth of approximately three-fourths of the full-length CEA groups each refueling cycle rather than the present methodology, which measures the worth of all full-length CEA groups each refueling cycle.

Date of issuance: October 10, 2000.

Effective date: October 10, 2000. Implementation includes incorporation of the changes into the Updated Final Safety Analysis Report (UFSAR) at the next update of the UFSAR in accordance with the schedule in 10 CFR 50.71(e).

Amendment Nos.: Unit 2–173; Unit 3–164.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments authorized revision of the UFSAR Section 4.2.1.5.2, CEA Performance Testing.

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15385).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 10, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 8, 2000.

Brief description of amendment: The amendment revises the Callaway Technical Specifications (TS) to annotate the frequency for Surveillance Requirement (SR) 3.5.2.5 that verification of the automatic closure function of the residual heat removal pump suction Valve BNHV8812A shall be performed prior to startup from the first shutdown to Mode 5 (cold shutdown) occurring after September 8, 2000, but no later than June 1, 2001. The next refueling outage is scheduled for April 2001. This amendment defers the test of the automatic closure function until the next plant shut down to cold shutdown.

Date of issuance: October 6, 2000.

Effective date: October 6, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 140.

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

Public comments requested as to proposed no significant hazards consideration: Yes (65 FR 56943 dated September 20, 2000). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 20, 2000, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated October 6, 2000.

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

NRC Section Chief: Stephen Dembek.

Dated at Rockville, Maryland, this 25th day of October 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–27938 Filed 10–31–00; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–24697; 812–11786]

PMC Capital, Inc., et al.; Notice of Application

October 25, 2000.

AGENCY: Securities and Exchange Commission (''Commission'' or ''SEC'').

ACTION: Notice of application for an order under sections 6(c) and 57(c) of the Investment Company Act of 1940 (the ''Act'') for an exemption from sections 57(a)(1) and 57(a)(2) of the Act, and under section 57(i) of the Act and rule 17d–1 under the Act authorizing certain joint transactions otherwise prohibited by section 57(a)(4) of the Act.

SUMMARY OF APPLICATION: Applicants request an order that would permit (1) a business development company (''BDC'') to engage in a loan origination agreement with an affiliated real estate investment trust, (2) investment management agreements between subsidiaries of the BDC and the real estate investment trust, and (3) the establishment of special purpose entities owned by the BDC and the real estate investment trust to engage in joint loan securitizations. The requested order would supersede an existing order.

APPLICANTS: PMC Capital, Inc. (''PMC''), PMC Commercial Trust (the ''REIT''), PMC Advisers, Ltd. (''Advisers''), and PMC Asset Management, Inc. (''Managers'').

FILING DATES: The application was filed on September 9, 1999 and amended on March 29, 2000 and October 24, 2000.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicants with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on November 20, 2000, and should be accompanied by proof of service on applicants, in the form of an affidavit, or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the SEC's Secretary.

ADDRESSES: Secretary, SEC, 450 Fifth Street, N.W., Washington, D.C. 20549. Applicants, 18111 Preston Road, Suite 600, Dallas, Texas 75252.

FOR FURTHER INFORMATION CONTACT: Marilyn Mann, Senior Counsel, at (202) 942–0582, or Mary Kay Frech, Branch Chief, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

SUPPLEMENTARY INFORMATION: The following is a summary of the application. The complete application may be obtained for a fee from the SEC's Public Reference Branch, 450 5th Street, N.W., Washington D.C. 20549–0102 (telephone (202) 942–8090).

Applicants' Representations

1. PMC, a Florida corporation, is a closed-end diversified management investment company. On June 7, 1994, PMC filed notification of its election to operate as a BDC. PMC provides early stage financing and makes available significant managerial assistance to small businesses and receives interest income, loan servicing and other fees