Connecticut Yankee Atomic Power Company (CYAPC, or the licensee) announced permanent cessation of power operations of HNP on December 5, 1996. In accordance with NRC regulations, CYAPC submitted a Post-Shutdown Decommissioning Activities Report (PSDAR) for HNP to the NRC on August 22, 1997. The facility is undergoing active decontamination and dismantlement.

In accordance with 10 CFR 50.82(a)(9), all power reactor licensees must submit an application for termination of their license. The application for termination of license must be accompanied or preceded by an LTP to be submitted for NRC approval. If found acceptable by the NRC staff, the LTP is approved by license amendment, subject to such conditions and limitations as the NRC staff deems appropriate and necessary. CYAPC submitted the proposed LTP for HNP by application dated July 7, 2000. In accordance with 10 CFR 20.1405 and 10 CFR 50.82(a)(9)(iii), the NRC is providing notice to individuals in the vicinity of the site that the NRC is in receipt of the HNP LTP and will accept comments from affected parties. Also, the NRC staff will conduct a public meeting in the vicinity of the HNP site in the near future to discuss the HNP LTP. A separate notice regarding this meeting will be published in the **Federal Register** when specific arrangements for the meeting have been made.

Written comments should be sent to: Chief, Rules and Directives Branch, Division of Administrative Services, Mail Stop T–6 D59, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Comments may be hand-delivered to the NRC at 11545 Rockville Pike, Rockville, Maryland, between 7:45 a.m. and 4:15 p.m. on Federal workdays.

The HNP LTP (ADAMS Accession Number ML003735143) is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, N.W, Washington, DC and is accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Public Electronic Reading Room). The LTP may also be viewed at the CYAPC Web site at *www.connyankee.com.*

For further information, contact: Mr. Louis L. Wheeler, Mail Stop O–7-C2, Project Directorate IV & Decommissioning, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, telephone 301–415–1444, or e-mail dxw@nrc.gov.

Dated at Rockville, Maryland, this 16th day of August 2000.

For the Nuclear Regulatory Commission

Louis L. Wheeler,

Acting Chief, Decommissioning Section, Project Directorate IV & Decommissioning, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 00–21516 Filed 8–22–00; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

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 July 31, 2000, through August 11, 2000. The last biweekly notice was published on August 9, 2000 (65 FR 48744).

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

By September 22, 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, the Gelman Building, 2120 L Street, NW., Washington, DC, 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, the Gelman Building, 2120 L Street, NW., Washington, DC, and 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 amendment request: July 14, 2000

Description of amendment request: The proposed amendment would slightly reduce the required minimum reactor cavity water level.

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 any accident previously evaluated.

The proposed change to the Technical Specifications involves the minimum or reference reactor cavity water level requirement (relative to the reactor pressure vessel [RPV] flange) during refueling operations. Reactor cavity water level can affect the consequences of events that may be postulated to occur during shutdown conditions (including fuel handling operations), namely a fuel handling accident, loss of normal decay heat removal capability, or inadvertent reactor draindown. Such events, however, are caused by equipment failures or human errors. The proposed change has no impact on such failures or errors, particularly their probability of occurrence. Therefore, the proposed change will not significantly increase the probability of a fuel handling accident, loss of decay heat removal, or inadvertent reactor draindown.

With regard to impact on the consequences of postulated events/accidents, the effect of the change on the consequences of a fuel handling accident is minimal. The accident producing the largest number of failed irradiated fuel rods is the drop of an irradiated fuel assembly onto the reactor core when the reactor vessel head is removed (Reference USAR 15.7.4.1.1). Since this event takes place only in the containment and the release associated with this event must be transferred from the containment atmosphere to the secondary containment, the accident which produces the most severe radiological release is a drop of channeled fuel onto unchanneled spent fuel in the fuel storage racks in the fuel building i.e. directly within the secondary containment. The proposed change has no impact on a fuel handling accident in the fuel building. A drop of a fuel bundle on the RPV flange may involve a release of fission products from the dropped fuel bundle, but such a release would be less severe as it would involve much less fuel damage (notwithstanding potentially less pool depth), compared to the drop of a fuel bundle onto the reactor core. It has therefore been determined that lowering the minimum water level from 23 feet (ft) to 22 ft, 8 inches has no significant effect on the consequences of a fuel handling accident.

With respect to a loss of normal decay heat removal capability, or an inadvertent reactor draindown, the change reduces slightly the volume of water required for decay heat removal capability and reactor coolant inventory to mitigate a draindown event. Since the volume change has an insignificant effect on the reactor/pool volume's total available decay heat removal capability (as a backup in the event of a loss of normal decay heat removal capability) and has a negligible effect on the operator's ability to mitigate a draindown event, lowering the minimum specified water level from 23 feet to 22 ft, 8 inches will not increase the consequences of such events.

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

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

The proposed change to the Technical Specifications involves a slight change to the minimum required/reference reactor cavity water level during refueling operations. No new modes of operation or the utilization of equipment are involved. No new accident initiators are introduced as a result of allowing a lower minimum/reference water level. Therefore, this change does not involve a new or different kind of accident from any accident previously evaluated.

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

The margin of safety involved with this change involves the consequences that could result from the release of radioactive material from damaged fuel following a fuel handling accident, loss of decay heat removal, or inadvertent reactor draindown. The consequences of a dropped fuel bundle in the upper containment pool are insignificantly affected by allowing a slightly lower reactor cavity water level, as such an event would remain bounded by a dropped fuel bundle in the fuel building. Allowing a slightly lower required minimum reactor cavity water level during refueling operations would also have an insignificant effect on the volume of water available for decay heat removal capability, or to mitigate a draindown event. Therefore, the changes will not result in a significant reduction in the margin of safety.

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

Attorney for licensee: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW, Washington, DC 20036–5869.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: July 27, 2000.

Description of amendment request: The proposed amendment revises the Safety Limit Minimum Critical Power Ratio (SLMCPR) in the Technical Specifications (TSs) and makes some administrative changes associated with the revised SLMCPR to the TSs.

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

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

The SLMCPR, which is determined by using NRC approved methods, ensures that during normal operation and/or anticipated operational occurrences greater than 99.9% of all fuel rods in the core avoid the onset of transition boiling. (The operating limit for MCPR is determined by adding the change in Critical Power Ratio for anticipated operational occurrences to the SLMCPR. For limiting faults such as a loss of coolant accident, SLMCPR does not apply.) Although the SLMCPR is established to minimize the potential for fuel damage in response to anticipated operational occurrences, it has no impact on the cause of such occurrences. That is, establishment of the SLMCPR has no impact on the equipment failures or events that can lead to such occurrences. Therefore, the proposed change does not involve an increase in the probability of an accident.

The derivation of the cycle-specific SLMCPRs for incorporation into the TS has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), June 2000. Amendment 25, which describes the methodology for determining the SLMCPR, was incorporated into GESTAR–II in June 2000. GESTAR–II, Amendment 25 was approved by the NRC as of a March 11, 1999 safety evaluation report.

The basis of the MCPR safety limit is to ensure that greater than 99.9% of all fuel rods in the reactor core avoid the onset of transition boiling if the limit is not violated. The proposed SLMCPR preserves the existing margin to transition boiling and fuel damage in the event of a postulated transient/accident. The fuel licensing acceptance criteria for the SLMCPR calculation apply to the next operating cycle at CPS (Cycle 8) in the same manner as they have applied previously. The new core design for two-loop and single-loop operation that includes GE14 fuel, is in compliance withAmendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000 (GESTAR–II) which provides the NRC approved fuel licensing criteria. Since the basis of the MCPR safety limit remains unchanged, the probability of fuel damage and the potential consequences of anticipated operational occurrences is not increased. Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

In addition to the proposed change to the single-loop SLMCPR, the Note preceding TS 2.1.1.2 previously incorporated as part of License Amendment 113 is being proposed to be deleted. The Note associated with TS 2.1.1.2 was originally included to ensure that the SLMCPRs values were only applicable for the identified cycle (Cycle 7). Since that time, Amendment 25 to NEDE–24011–P–A–14 has been approved by the NRC, and new SLMCPRs have been calculated for the forthcoming fuel cycle, so this Note is no longer necessary. The Note was for information only and has no impact on the design or operation of the reactor. The proposed deletion of the Note is an administrative change that does not involve an increase in the probability or consequences of an accident previously evaluated.

The analysis contained in TS 5.6.5, ''Core Operating Limits Report (COLR),'' Paragraph b., is proposed to be updated to remove the references to the three letters that were submitted to the NRC to support Cycle 7 and which are not applicable to subsequent operating cycles, and to retain the reference to the ongoing standard non-cycle specific analysis approved by the NRC (i.e., GESTAR). This is an administrative change to ensure that the references contained in the CPS TS are accurate and consistent with other licensing documents. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

Based on the above, the proposed changes to the TS do not involve an increase in the probability or consequences of an accident.

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

The new SLMCPR limit for CPS nuclear fuel, including GE–14 fuel, has been determined using NRC approved methods. Use of the NRC-approved methodology preserves the basis for the MCPR safety limit which ensures that during normal operation and during an anticipated operational occurrence greater than 99.9% of all fuel rods in the core avoid the onset of transition boiling. For other accidents such as a loss of coolant accident, the SLMCPR does not apply. The proposed change does not involve any new modes of operation, modifications to plant equipment, and any setpoint changes. As a result, the proposed change does not involve a new or different kind of accident from any accident previously evaluated.

With regard to the previously described changes concerning the Note associated with TS 2.1.1.2 and references in TS 5.6.5, Paragraph b, these changes are administrative in nature. As such, these changes do not create the possibility of a new or different kind of accident from any that were previously evaluated.

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

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

The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid the onset of transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity. In addition, appropriate MCPR Operating Limits will continue to be enforced by procedures such that in the event of a transient, there will be adequate margin to the SLMCPR. The MCPR Operating Limits are based on the SLMCPR and NRC approved methods in GESTAR–II. Therefore, the proposed change to the single-loop SLMCPR will not involve a reduction in the margin of safety previously approved by the NRC.

Additionally, the proposed changes that remove the note preceding TS 2.1.1.2 and the removal of outdated references in TS 5.6.5, Paragraph b, are administrative changes that will not reduce the margin of safety previously approved by the NRC.

Based on the above, the proposed changes to the TS do not involve any 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: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW, Washington, DC 20036–5869.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 1, 1999, as supplemented June 14, and July 27, 2000, (the April 1, 1999 application was submitted by GPU Nuclear, Inc., but has subsequently been adopted by AmerGen Energy Company, LLC). The June 14 and July 27, 2000, supplements did not supercede the original April 1, 1999 application in its entirety. The April 1, 1999, application was noticed in the **Federal Register** on July 28, 1999 (64 FR 40906).

Description of amendment request: The June 14 and July 27, 2000, supplements revised the original application to change the Technical Specification (TS) limiting conditions for operation (LCOs) and the surveillance requirements related to the core flood tanks to be more consistent with the Standard Technical Specifications for B&W [Babcock & Wilcox] Plants (NUREG–1430 Rev. 1) than the proposed TS changes of the original application. This included the addition of a new surveillance Table 4.1–5. The supplements also revised TS 3.3.1.3.b and c related to the sodium hydroxide tank limits by moving them to a new TS 3.3.2.1. The proposed change to TS 4.5.3.1.b.2 has been revised to reflect the issuance of Amendment No. 212 on June 21, 1999, which had previously changed that TS.

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The proposed amendment makes administrative corrections, adds conditions to the limiting conditions for operation [LCOs], revises selected time clocks and surveillance requirements consistent with NUREG 1430, and adds a time clock to a unique LCO. These changes have no effect upon the plant design or operation. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed changes. Therefore, operation in accordance with the proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated, because no new accident initiators would be created.

(3) Operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety because no changes to plant operating limits or limiting safety system settings are proposed.

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.77

Attorney for licensee: Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: June 29, 2000, as supplemented by letter dated July 27, 2000.

Description of amendment request: The amendments would revise McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2 Technical Specification 5.6.5 Core Operating Limits Report, and the Bases of Sections 3.2.1 Heat Flux Hot Channel Factor $F_Q(X,Y,Z)$, 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor $F_{DH}(X,Y)$, 3.2.4 Quadrant Power Tilt Ratio, 3.5.1 Accumulators, and 3.5.2 ECCS-Operating.

These changes are being proposed to incorporate the Westinghouse Best-Estimate Large Break Loss of Coolant Analysis Methodology into the licensing basis for McGuire and Catawba units.

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

Pursuant to 10 CFR 50.92, Duke Energy Corporation has made the determination that this license amendment involves no significant hazards considerations by applying the standards established by NRC regulations in 10 CFR 50.92(c). This ensures that operation of the facility in accordance with the proposed amendment would not:

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

No. The proposed changes involve use of the Best-Estimate Large Break Loss of Coolant Accident (LOCA) Analysis Methodology and implementation of associated Technical Specifications changes. The plant conditions assumed in the analysis are bounded by the design conditions for all of the equipment in the plant. Therefore, there will be no increase in the probability of a LOCA. Additionally, the consequences of a LOCA are not being increased, since it has been demonstrated that the Emergency Core Cooling System performance conforms to the criteria contained in 10 CFR 50.46(b). No other accidents are potentially affected by this change.

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

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

No. The proposed changes to the Technical Specifications are to support implementation of Best-Estimate Large Break LOCA Analysis Methodology. There are no new modes of plant operation being introduced. The plant parameters assumed in the analysis are within the design limits of the existing plant equipment.

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

Involve a significant reduction in a margin of safety?

No. The analytic technique used in the analysis realistically describes the expected behavior of the McGuire/Catawba reactor system during a postulated LOCA. Uncertainties were accounted for as required by 10 CFR 50.46. A sufficient number of LOCA cases with different break sizes, different locations, and other variations in properties were analyzed to provide assurance that the most severe cases are calculated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46(b) are met.

Therefore the proposed amendment does not involve a significant reduction in any margin of safety.

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

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

Attorney for licensee: Catawba Nuclear Station, Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006. McGuire Nuclear Station, 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: February 29, 2000, supplemented by letter dated July 5, 2000.

Description of amendment request: The proposed amendments would revise Technical Specifications, Table 3.3.2–1, Engineered Safety Feature Actuation System Instrumentation, Function 6.f, Auxiliary Feedwater Pump Suction Pressure-Lo.

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

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

No. Only the trip setpoint and allowable value for CA pump low suction pressure auto-realignment to RN System are being modified in the Technical Specifications to accurately document the valid analyzed values stated in the calculations. The proposed change is consistent with the current licensing basis for the McGuire Nuclear Station, the setpoint methodologies used to develop the trip setpoints, the McGuire Safety Analyses, and current station calibration procedures and practices. 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 which will involve a reduction in reliability or effectiveness of the CA System. Consequently, any previous evaluations associated with accidents will not be affected by these changes.

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

No. Only the trip setpoint and allowable value for CA pump low suction pressure auto-realignment to RN System are being modified in the Technical Specifications to accurately document the valid analyzed values stated in the calculations. No changes are being made to actual plant hardware which will result in any new failure modes or new accident initiation mechanisms. Also, no changes are being made to the way the plant is being operated. The McGuire Nuclear Station will continue the current practice of using the valid trip setpoint values documented in the instrumentation procedure. Consequently, no new plant accidents will be created by these changes.

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

No. Only the trip setpoint and allowable value for CA pump low suction pressure auto-realignment to RN System are being modified in the Technical Specifications to accurately document the valid analyzed values stated in the calculations. The methods used for analyzing the allowable value are endorsed by Duke Power's EDM 102, ''Instrument Uncertainty Calculations''. 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. The impact of the proposed change will not challenge or exceed any safety limits or design limits during a design basis accident. Consequently, the integrity of the fission product barriers will still be maintained.

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

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina

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: June 16, 1999, as supplemented on May 4 and July 10, 2000.

Description of amendment request: NEDE–24011–P–A ''General Electric Standard Application for Reactor Fuel'' (GESTAR–II) is one of the approved analytical methods for performing the reload analysis as specified in Technical Specification (TS) 5.6.5.b.1. The proposed amendment incorporates 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. NEDO–21231 forms the current basis for the Pilgrim reactor core design process. The Nuclear Regulatory Commission (NRC) staff approved NEDO–21231 by a letter to General Electric dated January 25, 1985. NEDO–21231 describes a revised method for developing control rod withdrawal sequences to mitigate the consequences of the control rod drop accident (CRDA) in the startup and low power operating ranges of 20% RTP and 280 cal/gram peak fuel enthalpy. The proposed 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 proposed TS changes also include changes to the control rod worth limits to resolve License Event Report (LER) 98–006–00, dated April 30, 1998, and its supplement LER 98–006–01, dated August 27, 1988.

The proposed changes are modeled after NUREG–1433, Rev. 1, BWR/4 Standard Technical Specifications (STS) for incorporating the Pilgrim cycle-specific data for CRDA and BPWS for 20% RTP. The STS format is adopted based upon GESTAR II to reflect the Specifications, Actions, and BASES derived from NEDO–21231. The proposed TS changes consist of (i) administrative changes, (ii) more restrictive changes, and (iii) less restrictive changes to comply with TS 5.6.5.b.1 incorporating the current Pilgrim core design based upon the NRC approved NEDO–21231 and NEDE–24011–P–A.

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:

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

The proposed changes do not adversely affect accident initiators or precursors nor alter the design, conditions, and configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR).

This proposed change relocating the details of the methods for timing control rod drives from the Specifications to the BASES involves no technical changes to the Specifications. The requirement to verify scram times is incorporated into proposed SR 3.3.B.1.4; therefore, it does not eliminate any requirements, or impose a new or different treatment of the requirements. The BASES are subject to the Technical Specifications Bases Control Program contained in the Administrative Controls Section of the Technical Specifications. Since any changes to the BASES will be in accordance with these requirements, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed without prior staff approval.

The proposed changes provide more stringent requirements than those currently in the Technical Specifications. The more restrictive requirements will not alter the operation of process variables or SSCs as described in the safety analyses; therefore, they will not involve a significant increase in the probability of an accident occurring.

The proposed changes will ensure compliance with ''NEDO–21231, ''Banked Position Withdrawal Sequence (BPWS)''. The NEDO–21231 limits the maximum rod worth such that fuel enthalpy addition due to a control rod drop accident (CRDA) will not exceed 280 cal/gm, or require the plant to be placed in a condition where the LCOs do not apply sooner. In addition, changes are proposed to require entering a MODE in which the LCOs do not apply sooner than currently required. Therefore, the new requirements would decrease the consequences of an analyzed event.

Elimination of the requirement to shut down if one rod is stuck due to potential collet finger failure is being made concurrently with another change that will require a reactor shutdown if more than one rod is stuck for any reason. This additional restriction ensures that the reactor will be shut down as soon as it is determined that more than one rod may fail to scram. This differs from the existing requirement that allows operation with multiple stuck rods that are not fully inserted, provided reactivity margin is met. The consequences of an accident previously evaluated are not increased because the failure of a single control rod to insert will not prevent the reactor from reaching a subcritical condition as long as shutdown margin requirements are met.

The proposed SRs 4.3.B.1.1 and 4.3.B.1.2 only increase the interval between performance of a surveillance for about 10% to 20% of the control rods (those that are partially withdrawn). The purpose of the surveillance is to verify that rods can be inserted, thus verifying that rods are not stuck and scram capability is maintained. The 80% to 90% of the control rods that are fully withdrawn will continue to be tested at the 7-day frequency and should a stuck control rod be found, all withdrawn control rods will have to be tested within 24 hours. This change does not affect any initiating events for accidents previously evaluated. In addition, this change is being implemented concurrently with more restrictive requirements governing continued operation with stuck and inoperable control rods, which ensure the mitigative features of the control rods are maintained. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change eliminates the requirement to verify discernible neutron instrument response to control rod motion, the first time a rod is withdrawn after refueling or maintenance. The probability of an accident is not increased because the proposed change will not involve any physical changes to plant SSCs, or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The consequences of an accident are not increased because the CRDA analysis assumes a single failure of the control rod drive system when a single control rod drops out of the core from the fully inserted position after being disconnected from its drive and after the drive has been retracted to the fully withdrawn position while reactor power is less than 20%. During startup and before exceeding 20% reactor power, a large percentage of the rods are fully withdrawn in the normal course of a startup. All fully withdrawn rods are subjected to verification of coupling by the overtravel test, which verifies that the accident mitigation feature of the control rods is maintained.

The proposed change will allow either a second licensed operator or other qualified members of the technical staff to verify movement of control rods when the rod worth minimizer (RWM) is inoperable. The function of the RWM is to control adherence to the control rod withdrawal and insertion sequence. The use of a second licensed operator or other qualified members of the technical staff to perform these control rod movement verifications provides alternate means to accomplish the same function, thus, there is no change in the probability or consequences of an accident previously evaluated. Also, the proposed change will only require that the RWM sequence be verified when it is changed. The RWM does not monitor core thermal conditions, but simply enforces preprogrammed rod patterns as a backup intended to prevent reactor operator error in selecting or positioning control rods. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The more restrictive and new requirements will not alter the plant configuration (no new or different type of equipment will be installed nor is any equipment being removed) or change methods governing normal plant operation. The changes do impose different requirements; however, they are consistent with assumptions made in the safety analyses, therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed less restrictive change that increases the interval between performance of surveillance designed to verify that rods can be inserted for only 10% to 20% of the control rods (those that are partially withdrawn) not the manner in which the surveillance is performed does not impact reactivity controls. The changes in reactivity are not SSCs; therefore, the proposed changes will not involve any physical changes to the plant or the manner in which the plant is operated; and, therefore, the proposed changes will 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 a margin of safety?*

The proposed change relocating the details of the methods for timing control rod drives to the Bases does not impact the safety margin. The requirement to verify scram times is incorporated into proposed SR 3.3.B.1.4; thereby, preserving the analytic assumptions for the accident analyses, which also preserves the current margin of safety. The requirements to be transposed from the Technical Specifications, and are not being modified by the proposed change. Thus, there will be no significant reduction in the margin of safety.

Adding new requirements and making existing ones more restrictive does not involve a physical alteration of the plant (no new or different type of equipment will be installed nor is any equipment being removed), introduce any new tests, or change methods governing normal plant operation. The BPWS limits the maximum rod worth such that fuel enthalpy addition due to a CRDA will not exceed 280 cal/gm, the current bases for the TS limit. Therefore, the proposed change does not involve a reduction in the margin of safety.

Elimination of the requirement to shut down if one rod is stuck due to potential collet finger failure will not decrease a margin of safety because this change is being made concurrently with another change that will require a reactor shutdown if more than one rod is stuck for any reason. This additional restriction ensures that the reactor will be shut down as soon as it is determined that more than one rod may fail to scram, which ensures that the reactor is shut down when assumptions used in the analysis of those accidents and transients that depend on a scram may no longer be met. The failure of a single control rod to insert will not prevent the reactor from reaching a subcritical condition as long as shutdown margin requirements are met. Therefore, the proposed change does not involve a significant reduction in margin of safety.

The proposed increase in the interval from weekly to monthly for partially withdrawn control rods for performances of a surveillance may increase the time before a partially withdrawn control rod is discovered to be stuck. Changing the interval between surveillances does not affect the surveillance acceptance criteria, thus, the proposed change does not affect the analysis assumptions concerning the number of control rods that insert following a scram.

The proposed change will allow control rod movement verification, by licensed operators or other qualified members of the technical staff (i.e., personnel trained in accordance with an approved training program) when the RWM is inoperable, and limit the use of this alternate method to once per 12 months. This change does not impact the margin of safety because the verification of rod sequence and thus the assumed reactivity insertion rates following a reactor trip are maintained.

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: W. S. Stowe, Esquire, Entergy Nuclear Generation Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199

NRC Section Chief: James W. Clifford

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2 (ANO–1&2), Pope County, Arkansas

Date of amendment request: September 17, 1999, as supplemented by letters dated June 29 and August 3, 2000. The September 17, 1999, application was originally noticed in the **Federal Register** on February 23, 2000 (65 FR 9004).

Description of amendment request: The proposed amendment would change the Arkansas Nuclear One, Unit 2 (ANO–2) heavy load handling requirements and transportation provisions to permit the movement of the original and replacement steam generators through the ANO–2 containment construction opening during the steam generator replacement outage.

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:

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

During the 2R14 refueling outage/steam generator replacement outage, the OSGs [original steam generators] and the RSGs [replacement steam generators] will be moved between the new steam generator storage area / original steam generator storage facility and the runway beam support system (RBSS) / outside lift system (OLS). The RBSS/OLS is the structure used to rig the SGs [steam generators] in and out of the reactor containment building. In consideration of the magnitude of the loads being handled, the RBSS, OLS and transporters are of a robust, rugged design, proven by many prior steam generator replacements and other heavy load handling operations. However, due to the location of safety related underground structures, systems, and components (SSCs) in the vicinity of the RBSS/OLS and along the steam generator (SG) haul route, potential load handling accidents along the load paths must be considered for their effects on the SSCs. At ANO–2, the ground cover over several buried SSCs is not sufficient to be able to rule out the potential for a load drop to damage or cause failure of these SSCs. The functions of the SSCs in question are as support systems to the ANO–1 [Arkansas Nuclear One, Unit 1] and ANO–2 emergency diesel generators and the ANO–1 service water system. The fire protection system, a non-safety related system, was also considered. Existing plant procedures adequately address the scenario in question for the fire protection system.

The cause of a SG drop is assumed to be a non-mechanistic failure of the RBSS/OLS (or associated rigging), a failure of the SG transporter leveling hydraulics, or a seismically-induced failure of the loaded RBSS/OLS or SG transporter. The possibility of drops associated with other external events, such as tornadoes, high winds, and tornado missiles will be substantially minimized by procedures that prevent load handling under these weather conditions.

With ANO–2 defueled, the impact on ANO–2 due to loss of the emergency diesel generators fuel oil transfer system will be minimal. Long term actions to provide makeup water to the spent fuel pool may be necessary, but no immediate actions are required.

For ANO–1, a steam generator drop could render both diesel generators inoperable due to the loss of the fuel oil transfer system, and the emergency cooling pond inoperable due to the loss of the service water return line to the pond. Since ANO–1 is expected to be at full power operation, these conditions would require prompt action in accordance with technical specifications. Immediately following a drop from the OLS or from the transporter in the vicinity of the OLS, where damage to these systems is possible, ANO–1 will begin a shutdown and cooldown to cold shutdown conditions. In conjunction with the unit shutdown, contingency measures will be taken to compensate for the loss of the normal fuel oil supply to the emergency diesel generators.

The ability of ANO–1 to safely respond to analyzed events would be undiminished with the possible exception of the functions affected by the damaged equipment. With the compensatory measures to be established prior to the steam generator handling operations, and with the planned responses to a steam generator drop, the support system functions of the diesel generators and the service water system can be assumed to be maintained following the drop. Therefore, the drop will not affect the consequences of any analyzed event.

While the drop of a steam generator could cause damage to some safety related plant equipment, the failures of these components are not precursors to any analyzed accident. The drop of a steam generator will not have any other impact on plant equipment, and thus will not induce any analyzed plant transient. It will, however, result in a malfunction of equipment important to safety of a different type than any previously evaluated. Based on the compensatory measures and the low likelihood of the event during SG movement, this temporary condition is considered to be acceptable. On these bases, it is concluded that the proposed load handling operations will not significantly increase the probability or the consequences of accidents previously analyzed.

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

As noted in the response to the first question above, the only potential for a new or different kind of accident associated with this change request arises from a drop of a steam generator which is assumed to cause the loss of emergency power support systems for ANO–1. The cause of a SG drop is assumed to be a non-mechanistic failure of the RBSS/OLS (or associated rigging), a failure of the SG transporter leveling hydraulics, or a seismically-induced failure of the loaded RBSS/OLS or SG transporter. In the absence of a seismic event, there is no initiator for any consequential events (e.g., loss of offsite power) other than those directly caused by impact of the SG. Given this scenario, the plant response to a SG drop event would be governed by the technical specifications and existing plant procedures.

If a SG drop is seismically-induced, the simultaneous loss of normal offsite power sources is also assumed in this case since these sources are not seismically qualified. While this event is very unlikely due to the low frequency of earthquakes and the small amount of time that a steam generator will be in a position to cause damage, Entergy [Operations, Inc.] will provide contingency plans and compensatory measures to compensate for the loss of the normal fuel oil supply to the emergency diesel generators. Long term actions to provide makeup water to the spent fuel pool may be necessary, but no immediate actions are required.

Availability of the redundant ANO–1 service water heat sink, the Dardanelle Reservoir, during a seismic event assures that an uninterrupted source of service water will be available to support shutdown cooling of ANO–1.

The proposed load handling plans will not create the possibility of a new or different kind of accident from any previously evaluated.

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

ANO–1 Technical Specification 3.7.1.C requires both EDGs [emergency diesel generators] to be operable when the reactor temperature is ≥ 200 °F. If this condition is not met, Limiting Condition for Operation 3.0.3 applies. It requires that within one hour, action shall be initiated to place the unit in an operating condition in which the specification does not apply by placing it, as applicable, in at least hot standby within the next 6 hours, at least hot shutdown within the following 6 hours, and at least cold shutdown within the subsequent 24 hours. The bases for technical specification 3.7.1.C indicate that these operability requirements ensure that an adequate, reliable power source is available for all electrical equipment during startup, normal operation, safe shutdown, and handling of all emergency situations. The bases for EDG operation also require at least a seven day total diesel oil inventory during complete loss of electrical power conditions.

The postulated loss of both trains of the ANO–1 EDG fuel oil transfer system due to a SG drop would require that ANO–1 be shut down. This situation could be considered to involve a reduction in the margin of safety, because a new common cause failure mechanism is being introduced by the movement of the SGs over the EDG fuel oil lines and transfer pump power cables. To restore the margin of safety and return the EDGs to functionality, temporary compensatory measures are being proposed.

Based on the above discussions, with the implementation of the proposed compensatory measures and the low likelihood of such an event, the failures caused by a SG drop event will not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 18, 1999, as supplemented by letters dated June 29, July 19, and August 9, 2000. The August 18, 1999, application was originally noticed in the **Federal Register** on February 23, 2000 (65 FR 9005).

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.4.5, ''Steam Generators,'' to note that the requirements for inservice inspection do not apply during the steam generator replacement outage (2R14), to delete inspection requirements associated with steam generator tube sleeving and repair limits, to revise the requirement for tube inspection to mean an inspection from tube end (cold leg side) to tube end (hot leg side), 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 would also be made.

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:

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

The accidents of interest are a tube rupture, loss of coolant accident (LOCA) in combination with a safe shutdown earthquake and a steam line break in combination with a safe shutdown earthquake. A reduction in tube integrity could increase the possibility of a tube rupture accident and increase the consequences of a steam line break or LOCA. The tubing in the replacement steam generators is designed and evaluated consistent with the margins of safety specified in the ASME [American Society of Mechanical Engineers] Code [Boiler and Pressure Vessel Code], Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action if tube degradation is present. The ASME [Code], Section XI basis for the 40% through wall plugging limit is applicable to the replacement steam generators just as it was to the original steam generators. As a result there is no reduction in tube integrity for the replacement steam generators.

Addition of a ''Note'' to clarify that inservice inspection is not required during the steam generator replacement outage is an administrative change that provides clarification regarding inservice inspection requirements. The change in reporting requirements is also an administrative change. The requirements for inservice inspection or the plugging limit for the tubes are not altered by these administrative changes. Additionally, changes were made to the bases to remove potentially misleading information. Bases changes are considered to be administrative in nature.

Elimination of the repair option and the associated references to repair of the original steam generator tubes is an administrative adjustment since the sleeve design is not applicable to the replacement steam generators. The elimination of the repair option does not alter the requirements for inservice inspection or reduce the plugging limit for the tubes.

A preservice eddy current inspection will be performed onsite prior to installation of the replacement steam generators. The orientation of the replacement steam generators during the eddy current exam will not impact the results. The hydrostatic test required by the ASME Code, Section III for the replacement steam generators is to be performed in the manufacturing facility and not as part of a reactor coolant system hydrostatic test.

The post-repair leakage test required by the ASME Code, Section XI for an operating plant is performed at a much lower pressure. No evolutions subsequent to the replacement steam generator hydrostatic test are expected to occur that will change the condition of the tubes prior to operation. This change does not alter the requirement to perform a preservice inspection. As a result, an inservice inspection is not required during the steam generator replacement outage.

The requested ANO–2 [Arkansas Nuclear One, Unit 2] Technical Specification changes do not alter the requirements for tube integrity or tube plugging limits. The change to the definition of tube inspection is a conservative change; therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes do not affect the design or function of any other safety-related component. There is no mechanism to create a new or different kind of accident for the replacement steam generators by eliminating repair criteria or by clarifying the applicable preservice and inservice inspection requirements because a baseline of tube conditions is established and plugging limits are maintained to ensure that defective tubes are removed from service.

The requested ANO–2 Technical Specification changes do not alter the requirements for tube integrity or tube plugging limits. The change to the definition of tube inspection is a conservative change; therefore, this change does not [create the possibility of a new or different kind of accident from any previously evaluated].

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety.

The tubing in the replacement steam generators is designed and evaluated consistent with the margins of safety specified in the ASME Code, Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action to preserve the design margin if tube degradation is present.

Based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations [, Inc.] has determined that the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida.

Date of amendment request: July 19, 2000.

Description of amendment request: The amendment will extend the applicability of the current reactor coolant system (RCS) pressure/temperature limits and maximum allowed RCS heatup and cooldown rates to 21.7 effective full power years (EFPY) of operation. The associated low temperature overpressure protection (LTOP) temperature limits, which are based on the pressure/temperature limits, will also be extended to 21.7 EFPY of operation.

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

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

The pressure-temperature (P/T) limit curves in the Technical Specifications are conservatively generated in accordance with the fracture toughness requirements of 10 CFR 50 Appendix G as supplemented by the ASME Code Section XI, Appendix G recommendations. The adjusted reference temperature (ART) values are based on the Regulatory Guide 1.99, Revision 2 shift prediction and attenuation formula and have been validated by a credible reactor vessel surveillance program. There are no changes to the limit curve, only a change in the period of applicability based on more recent fluence predictions. Based on the current fluence projections, analysis has demonstrated that the current P/T limit curves will remain conservative for up to 21.7 EFPY.

In conjunction with extending the effectiveness of the existing P/T limit curves, the low temperature overpressure protection (LTOP) analysis for 15 EFPY is also extended. The LTOP analysis confirms that the current setpoints for the power-operated relief valves (PORV) will provide the appropriate overpressure protection at low RCS temperatures. Because the P/T limit curves have not changed, the existing LTOP values have not changed, this includes the PORV setpoints.

The P/T limit curves and LTOP analysis have not changed; therefore, the proposed amendment does not represent a change in the configuration or operation of the plant. The results of the existing LTOP analysis have not changed, and the limiting pressures for given temperatures will not be exceeded for the postulated transients. Therefore, assurance is provided that reactor vessel integrity will be maintained. Thus, the proposed amendment does not involve an increase in the probability or consequences of accidents previously evaluated.

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

The requirements for P/T limit curves and LTOP have been in place since the beginning of plant operation. The only changes in these curves are the extension of the period of applicability (EFPY), which is based on new fluence data and the operating time (EFPY) required to reach the same limiting fluence used for the current 15 EFPY P/T curves. Since there is no change in the configuration or operation of the facility as a result of the proposed amendment, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Analysis has demonstrated that the fracture toughness requirements of 10 CFR 50 Appendix G are satisfied and that conservative operating restrictions are maintained for the purpose of low temperature overpressure protection. The P/T limit curves will provide assurance that the RCS pressure boundary will behave in ductile manner and that the probability of a rapidly propagating fracture is minimized. Therefore, operation in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: March 7, 2000, as supplemented on April 21, 2000 and June 14, 2000.

Description of amendment request: The proposed amendment would revise the surveillance requirements from once per refueling interval for each excess flow check valve (EFCV) to testing a representative sample of EFCVs once per 24 months.

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

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

This change will not alter the physical design of the plant. The proposed Amendment would modify the testing of excess flow check valves (EFCV) from each valve being tested once per refueling interval to testing a representative sample of EFCVs once per 24 months (the length of a refueling interval). The EFCVs installed at Oyster Creek are extremely reliable. Oyster Creek records demonstrate that there has never been a failure of an EFCV to isolate in the thirty-year history of Oyster Creek.

A GE [General Electric] Topical Report evaluated the reliability of EFCVs installed at Oyster Creek and other plants. Oyster Creek and three other facilities have installed Chemquip excess flow check valves. Chemquip EFCVs were shown in the Topical Report to have a failure rate of 1.78E–7, which was the lowest of the valve manufacturers included in the evaluation. The current Oyster Creek accident analysis does not take credit for any flow restriction provided by EFCVs although the valve design does restrict flow. Therefore, changing the surveillance requirements for the EFCVs does not involve a significant increase in the probability of an accident.

EFCVs limit the reactor coolant release following the failure of an instrument line, valve or component on an instrument line. The valves isolate at a given flow and are periodically functionally tested to ensure proper isolation with resulting minimal flow. The radiological consequences of an instrument line break have been evaluated at Oyster Creek. That evaluation does not take credit for the excess flow check valve when assessing the radiological consequences of the accident. The analysis was submitted to the NRC and was approved in NUREG 1382 ''Safety Evaluation Report related to the full term operating license for Oyster Creek Nuclear Generating Station.''

This change will not increase the consequences of an instrument line break or any postulated accident.

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

Operation of the facility in accordance with the proposed amendment would modify the testing frequency of EFCVs. This change does not add components or make any other physical change to the plant. The valves will be tested in the same manner as they are now although less frequently. EFCVs are located exclusively in instrument lines and the failure of an instrument line is currently analyzed in the FSAR [final safety analysis report]. The plant is not being physically changed, and the consequences of a valve failing to isolate are within the FSAR analyzed event. Therefore, this change does not create the possibility of a new or different accident not previously analyzed.

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

The proposed Amendment would modify the testing frequency of excess flow check valves (EFCVs) which are located in instrument lines. The only function of EFCVs is to limit the reactor coolant release following the failure of an instrument line, valve or component on an instrument line. The current Oyster Creek accident analysis does not take credit for any flow restriction provided by EFCVs, although the valve design does restrict flow. The proposed change does not alter the plant design in any manner. Furthermore, the instrument line break analysis assumptions also remain unchanged. Therefore, there is no impact on the current procedures or accident analysis. As a result, operating the plant in accordance with the proposed Amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Marsha Gamberoni.

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 amendment requests: April 6, 2000

Description of amendment requests: The proposed amendments would approve an unreviewed safety question allowing a change to the Updated Final Safety Analysis to allow a change to the analysis methodology used in the High Energy Line Break (HELB) program to incorporate the recommendations of NUREG/CR–2913.

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 occurrence or consequences of an accident previously evaluated?

The proposed changes do not impact the design of these high-energy lines such that previously analyzed [structures, systems and components] SSCs would now be more likely to fail. The changes will not modify high-energy lines to reduce their design capability of maintaining pressure boundary integrity during normal operating and accident conditions. The use of the NUREG/CR–2913 methodology to more accurately define the dynamic effects from high-energy line breaks and cracks does not affect the probability of any analyzed piping break or critical crack events. The use of the NUREG/CR–2913 methodology does not affect high-energy line break or crack initiators or precursors. The [steam generator blowdown] SGBD and [chemical, volume and control system] CVCS letdown piping will be modified and analyzed, as required, to ensure that the piping stresses remain below the threshold for postulation of a critical crack or break. Also, the effects of breaks or critical cracks outside of the break exclusion zones have been reviewed and determined to not have an adverse impact on the piping within the exclusion zone. The modified SGBD piping in the normal flash tank room will be analyzed to ensure that the application of the [Standard Review Plan] SRP for postulating cracks based on piping stresses is acceptable. Therefore, incorporating these new methodologies does not affect equipment malfunction probability, nor does it affect or create new accident initiators or precursors. Additionally, the NRC expected the results of revisions to SRP Section 3.6.2 requirements to yield more efficient regulatory practices, improve plant piping systems design, increase plant reliability, and decrease occupational radiation exposure associated with inspections and repairs.

The proposed changes permit relaxation of protective requirements that may represent a potential increase in the consequences of an accident. However, the proposed changes are consistent with the current regulatory guidelines for HELB evaluations and continue to ensure that protection of SSCs required for accident mitigation is maintained. The NUREG/CR–2913 methodology for determining the effects of jet flow from HELB events shows that SSCs outside the distance of ten piping diameters from the break or critical crack are undamaged. The SRP allowances for break and crack exclusions embody the understanding that the probability of breaks or critical cracks in piping systems that satisfy the stress criteria is extremely low. For those areas addressed by the methodology changes, protection is not required while still providing reasonable assurance that there is no undue risk to the health and safety of the public. Therefore, protection of SSCs required for accident mitigation is assured by use of these well-defined design methodologies. Thus, there will be no reduction in the capability of those SSCs in limiting the consequences of previously evaluated accidents. Malfunctions caused by HELBs and critical cracks have been previously analyzed in the Updated Final Safety Analysis Report (UFSAR). Thus, no additional radiological source terms are generated, and the consequences of an accident previously evaluated in the UFSAR will not be increased.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed changes do not impact the design of these high-energy lines such that previously unanalyzed breaks would now occur. The change to incorporate the NUREG/CR–2913 methodology does not introduce any new malfunctions; it more accurately defines the effects from the high-energy line breaks and cracks for use in the HELB program.

Regarding the incorporation of the SRP break exclusion zones, the break exclusion stress thresholds provide assurance that the piping is capable of withstanding the design loadings without the possibility of developing a through wall crack or break. The piping will be modified and completely analyzed to ensure that the piping stresses are below the threshold for break exclusion. The effects of breaks outside of the break exclusion zones have been reviewed and determined to not have an adverse impact on the piping within the exclusion zone. The modified SGBD piping in the normal flash tank room will be analyzed to ensure that the application of the SRP for postulating cracks based on piping stresses is acceptable. The proposed changes do not result in modification to high-energy lines that would reduce their design capabilities to maintain pressure boundary integrity during normal operating and accident conditions. Therefore, use of the new design methodologies does not affect or create new accident initiators or precursors or create the possibility of a new or different kind of accident.

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

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

The approval of the license amendment will not result in any modifications to high-energy lines that would reduce their design capabilities to maintain pressure boundary integrity during normal operating and accident conditions. By using these new design methodologies, protection of SSCs required for accident mitigation is assured.

The NUREG/CR–2913 methodology better defines the extent of impingement loads from the postulated high-energy line breaks and cracks. Use of the NUREG/CR–2913 methodology establishes that unprotected components located more than ten diameters from a pipe break or crack in piping containing fluids within the assumptions of NUREG/CR–2913 are without further analysis assumed undamaged by a jet. This conclusion has been reviewed and accepted by the NRC as providing adequate safety margin for high-energy piping. Protection of SSCs required for accident mitigation will continue to be assured by use of the NUREG/CR–2913 methodology if modifications to those SSCs are implemented in the future.

The use of the SRP break exclusion zones incorporates industry lessons learned and ensures that an adequate safety margin is maintained. The SGBD and CVCS letdown piping will be analyzed after modifications are performed in accordance with the original piping design code to ensure that the piping stresses are below the SRP threshold for break exclusion. Also, the effects of breaks outside of the break exclusion zones have been reviewed and determined to not have an adverse impact on the piping within the exclusion zone. The modified SGBD piping in the normal flash tank room will be analyzed to ensure that the application of the SRP for postulating cracks based on piping stresses is acceptable. Therefore, the capability of those SSCs to limit the offsite dose consequences of previously evaluated accidents to levels below the approved acceptance limits will continue to be assured.

The SRP presents the most definitive basis available for specifying the NRC's design criteria and design guidelines for an acceptable level of safety. The SRP guidelines resulted from many years of experience gained by the NRC in establishing and using regulatory requirements in the safety evaluation of nuclear facilities. The implementation of the design guidelines contained in MEB 3–1 assures that adequate protection is provided and a consistent level of safety is maintained. In addition, some regulatory requirements developed over the years as part of the licensing process have resulted in additional safety margins that overlap the safety margins provided by the criteria of MEB 3–1. Consequently, use of these new design methodologies instead of the previous licensing basis requirements cannot significantly reduce the existing margin of safety.

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

In summary, based upon the above evaluation, I&M has concluded that the proposed changes involve no significant hazards consideration.

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: Claudia M. Craig

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 amendment requests: June 12, 2000

Description of amendment requests: The proposed amendments would allow the licensee to use the methodology and the alternative source term (AST) contained in 10 CFR 50.67 as described in NUREG–1465, ''Accident Source Terms for Light-water Nuclear Power Plants'' to show compliance with 10 CFR Part 50, Appendix A, Criteria 19.

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 occurrence or consequences of an accident previously evaluated?

The proposed change to implement the AST involves changes to the methodologies and acceptance criterion associated with the control room dose analysis. The actual sequence and progression of accidents are not changed. However, the regulatory assumptions regarding the analytical treatment of the accidents are affected by the change. The use of an AST alone cannot increase the probability of an accident or the core damage frequency. The proposed change to use the AST does not make any changes to equipment, procedures, or processes that increase the likelihood of an accident. It does not affect any accident initiators or precursors. The methodology is used to determine consequences of an accident and has no impact on their likelihood of occurrence. Therefore, this proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The current acceptance criterion specify the dose to personnel in terms of ''rem whole body'' or equivalent for the duration of the accident, where the dose derived using the AST is given in rem [total effective dose equivalent] TEDE, as described in 10 CFR 50.67. TEDE includes internal and external exposure; whole body includes external exposure only. The current acceptance criterion focuses on doses to the thyroid and the whole body. It is based on the assumption that the major contributor to dose will be radioiodine. Although this may be appropriate with the Technical Information Document (TID)–14844, ''Calculation of Distance Factors for Power and Test Reactor Sites'', source term implemented by RGs 1.4, it may not be true for a source term based on a more complete understanding of accident sequences and phenomenology. The AST includes a larger number of radionuclides than did the TID–14844 source term as implemented in regulatory guidance. The whole body and thyroid dose criteria considered the noble gases and iodine contributors as the limiting factors. The acceptance criteria of 5 rem TEDE and 5 rem whole body are not equivalent, so they cannot be compared directly. I&M has reanalyzed the loss of coolant accident (LOCA) and non-LOCA events to determine the limiting condition for control room dose using the AST. The calculated dose for all the analyzed events meets the acceptance criterion for GDC–19 as described in 10 CFR 50.67. Therefore, the consequences are not significantly increased.

The [control room emergency ventilation system] CREVS is designed to mitigate the consequences of an accident. It is not assumed to operate in the pressurization mode until after an accident has occurred. The system itself has no impact on the initiation of any evaluated accidents. Therefore, the changes to the CREVS requirements do not increase the probability of an accident previously evaluated.

The proposed changes to the CREVS requirements do not affect the ability to maintain a control room pressure boundary. The changes ensure that the control room will be pressurized following an accident where the CREVS is required to operate to minimize unfiltered inleakage. The proposed 24-hour allowed outage time and subsequent shutdown action are consistent with the requirements for an inoperable filter unit and are reasonable due to the low probability of the initiation of an accident requiring actuation of the CREVS occurring when the pressure boundary is inoperable. Control room dose is significantly increased with increased unfiltered inleakage. Specifying the test condition in the surveillance allows increases in unfiltered inleakage to be identified and evaluated. Preserving the control room pressure boundary provides assurance that the consequences of an accident previously evaluated are not significantly increased.

The proposed applicability and action requirements during the movement of irradiated fuel assemblies ensure the CREVS is operable for the protection of control room personnel in the event of a fuel handling accident.

The proposed changes to the Limiting Conditions for Operation (LCO) address redundant dampers that are being installed. Adding the new equipment to the LCO and action requirements ensures all components associated with the CREVS are operable or action is taken to restore them. The proposed changes do not affect equipment design or operation. Therefore, the consequences of accidents previously evaluated are not increased.

The proposed changes for the charcoal testing method affect activities in the laboratory only and have no impact on plant operation. Sampling and testing charcoal will not initiate an accident. The charcoal adsorbers are used to mitigate the consequences of an accident and are not operated until after an accident has occurred. Therefore, the probability of an accident previously evaluated is not affected. Charcoal testing verifies the ability of the charcoal adsorbers to function as assumed following an accident. The new method for testing the CREVS samples provides more accurate and reproducible laboratory results. These results provide assurance that the charcoal adsorbers will meet the assumed radioiodine removal efficiency following an accident. Therefore, the consequences of accidents previously evaluated are not increased.

The [high-efficiency particular air] HEPA filter/charcoal adsorber units in the CREVS, [engineered safety features ventilation system] ESFVS, and [storage pool ventilation system] SPVS are designed to mitigate the consequences of an accident. They are not assumed to operate until after an accident has occurred. The adsorber units have no impact on the initiation of any evaluated accidents. Therefore, the proposed change to reduce the differential pressure does not increase the probability of an accident previously evaluated. The proposed change to surveillance requirements to reduce the allowable pressure drop across the HEPA filter/charcoal adsorber unit ensures the system flow rates can be maintained so that the system performs as designed. The change ensures that filter units are replaced before airflow is restricted. This allows the required area to be pressurized so that unfiltered inleakage remains within the amount assumed in the accident analysis. Therefore, the proposed revision to reduce the allowable pressure drop requirement does not significantly increase the consequences of an accident previously evaluated.

The remaining changes are administrative in nature. The proposed editorial changes involve reformatting of the individual T/S pages to standardize page appearance and readability and do not alter any requirements. The proposed change to separate the CREVS functions into individual specifications does not affect the system operability requirements or make any changes in how the equipment is operated. The separation of the two functions does not affect the ability of the CREVS to cool or pressurize the control room envelope. The proposed change to incorporate the new laboratory testing standard for charcoal adsorbers in the ESFVS and SPVS is administrative because the test conditions are consistent with the standard referenced in the T/S. These changes are administrative in nature and do not affect the probability or consequences of accidents previously evaluated.

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

The use of an AST alone cannot create the possibility of a new or different kind of accident. The proposed change to use the AST does not make any changes to equipment, procedures, or processes. The AST does not create any new accident initiators or precursors. It is merely a method used to predict radionuclides released following an accident. Therefore, this proposed change does not increase the possibility of a new or different kind of accident than previously evaluated.

The CREVS is designed to mitigate the consequences of an accident. It is not assumed to operate until after an accident has occurred. The proposed LCO requirement to maintain the control room envelope/pressure boundary operable and expand the area to include the control room heating ventilation and air conditioning equipment room and plant process computer room does not affect system design or operation. The area defined as the control room envelope includes all of the areas that communicate with the control room. A tracer gas test confirmed that the defined control room envelope can be pressurized to greater than or equal to 1/16 inch of water gauge, as assumed in the accident analysis. The proposed surveillance requirement to specify a makeup airflow rate of less than or equal to 1000 cubic feet per minute (cfm) allows periodic verification that the assumed unfiltered inleakage is within the assumptions of the accident analysis. The new requirement provides added assurance that the pressure boundary is maintained operable. The proposed changes to the CREVS requirements do not introduce any new plant equipment or new methods of operating the equipment. No new failure mechanisms are introduced.

The proposed change to incorporate the new testing requirements of ASTM D3803–1989 is administrative in nature. It affects activities in the laboratory only and has no impact on plant operation. The change does not affect the method for obtaining the charcoal sample. It does not cause any of the ventilation equipment to be operated in a new or different manner.

The change to reduce the allowable pressure drop across the pressurization filter train to 4 inches water gauge ensures system performance is consistent with design. The revised value is more restrictive and provides assurance that the affected components of the filter unit are replaced before airflow is reduced to the extent that it affects the pressurization capability of the CREVS, ESFVS, and SPVS. No new failure mechanism is created.

The remaining changes are administrative in nature. The proposed editorial changes involve reformatting of the individual T/S pages to standardize page appearance and readability and do not alter any requirements. The proposed change to separate the CREVS functions into individual specifications does not affect the system operability requirements or make any changes in how the equipment is operated. The separation of the two functions does not affect the ability of the CREVS to cool or pressurize the control room envelope. The proposed change to incorporate the new laboratory testing standard for charcoal adsorbers in the ESFVS and SPVS is administrative because the test conditions are consistent with the standard referenced in the T/S. These changes are administrative in nature and do not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

The proposed change to implement the AST for the revised analysis incorporates the guidance for application of the AST provided in NUREG–1465 and draft RG–1081. The change involves the use of new terminology for the acceptance criterion expressed as 5 rem TEDE. The term TEDE is defined in 10 CFR 20 as the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). The acceptance criteria of 5 rem TEDE and 5 rem whole body are not equivalent. The NRC has revised the current GDC–19 whole body dose criterion with a criterion in terms of rem TEDE for the duration of the accident in 10 CFR 50.67 for the licensee that seeks to revise its current radiological source term with an AST.

The NRC recognizes that an analysis using the AST may represent a reduction in the margin of safety for some applications. The margin of safety is typically defined as the difference between the calculated parameters (offsite and control room dose) and the associated regulatory or safety limit. Implementing the AST in accordance with draft RG–1081 and 10 CFR 50.67 revises the acceptance criterion (regulatory limit) contained in GDC–19 to 5 rem TEDE. The calculated control room dose is below the new acceptance criterion. In 10 CFR 50.67, the rule considers the 5 rem whole body, or its equivalent to any part of the body is accounted for in the definition of TEDE and by the 5 rem TEDE annual limit. Therefore, revising the control room dose analysis using the new terminology for the AST does not involve a significant reduction in a margin of safety.

The margin of safety associated with the CREVS T/S is to maintain control room dose within the limits of GDC–19. The proposed changes to the CREVS requirements ensure that accident analysis assumptions are preserved so that the dose limit is met. The proposed change for control room envelope/pressure boundary provides assurance that positive pressure is maintained in the envelope and that unfiltered inleakage is bounded by the accident assumption. Adding a test requirement for filtered makeup airflow also supports this requirement. The proposed change to expand the applicability requirements and actions provides assurance that the CREVS is operable during times when an accident could occur that may affect the control room environment. The proposed changes that reflect addition of the dampers provides assurance that the control room pressure boundary will be isolated and the envelope will be pressurized when CREVS is actuated following an accident. The proposed change to reduce the allowable pressure drop across the HEPA filter/charcoal adsorber units provides assurance that the CREVS, ESFVS, and SPVS provide the required airflow. This allows areas to be pressurized as required.

The proposed change to incorporate the testing standards recommended for the charcoal adsorbers in GL 99–02 provides assurance that the charcoal adsorbers will remove radioiodine as assumed in the accident analysis. Additional margin is gained by applying a safety factor to the iodine removal efficiency assumed in the accident analysis. This safety factor applies to CREVS , ESFVS, and SPVS. The T/S have also been revised to reflect the iodine removal efficiency assumed in the accident analysis. The acceptance criterion reflects the analysis assumption and the safety factor.

The remaining changes are administrative in nature. The proposed editorial changes involve reformatting of the individual T/S pages to standardize page appearance and readability and do not alter any requirements. The proposed change to separate the CREVS functions into individual specifications does not affect the system operability requirements or make any changes in how the equipment is operated. The separation of the two functions does not affect the ability of the CREVS to cool or pressurize the control room envelope. The proposed change to incorporate the new laboratory testing standard for charcoal adsorbers in the ESFVS and SPVS is administrative because the test conditions are consistent with the standard referenced in the T/S. These changes are administrative in nature and do not involve a significant reduction in the margin of safety.

The proposed changes support the control room dose calculations that demonstrate that the GDC–19 requirement will be met. Therefore, these changes do not involve a significant reduction in the margin of safety.

In summary, based upon the above evaluation, I&M has concluded that these changes involve no significant hazards consideration.

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: Claudia M. Craig

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

Date of amendment request: November 30, 1999, as supplemented June 28, 2000

Description of amendment request: The licensee proposed to amend the unit's Technical Specifications (TS), Section 3.7.2, ''Control Room Envelope Filtration (CREF) System'' and Section 5.5.7, ''Ventilation Filter Testing Program (VFTP),'' to require testing consistent with American Society for Testing and Materials (ASTM) Standard D3803–1989, in lieu of the current D3803–1979. This application for amendment is a response to the NRC's Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The licensee's November 30, 1999, application proposed to amend only the TS that was then in effect; the TS was fully overhauled in style and format by Amendment No. 91. In anticipation of such overhaul of the TS, the staff declined to proceed with review of the November 30, 1999, application. The licensee's June 28, 2000, application proposes to amend the TS in its current form (i.e., as revised by Amendment No. 91 to the Improved Technical Specification format). The licensee's two submittals differ only in form and style; the proposed TS requirements and supporting analyses in these submittals are identical.

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 in its November 30, 1999, application. The NRC staff has reviewed the licensee's analysis against the standard of 10 CFR 50.92(c). The NRC staff's review is presented below:

1. 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 change will require testing the Standby Gas Treatment (SGT) System and CREF System charcoal filters in accordance with ASTM D3803–1989 versus the current ASTM D3803–1979. Neither the SGT nor CREF system is an initiator or precursor to an accident previously evaluated; both systems perform mitigative functions in response to an accident. Failure of either system would result in the inability to perform its mitigative function but no failure would increase the probability of an accident. Accordingly, changing the test methodology of the charcoal filters will not affect any accident precursors. Therefore, the probability of an accident previously evaluated is not increased.

The SGT system is designed to limit the release of radioactive gases to the environment within the guidelines of 10 CFR 100 for analyzed accidents. The CREF system is designed to limit doses to control room operators to less than the values allowed by General Design Criterion 19. Both systems contain charcoal filters which require laboratory carbon sample analysis be performed in accordance with RG 1.52 as required by TS. Charcoal filter samples are tested to determine whether the filter adsorber efficiency is greater than that assumed in the design basis accident analysis. The proposed TS changes to test the charcoal material in accordance with

ASTM D3803–1989 (versus ASTM D3803–1979) will assure the ability of the subject systems to perform their intended function. As long as these systems perform their intended functions, there will not be any increase in the consequences of an accident previously evaluated.

2. 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.

The proposed TS change will require testing the SGT and CREF charcoal filters in accordance with ASTM D3803–1989 versus ASTM D3802–1979. This change will not involve placing these systems in new configurations or operating the systems in a different manner that could result in a new or different kind of accident. Testing in accordance with the ASTM D3803–1989 standard will assure the ability of the subject systems to perform their intended function by providing a more realistic prediction of the capability of the charcoal filters. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. 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 of the SGT or CREF System nor will it affect the ability of these systems to perform their intended functions. Charcoal filter samples are tested to determine whether the filter adsorber efficiency is greater than that assumed in the design basis accident analysis. The proposed TS changes to test the charcoal material in accordance with ASTM D3803–1989 (versus ASTM D3803–1979) will assure the ability of the subject systems to perform their intended function by providing a more realistic prediction of the capability of the charcoal filters. Also, the proposed changes are consistent with the changes recommended in NRC GL 99–02. 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. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: April 19, 2000

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) Definition 1.7, ''CONTAINMENT INTEGRITY''; Sections 3/4.6.1.1, ''Containment Systems, Primary Containment, CONTAINMENT INTEGRITY''; 3/4.6.1.2, ''Containment Systems, Primary Containment, Containment Leakage''; 3/4.6.1.3, ''Containment Systems, Primary Containment, Containment Air Locks''; 3/4.6.1.6, ''Containment Systems, Primary Containment, Containment Structural Integrity''; 3/4.6.6.3, ''Containment Systems, Secondary Containment Structural Integrity''; and 6.8, ''Procedures and Programs.'' The use of this option requires the implementation of a program based on Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' and modification of the Technical Specifications to reflect this program. The proposed Technical Specifications changes will implement a performance-based Containment Leakage Testing Program in accordance with 10 CFR Part 50, Appendix J, Option B as a substitute for the requirements of 10 CFR Part 50, Appendix J, Option A. The Bases for these Technical Specifications will be modified to address the proposed changes.

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

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

The changes involved in this license amendment request revise the testing criteria for the containment penetrations. The revised criteria will be based on the guidance in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' This guidance allows for the use of relaxed testing frequencies for containment penetrations that have performed satisfactorily on a historical basis.

The Containment Leakage Rate Testing Program considers the type of service, the design of the penetration, and the safety impact of the penetration in determining the testing interval of each penetration. The [Nuclear Regulatory Commission] Staff has reviewed the potential impact of performance-based testing frequencies for containment penetrations during the development of the Option B regulation. The NRC Staff review is documented in NUREG–1493 ''Performance-Based Containment Leak-Test Program.'' The review concluded that reducing the frequency of Type A tests (Integrated Leak Rate Tests) from three per ten years to one per ten years leads to an imperceptible increase in risk. EPRI Research Project Report TR–104285, ''Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,'' also concluded that a relaxation of the test intervals for Type B and C penetrations results in a negligible increase in total plant risk.

The use of Option B will allow the extension of testing intervals with a minimal impact on the radiological release rates since most penetration leakage is continually well below the specified limits. In the accident risk evaluation, the NRC Staff noted that the accident risk is relatively insensitive to the containment leakage rate because the accident risk is dominated by accident sequences that result in failure of or bypass of the containment. The containment leak rate and component performance history at Millstone Unit No. 3 are consistent with the conclusions reached in NUREG–1493.

Therefore, the proposed license amendment adopting a performance-based approach for verification of leakage rates for isolation valves, containment penetrations, and the containment overall will continue to meet the regulatory goal of providing an essentially leak-tight containment boundary, and will provide an equivalent level of safety as the current requirements.

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

Changes to the Administrative section describe the containment testing program only and cannot increase the probability or consequences of an accident previously analyzed.

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

The proposed license amendment does not change the operation of the plant. The proposed changes do not involve any physical or operational changes to structures, systems or components. No new failure mechanisms beyond those already considered in the current plant safety analyses are introduced. Since there is no change to the equipment or the operation of the plant, there is no possibility of creating a new or different kind of accident than previously analyzed. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

Changes to the Administrative section describe the containment testing program only and cannot create a different accident from any previously analyzed.

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

During the development of 10 CFR Part 50, Appendix J, Option B, the NRC Staff determined the reduction in safety associated with the implementation of the performance-based testing program. The results of this review are documented in NUREG–1493. The review concluded that reducing the frequency of Type A tests (Integrated Leak Rate Tests) from three per ten years to one per ten years leads to an imperceptible increase in risk. The use of Option B will allow the extension of testing intervals with a minimal impact on the radiological release rates since most penetration leakage is continually well below the specified limits. In the accident risk evaluation, the NRC Staff noted that the accident risk is relatively insensitive to the containment leakage rate because the accident risk is dominated by accident sequences that result in failure of or bypass of the containment. The use of a performance-based testing program will continue to provide assurance that the accident analysis assumptions remain bounding. Therefore, these changes do not involve a significant reduction in the margin of safety.

Changes to the Administrative section describe the containment testing program only and cannot reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: April 19, 2000

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.7.1.5, ''Plant Systems—Main Steam Line Isolation Valves.'' Specifically, the change will remove the requirement to perform partial stroke testing of the main steam line isolation valves during power operation, modify the TS wording for clarity, combine two surveillance requirements into one, and modify the associated Bases for consistency.

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

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

The proposed changes to Technical Specification 3.7.1.5 will not affect the operability requirements for the MSIVs [Main Steam Isolation Valves] during plant operation in Modes 1 through 4. If a MSIV is not operable, restoration of operability is still required, or the valve will be closed. Once closed, the MSIV is performing the accident mitigation function.

The addition of a footnote to allow performance of the full valve stroke surveillance requirement when the MSIVs are closed to comply with action requirements will allow testing to be performed that may be necessary to demonstrate MSIV operability. Since proper operation of the MSIVs would be expected when utilizing this provision, and this test is used to confirm valve operability, the MSIVs should function properly to mitigate an accident.

The proposed change to remove the requirement to perform partial stroke testing of the MSIVs when the plant is in Modes 1 or 2 will eliminate a high risk activity that is not necessary to ensure the ability of the MSIVs to perform their safety function. Recent valve design changes and improvements to the MSIV solenoid valves have increased reliability in proper main valve operation. Additionally, redundant solenoid valve design precludes a single failure from affecting the ability of the main valve to close within the required time. Thus, the full stroke test is sufficient to ensure operability.

The proposed changes will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed changes can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The changes do not alter the way any structure, system, or component functions and do not adversely alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. The proposed changes will reduce the likelihood of a transient by eliminating a high risk surveillance. Also, the response of the plant and the operators following these accidents is unaffected by the change. Therefore, the proposed changes 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.

The proposed changes modify the LCO [Limiting Condition for Operation], applicability, action requirements, and surveillance requirements of Technical Specification 3.7.1.5. These changes have no adverse effect on equipment important to safety. This equipment will continue to function as assumed in the design basis accident analyses. The proposed changes will not result in any plant configuration changes. There will be no adverse effect on plant operation or accident mitigation equipment. The plant response to design basis accidents will not change. Therefore, there will be no 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.

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

Date of amendment request: April 19, 2000.

Description of amendment request: The proposed amendment would modify Technical Specification Sections 3.8.4.1, ''Electrical Power Systems—Containment Penetration Conductor Overcurrent Protective Devices''; 3.8.4.2.1, ''Electrical Power Systems—Motor-Operated Valves Thermal Overload Protection''; and 3.8.4.2.2, ''Electrical Power Systems—Motor-Operated Valves Thermal Overload Protection Not Bypassed. The proposed changes will relocate the requirements for containment penetration conductor overcurrent and motor-operated valve thermal overload protective devices from the Technical Specifications to the Technical Requirements Manual (TRM). The Bases for these Technical Specifications will be modified to address the proposed changes.

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

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

The proposed changes to relocate the requirements for containment penetration conductor overcurrent and motor-operated valve thermal overload protective devices from Technical Specifications to the TRM will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. Operation of the containment penetration conductor overcurrent and motor-operated valve thermal overload protective devices are not accident initiators and cannot cause an accident. Whether the requirements for the containment penetration conductor overcurrent and motor-operated valve thermal overload protective devices are located in Technical Specifications or the TRM will have no effect on the probability or consequences of any accident previously evaluated. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to relocate the requirements from Technical Specifications to the TRM will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed changes will not introduce any new failure modes that could result in a new accident. Also, the response of the plant and the operators following the design basis accidents is unaffected by the changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will relocate the requirements for containment penetration conductor overcurrent and motor-operated valve thermal overload protective devices from Technical Specifications to the TRM. Any future changes to the relocated requirements will be in accordance with 10 CFR 50.59 and approved station procedures. The proposed changes will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the relocated requirements do not meet any of the 10 CFR 50.36c(2)(ii) criteria on items for which Technical Specifications must be established. Therefore, there will be no 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota.

Date of amendment request: July 18, 2000.

Description of amendment request: The proposed amendment would change the Technical Specifications to add operability requirements for the No. 12 residual heat removal service water (RHRSW) pump.

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 12 RHRSW Pump is not an accident (fire) initiator. During a fire in the Control Room or Cable Spreading Room, the ASDS [alternate shutdown system] panel provides alternate shutdown capability. The proposed amendment provides operability requirements to ensure 12 RHRSW Pump is available when alternate shutdown is required so that safe shutdown can be achieved and maintained in accordance with existing procedures. The proposed operability requirements are consistent with previous ASDS requirements for 12 RHRSW Pump and other equipment required for alternate shutdown. Dose to the public and the Control Room operators are not affected by the proposed change.

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes.

Therefore, the proposed amendment will not significantly increase the probability or the 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 analyzed.

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes.

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

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

The proposed amendment is within current Technical Specification requirements for other equipment required for alternate shutdown and ensures that 12 RHRSW Pump will be available for alternate shutdown when required. The allowed ASDS outage time for 12 RHRSW Pump is consistent with that allowed for other alternate shutdown equipment. The proposed amendment maintains margins of safety. Therefore, the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Claudia M. Craig.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota.

Date of amendment request: July 20, 2000.

Description of amendment request: The proposed amendment would (1) revise the Technical Specifications (TSs) to include the automatic reactor water cleanup (RWCU) system isolation feature, (2) restore the dose equivalent iodine-131 (DEI) limit to 2 microcuries per gram, (3) change the RWCU reactor water level automatic isolation signal from Low to Low-Low reactor water level, add TSs for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) low steam line pressure isolation instrumentation, (4) delete the HPCI 150,000 lb/hr low range high flow isolation instrumentation and add a time delay to the 300,000 lb/hr upper range high flow isolation instrumentation, and (5) change the suppression chamber water allowable water level from volume units to level units.

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 proposed new limiting conditions for operation and surveillance requirements for the RWCU high system flow and room temperature signals and the increase in the allowable reactor coolant DEI are administrative in nature and do not involve an increase in the probability or consequences of a previously evaluated accident. The RWCU automatic isolation instrumentation will reduce the consequences of a break in the RWCU System allowing the reactor coolant DEI to be increased, consistent with the value assumed in the Monticello USAR [updated safety analysis report] for the design basis MSLB [main steam line break] safety analysis.

Changing the reactor water level RWCU automatic isolation setpoint to Low Low Reactor Water Level will not increase the probability or consequences of a break in the RWCU system because new high flow and high area temperature instrumentation provides an improved capability for isolating a RWCU break independent of changes in reactor water level.

Changes to the HPCI steam supply line automatic isolation instrumentation will not increase the probability or consequences of a break in the HPCI steam line. Elimination of the 150,000 lb/hr isolation signal will improve the reliability of the HPCI system. The remaining 300,000 lb/hr delay high flow isolation signal provides more than adequate protection for a steam line break in this system. The consequences of a break in the HPCI remain bounded by the MSLB safety analysis.

Adding a description of the Group 3 logic and recirc [recirculation] sample valves isolation in the Bases; adding LCOs [limiting conditions for operation] and Surveillance Requirements for the HPCI and RCIC low steam line pressure isolation logic; and changing the method of describing the allowable suppression pool water inventory from volume to level are all administrative changes than [sic] cannot adversely affect the consequences of any evaluated accident.

The proposed changes do not present the opportunity for a new release path for radioactive material.

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

No system, structure, or component (SSC) described in the USAR as important to safety is adversely affected by these changes. No new type of credible event could be identified which would be created by the proposed Technical Specification changes. Nothing was identified in these changes which could create the possibility for a new or different kind of accident.

The RWCU line break accident was previously analyzed. However, non-conservative values for mass and energy release were used. New, more conservative, calculations prompted the prudent installation of an automatic RWCU break isolation system at Monticello. The RWCU line break outside of containment is once again bounded by previously analyzed design basis MSLB accident.

Changes to the HPCI steam line high flow instrumentation will improve the reliability of the HPCI system, while continuing to provide a high degree of protection for a break in the HPCI steam supply line.

Addition of LCOs and Surveillance Requirements for the HPCI and RCIC low steam line pressure isolation and changing the way in which suppression pool level is specified in the Technical Specifications are administrative changes that cannot result in a new or different kind of accident than any previously analyzed.

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

The margin of safety for the RWCU line break can be expressed in terms of the offsite and control room doses which could result from this event. The previous RWCU line break analysis relied on operator action to isolate the line break after an assumed delay of 10 minutes. The new RWCU automatic isolation instrumentation initiates isolation of a RWCU line break in less than 27 seconds. The new isolation logic limits the potential offsite radiological consequences from a RWCU line break to a fraction of the bounding MSLB accident. The proposed Technical Specification would incorporate LCOs and Surveillance Requirements for the new isolation logic and change the reactor water level setpoint to a value which helps prevent unnecessary isolation of the system following reactor scrams. Margins of safety are improved by these changes.

Proposed changes to the HPCI steam line high flow instrumentation will improve the reliability of the HPCI system and increase existing margins of safety for accidents in which HPCI operation is credited. The margin of safety for a HPCI steam line break can also be expressed in terms of the offsite and control room doses which could result from this event. The modified HPCI isolation logic limits the potential offsite radiological consequences from a HPCI steam line break to a fraction of the bounding MSLB accident.

Adding a description of the Group 3 recirc sample isolation to the Bases, adding LCOs and Surveillance Requirements for the HPCI and RCIC low steam line pressure isolation, and changing the method of describing the allowable suppression pool water inventory are administrative changes. No significant changes in plant equipment or plant operation will occur and no equipment important to safety is affected as a result of these changes. No margin of safety is therefore 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: Jay E. Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW, Washington, DC 20037

NRC Section Chief: Claudia M. Craig

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

Date of amendment request: July 27, 2000

Description of amendment request: The amendment would delete a note to Technical Specification Section 12.1.A to allow the Safety Limit Minimum Critical Power Ratio (SLMCPR) to be applicable beyond cycle 14. The amendment would also revise the reference to the General Electric Standard Application for Reactor Fuel (GESTAR) document in Section 6.9.a.4 to incorporate the latest revision.

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

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

Deletion of a note stating that the SLMCPR remains applicable through Cycle 14 does not affect the initiation of any accident. Operation in accordance with the current SLMCPR ensures the consequences of previously analyzed accidents are not changed. Therefore, this proposed change 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 SLMCPR establishes a performance limit for the fuel. This limit remains unchanged. Deleting a note to reflect this is an administrative change and will not initiate any accident. Therefore, this proposed change 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.

GE [General Electric] has performed an evaluation of the SLMCPR for Cycle 15 and found that the cycle specific value, based on current reload plans, is bounded by the generic value calculated for GE 12 fuel. The existing SLMCPR remains unchanged for Cycle 15 and the margin of safety for the prevention of onset of transition boiling is unchanged. Therefore, this 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: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Marsha K. Gamberoni

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: July 20, 2000 (PCN–488, Supplement 1). This application supersedes the licensee's application of August 11, 1999.

Description of amendment requests: The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Southern California Edison Company to withdraw its August 11, 1999, application for proposed amendments. The Commission had previously issued a Notice of Consideration of Issuance of Amendments published in the **Federal Register** on September 8, 1999 (64 FR 48866). However, by letter dated July 20, 2000, the licensee withdrew the proposed change. TAC Nos. MA6282 and MA6283 used for the review of the August 11, 1999, application have been closed.

As submitted by the licensee on July 20, 2000, the proposed amendments would modify the Technical Specifications for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 to revise Surveillance Requirement (SR) 3.3.7.3 by providing allowable values in place of analytical limits for certain degraded voltage parameters, and by deleting unnecessary parameter limits in cases where plant safety is not affected. The proposed change would also delete redundant SR 3.3.7.4.

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

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

No. Proposed Change Number (PCN)–488, Supplement 1, revises the Technical Specification (TS) Surveillance Requirement (SR) acceptance criteria of the Loss of Voltage Signal (LOVS), Degraded Grid Voltage with Safety Injection Actuation Signal (DGVSS), and Sustained Degraded Voltage Signal (SDVS) relay circuits. These circuits are not accident initiators.

PCN–488 Supplement 1 revises the TS SR acceptance requirements to make them more limiting than the present requirements. Because the revised acceptance criteria are more limiting than the present requirements, the consequences of accidents analyzed in the Updated Final Safety Analysis Report (UFSAR) are not increased. PCN–488 Supplement 1 also revises the TS SR acceptance requirements to delete or revise upper and lower bounds in cases where the deleted bound provides no safety benefit. Deleting or revising bounds having no safety significance does not involve a significant increase in the probability or consequences of an accident previously evaluated.

PCN–488 Supplement 1 deletes redundant SR 3.3.7.4, which is not in NUREG–1432, Standard Technical Specifications, Combustion Engineering Plants. Deleting a redundant requirement does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Consequently, the proposed amendment does not result in an increase in the probability or consequences of accidents evaluated in the UFSAR.

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

No. PCN–488 Supplement 1 revises the TS SR acceptance criteria of the LOVS, DGVSS, and SDVS relay circuits, which are not accident initiators, and deletes a redundant SR. PCN–488 Supplement 1 does not introduce any revision in the hardware configuration of the protective circuitry for LOVS, DGVSS or SDVS. The measurement required by the deleted, redundant surveillance is required elsewhere in the TS. For these reasons, PCN–488 Supplement 1 does not create the possibility of any new or different kind of accident from any previously evaluated.

(3) Does this amendment request involve a significant reduction in a margin of safety?

No. PCN–488 Supplement 1 provides allowable values for the acceptance criteria for the TS SR for LOVS, DGVSS and SDVS. As such, the revised values are more limiting than the current values, which represent design limits. Therefore, PCN–488 Supplement 1 does not involve a significant reduction in a margin of safety.

PCN–488 Supplement 1 also revises the TS SR acceptance requirements to delete or revise upper and lower bounds in cases where the deleted bound provides no safety benefit. Deleting or revising bounds having no safety significance does not involve a significant reduction in a margin of safety.

PCN–488 Supplement 1 additionally deletes a redundant SR. Because the deleted surveillance is required elsewhere in the TS, this action does not involve a significant reduction in a margin of safety.

For these reasons, PCN–488 Supplement 1 does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770

NRC Section Chief: Stephen Dembek

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: August 10, 2000.

Brief description of amendments: The proposed change would revise Technical Specification 5.6.5 entitled CORE OPERATING LIMITS REPORT. TXU Electric proposes to revise the Large Break Loss of Coolant Accident methodology used at Comanche Peak Steam Electric Station, Units 1 and 2.

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

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

Response: No

The proposed change involves an administrative change only. Designation of the Revised Large Break Loss of Coolant Accident analysis methodology, described in ERX–2000–002–P, as the approved Large Break Loss of Coolant Accident analysis methodology is required to maintain the accuracy of the Technical Specification 5.6.5 (Core Operating Limits Report) and to maintain consistency with the resolution of issues as prescribed in 10 CFR 50.46.

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 proposed change involves an administrative change only. Technical Specification 5.6.5, Item 15, is being changed to reference the revised Large Break Loss of Coolant Accident analysis methodology currently under NRC review. No actual plant equipment will be affected by the proposed change. An analysis for Unit 1, Cycle 8, is imbedded in the referenced Topical Report, from which it is concluded that no failure modes, not bounded by previously evaluated accidents, will be created.

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

Margin of safety is associated with the 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. This request involves an administrative change (subject to NRC approval of the revised Large Break Loss of Coolant Accident Analysis methodology) only to incorporate the revised Large Break Loss of Coolant Accident analysis methodology into the allowable analysis methodologies specified in Technical Specification 5.6.5.

No actual plant equipment will be affected by the proposed change. The compliance of the revised methodology with the requirements of 10 CFR 50.46 and Appendix K will be addressed through the NRC staff's review of the topical report. Therefore, it is concluded that the use of the proposed methodology will not degrade the confidence in the ability of the fission product barriers to limit the level of radiation dose to the public.

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

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

Date of application request: July 21, 2000 (ULNRC–04285)

Description of amendment request: The proposed amendment would revise Limiting Condition for Operation (LCO) 3.9.4, ''Containment Penetrations,'' of the Callaway Technical Specifications (TS) to allow containment penetrations with direct access to the outside atmosphere to be open under administrative controls during refueling operations, by adding a note to the LCO that states ''containment penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.'' In addition, there would be a format and editorial correction to TS 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Start Air,'' to correct an error in the conversion to the improved TS issued May 28, 1999, in Amendment No. 133. There are also revisions to the TS Bases for the proposed changes to LCO 3.9.4.

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 status of the penetration flow paths during refueling operations has no [effect] on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Since the consequences of a FHA [fuel handling accident] inside containment with open penetration flow paths are bounded by the current analysis described in the FSAR [Callaway Final Safety Analysis Report] and the probability of an accident is not affected by the status of the penetration flow paths, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to correct editorial/format errors involve corrections to the technical specifications that are associated with the original conversion application and supplements or the certified copy of the Improved Technical Specifications. As such, these changes are considered as administrative changes and do not modify, add, delete, or relocate any technical requirements in the technical specifications.

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

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

The open containment penetration flow paths are not accident initiators and do not represent a significant change in the configuration of the plant. The proposed allowance to open the containment penetrations during refueling operations will not adversely affect plant safety functions or equipment operating practices such that a new or different accident could be created.

The proposed changes to correct editorial/format errors involve corrections to the technical specifications that are associated with the original conversion application and supplements or the certified copy of the improved Technical Specifications. As such, these changes are considered as administrative changes and do not modify, add, delete, or relocate any technical requirements [in] the technical specifications.

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

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

Technical Specification LCO 3.9.4 closure requirements for containment penetrations ensure that the consequences of a postulated FHA inside containment during core alterations or irradiated fuel handling activities are minimized. The LCO establishes containment closure requirements, which limit the potential escape paths for fission products by ensuring that there is at least one integral barrier to the release of radioactive material. The proposed change to allow the containment penetration flow paths to be open during refueling operations under administrative controls does not significantly affect the expected dose consequences of a FHA because the limiting FHA is not changed. The proposed administrative controls provide assurance that prompt closure of the penetration flow paths will be accomplished in the event of a FHA inside containment thus minimizing the transmission of radioactive material from the containment to the outside environment. Under the proposed TS change, the provisions to promptly isolate open penetration flow paths provide assurance that the offsite dose consequences of a FHA inside containment will be minimized.

The proposed changes to correct editorial/format errors involve corrections to the technical specifications that are associated with the original conversion application and supplements or the certified copy of the Improved Technical Specifications. As such, these changes are considered as administrative changes and do not modify, add, delete, or relocate any technical requirements in the technical specifications.

Therefore, the proposed changes to the Technical Specifications 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: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Section Chief: Stephen Dembek

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

Date of amendment request: March 10, 2000

Description of amendment request: The proposed amendments would implement a Pressure and Temperature Limits Report (PTLR) concurrent with the implementation of the Improved Standard Technical Specifications. NRC Generic Letter 96–03 provides guidance for licensees allowing relocation of the reactor coolant system pressure temperature limit curves and low temperature overpressure protection system limits from the Technical Specifications to a PTLR.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The proposed changes relocate the pressure-temperature limits and low temperature overpressure protection limits from the Technical Specifications to a Pressure Temperature Limits Report (PTLR). The proposed changes also provide revised pressure-temperature limits and revised low temperature overpressure protection limits. Appropriate design and safety limits are retained in the Specifications, thereby meeting the requirements of 10 CFR 50.36. Specific, approved methodologies used to determine and evaluate the parameter requirements are added to the Specifications and a reporting requirement is added to ensure the NRC is apprised of all changes. Operation of the PBNP will continue to meet all design and safety analysis requirements because approved methodologies are required to be used to evaluate and change parameters, and appropriate safety and design limits maintained in the Technical Changes.

Therefore, neither the probability nor consequences of an accident previously evaluated can be increased.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendment does not create a new or different kind of accident from any accident previously evaluated.

Operation of PBNP, in accordance with the proposed changes, will continue to meet all design and safety limits. Appropriate design and safety limits continue to be controlled within the Technical Specifications as they are presently. These changes will not result in a change to the design and safety limits under which PBNP operation has been determined to be acceptable. These changes cannot result in a new or different kind of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendment does not result in a significant reduction in a margin of safety.

Appropriate safety limits continue to be controlled by the Specifications. Changes to the relocated pressure-temperature and low temperature overpressure protection limits will be accomplished using NRC approved methodologies, thereby ensuring operation will continue within the bounds of the existing safety analyses including all applicable margins of safety. Therefore, operation in accordance with the proposed changes cannot result in 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: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Section Chief: Claudia M. Craig

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.

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: February 1, 2000, as supplemented on June 1 and July 13, 2000

Description of amendment request: The proposed amendment would change the Technical Specification and Bases Sections associated with the requirements for the Reactor Coolant System (RCS) loops and Shutdown Cooling (SDC) System trains during all modes of plant operation. Many of the proposed changes are associated with the format and structure of the affected Technical Specifications and will not result in any technical changes to the current requirements. The proposed format changes will result in Technical Specifications that will be clear, concise, and easier for the control room operators to use. Some of the changes are proposed to achieve consistency with the Standard Technical Specifications for Combustion Engineering Plants in NUREG–1432, Rev. 1. The Bases for the Technical Specifications would also be revised to reflect the proposed changes.

*Date of publication of individual notice in **Federal Register:*** July 31, 2000 (65 FR 46748)

Expiration date of individual notice: August 31, 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, the Gelman Building, 2120 L Street, NW., Washington, DC, and 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: June 19, 2000 (U–603378)

Brief description of amendment: The amendment changes the leak rate test frequency for the primary containment feedwater penetrations sealed by the Feedwater Leakage Control System.

Date of issuance: August 11, 2000

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

Amendment No.: 131

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

*Date of initial notice in **Federal Register***: July 3, 2000 (65 FR 41103)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 11, 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: August 20, 1999, as supplemented February 18, April 19, and May 22, 2000.

Brief description of amendment: The amendment revised the calibration frequency of the 4kV (kilovolt)

Engineered Safeguards Bus Undervoltage Relays (Diesel Start) (item 43.a of Table 4.1–1 of the Technical Specifications (TSs)) from a refueling interval to annually. The TS Bases have also been changed to reflect that the degraded voltage relay setpoint tolerance is being changed from an ''as left'' to an ''as found'' reading. Additionally, the amendment approves a revision to the Updated Final Safety Analysis Report (UFSAR) to allow for manual operator action for voltage protection rather than full automatic voltage protection. These changes are reflected in the revised UFSAR pages 8.2–3 and 8.2–5.

The amendment also adds new TSs 3.7.2.a(ii) and 3.7.2.h to address voltage on the 230 kV grid as a precondition of criticality and to provide a time limit for when the 230 kV grid voltage is found to be insufficient to support loss-of-coolant accident electrical loading during power operation. Various minor editorial changes have also been made. The Bases have also been changed to reflect the addition of the two new TSs and to provide clarification of the components to which surveillance is applicable.

Date of issuance: August 3, 2000

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

Amendment No.: 224

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

Date of initial notice in ***Federal Register***: December 1, 1999 (64 FR 67334) and June 2, 2000 (65 FR 35404).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam

Electric Plant, Unit No. 2, Darlington County, South Carolina.

Date of application for amendment: June 14, 2000, as supplemented July 14, 2000.

Brief description of amendment: The amendment revises Technical Specification 5.6.5 to incorporate analytical methodologies that are used for the Core Operating Limits Report that have been accepted by the Nuclear Regulatory Commission for referencing in licensing in cycle-specific applications.

Date of issuance: August 3, 2000.

Effective date: August 3, 2000.

Amendment No.: 188.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 28, 2000 (65 FR 39957).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 11, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to increase allowable out-of-service times (AOTs) and surveillance test intervals (STIs) for selected actuation instrumentation. The amendments implement AOT/STI changes based on Topical Reports by General Electric Company and the Boiling Water Reactor Owners' Group which have previously been reviewed and approved by NRC.

Date of issuance: August 2, 2000.

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

Amendment Nos.: 177 and 173.

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

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 2000.

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York.

Date of application for amendment: November 18, 1999, incorporating supporting analyses provided by letter dated October 8, 1999, as supplemented by letters dated February 14, March 21, April 6, April 13, and May 11, 2000.

Brief description of amendment: The proposed amendment would remove the requirement for charcoal filters and high efficiency particulate filters in the containment fan cooler system, revise the time requirement for subcriticality prior to core alterations from 174 hours to 100 hours, revise flow rate requirements for containment fan coolers and control room ventilation units to be consistent with the design basis, state that the control room ventilation system, in the post-accident mode, will be operated with filtered intake of outside air, allow containment personnel access doors to be open during refueling operations, and allow an administrative substitution of ''monthly'' in place of ''every 31 days'' in various surveillance requirements.

Date of issuance: July 27, 2000.

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

Amendment No.: 211.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 20, 2000 (65 FR 3256).

The February 14, March 21, April 6, April 13, and May 11, 2000, submittals contained supplemental information that did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 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: November 29, 1999, as supplemented December 20, 1999.

Brief description of amendment: The amendment added a license condition authorizing a one-time extension of the steam generator inspection interval to permit the next inspection to coincide with the next scheduled refueling outage.

Date of issuance: August 4, 2000.

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

Amendment No: 112.

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

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17915). The December 20, 1999, letter provided additional information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

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

No significant hazards consideration comments received: No.

GPU Nuclear Corporation and Saxton Nuclear Experimental Corporation, Docket No. 50–146, Saxton Nuclear Experimental Facility (SNEF), Bedford County, Pennsylvania.

Brief description of amendment: The amendment changes the Technical Specification organizational and administrative controls for the SNEF to reflect changes in GPU Nuclear following the sale of the Oyster Creek Nuclear Generating Station.

Date of Issuance: August 10, 2000.

Effective date: The license amendment is effective as of its date of issuance.

Amendment No.: 16.

Amended Facility License No. DPR–4: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 28, 2000 (65 FR 39956). The Commission's related evaluation of the amendment is contained in a safety evaluation dated August 10, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 5, 1999, as supplemented by two letters dated April 6, 2000, and April 13, 2000.

Brief description of amendment: These amendments conform the license to reflect the transfer of Operating License No. DPR–16 for the Oyster Creek Nuclear Generating Station.

Date of Issuance: August 8, 2000.

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

Amendment No.: 213.

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

Date of initial notice in ***Federal Register:*** December 16, 1999 (64 FR 70292).

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

The April 6 and April 13, 2000, supplements did not change the initial proposed no significant hazards consideration determination and were within the scope of the initial application as originally noticed.

No significant hazards consideration comments received: No.

GPU Nuclear, Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, Dauphin County, Pennsylvania.

Date of application for amendment: April 6, 2000, as supplemented by letters dated May 25, 2000, July 18, 2000, and August 8, 2000.

Brief description of amendment: The amendment reflects an administrative name change from GPU Nuclear Corporation to GPU Nuclear, Inc. Furthermore, the license amendment makes an editorial change to better describe TMI–2's use of site physical security, guard training and qualification, and safeguard contingency plans that are maintained by the Three Mile Island Nuclear Station, Unit 1, licensee, AmerGen Energy Company, LLC. In addition, the licensee requested that minor changes (mainly in titles) be made in Section 6.0 of the Technical Specifications to reflect the TMI–2 organizational and administrative controls that will exist following the sale of the Oyster Creek Nuclear Generating Station, which occurred August 8, 2000. The May supplement provided a response to a staff request for additional information, and the July and August supplements related to the requested effective date of the amendment. The supplements did not expand the scope of the application, as noticed in the ***Federal Register*** (65 FR 21484, dated April 21, 2000), or change the proposed no significant hazards consideration determination.

Date of issuance: August 9, 2000.

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

Amendment No.: 54.

Facility Operating License No. DPR–73: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** April 21, 2000 (65 FR 21484).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 24, 1999, as supplemented February 4 and March 17, 2000.

Brief description of amendment: The amendment conforms the license to reflect the transfer of operating authority under Operating License No. DPR–49 to Nuclear Management Company, LLC, as approved by order of the Commission dated May 15, 2000.

Date of issuance: August 7, 2000.

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

Amendment No.: 232.

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

Date of initial notice in Federal Register: February 4, 2000 (65 FR 5703).

The February 4 and March 17, 2000, supplements were within the scope of the initial application as originally noticed. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 7, 1999

Brief description of amendment: This amendment removes the action requirement to suspend all operations involving positive reactivity additions from Technical Specification (TS) 3.4.2.1, ''Reactor Coolant System—Safety Valves,'' TS 3.4.2.2, ''Reactor Coolant System—Safety Valves,'' and TS 3.7.6.1, ''Plant Systems—Control Room Emergency Ventilation System.'' The associated Bases have also been revised.

Date of issuance: August 7, 2000.

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

Amendment No.: 248.

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

Date of initial notice in Federal Register: January 26, 2000 (65 FR 4285).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 2000.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: November 24, 1999, as supplemented February 2, 2000.

Brief description of amendment: The amendment conforms the license to reflect the transfer of operating authority under Operating License No. DPR–22 to Nuclear Management Company, LLC, as approved by order of the Commission dated May 15, 2000.

Date of issuance: August 7, 2000

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

Amendment No.: 110.

Facility Operating License No. DPR–22. Amendment revised the Operating License.

Date of initial notice in ***Federal Register:*** February 15, 2000 (65 FR 7574)

The February 2, 2000, supplement was within the scope of the initial application as originally noticed. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2000.

No significant hazards consideration comments received: No

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, and Docket No. 72–10, Prairie Island Independent Spent Fuel Storage Installation, Goodhue County, Minnesota

Date of application for amendments: November 24, 1999, as supplemented February 2, 2000.

Brief description of amendment: The amendments conform the licenses to reflect the transfer of operating authority under Operating License Nos. DPR–42 and DPR–60 and Materials License No. SNM–2506 to Nuclear Management Company, LLC, as approved by order of the Commission dated May 15, 2000.

Date of issuance: August 7, 2000

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

Amendment Nos.: 153 and 144

Facility Operating License Nos. DPR–42 and DPR–60 and Materials License No. SNM–2506: Amendments revised the Operating Licenses and Materials License.

Date of initial notice in ***Federal Register:*** February 15, 2000 (65 FR 7574).

The February 2, 2000, supplement was within the scope of the initial application as originally noticed. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 2000.

No significant hazards consideration comments received: No

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: March 1, 1999, as supplemented October 1, and October 6, 1999, and June 6, 2000.

Brief description of amendment: The amendment supports the installation of a digital Power Range Neutron Monitoring system and the incorporation of the long-term thermal-hydraulic stability solution hardware.

Date of issuance: August 1, 2000

Effective date: Effective as of date of issuance and shall be implemented prior to restart from the Peach Bottom Atomic Power Station, Unit 2, Fall 2000 refueling outage.

Amendment No.: 232

Facility Operating License No. DPR–44: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 2, 1999 (64 FR 29711). The October 1, and October 6, 1999, and June 6, 2000 submittals provided clarifying information that did not expand the scope of the original ***Federal Register*** notice or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 2000.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia.

Date of application for amendments: October 1, 1999.

Brief description of amendments: The amendments revise the minimum fuel oil level for the diesel generator day tanks in Surveillance Requirement 3.8.1.3 and revise the acceptable fuel oil level storage band in Required Action Statement B of Limiting Condition for Operation 3.8.3.

Date of issuance: July 27, 2000.

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

Amendment Nos.: 221 and 162.

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

Date of initial notice in ***Federal Register:*** November 17, 1999 (64 FR 62715).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 27, 2000.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia.

Date of application for amendments: August 30, 1999.

Brief description of amendments: The amendments revised the Technical Specifications 5.2.2, ''Unit Staff'', to raise the level of the approval authority for deviations above the guidelines provided to minimize unit staff overtime. Specifically, the amendments change the level of overtime approval authority from ''department superintend'' to ''department manager.''

Date of issuance: August 10, 2000.

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

Amendment Nos.: 113 and 91.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

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

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 18, 2000.

Brief description of amendments: These amendments revise Technical Specification (TS) Section 5.3, ''Design Features—Reactor Core,'' and TS Section 6.9, ''Administrative Controls—Reporting Requirements'' to identify M5 alloy as a material used in the construction of fuel assemblies and to cite the topical report that describes the fuel.

Date of issuance: July 31, 2000.

Effective date: July 31, 2000.

Amendment Nos.: 258 and 249.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the TSs.

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17920).

The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated April 10, 2000, and in a Safety Evaluation dated July 31, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 30, 1999, as supplemented June 16 and August 3, 2000.

Brief description of amendments: Updates Technical Specification (TS) requirements, and appropriate TS Bases sections for reactor coolant system (RCS) leakage detection and RCS operational leakage specifications to be consistent with the Improved Westinghouse Standard TS (NUREG–1431).

Date of issuance: August 4, 2000.

Effective date: August 4, 2000.

Amendment Nos.: 259 and 250.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the TS.

Date of initial notice in ***Federal Register:*** October 20, 1999 (64 FR 56533).

The June 16 and August 3, 2000, letter provided clarifying information and changes that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original notice.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 24, 1999, as supplemented January 31, 2000.

Brief description of amendments: The amendments conform the licenses to reflect the transfer of operating authority under Operating License Nos. DPR–24 and DPR–27 to Nuclear Management Company, LLC, as approved by order of the Commission dated May 15, 2000.

Date of issuance: August 7, 2000.

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

Amendment Nos.: 197 and 202.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Operating Licenses.

Date of initial notice in ***Federal Register:*** February 4, 2000 (65 FR 5705).

The January 31, 2000, supplement was within the scope of the initial application as originally noticed. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 24, 1999, as supplemented December 7, 1999, and February 8, 2000.

Brief description of amendment: The amendment conforms the license to reflect the transfer of operating authority under Operating License No. DPR–43 to Nuclear Management Company, LLC, as approved by order of the Commission dated May 15, 2000.

Date of issuance: August 7, 2000.

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

Amendment No.: 149.

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

Date of initial notice in ***Federal Register:*** February 4, 2000 (65 FR 5706).

The February 8, 2000, supplement was within the scope of the initial application as originally noticed. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 16th day of August 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–21340 Filed 8–22–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NUREG—1718]

Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility; Notice of Availability

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: The Nuclear Regulatory Commission (NRC) has issued NUREG–1718 entitled Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility in final.

ADDRESSES: NUREG–1718 is available for inspection and copying for a fee at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Public Electronic Reading Room).

A free single copy of NUREG–1718, to the extent of supply, may be requested by writing to the U.S. Nuclear Regulatory Commission, Distribution Services, Washington, DC 20555–0001 or submitting an e-mail to distribution@nrc.gov. NUREG–1718 is available on the World Wide Web at *http://www.nrc.gov/NRC/NUREGS/indexnum.html.*

FOR FURTHER INFORMATION CONTACT: For further information regarding NUREG–1718 contact Andrew Persinko, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415–6522.

SUPPLEMENTARY INFORMATION:

Background

The NRC expects to receive a license application from Duke Cogema Stone and Webster, commonly referred to as DCS, to license a Mixed Oxide (MOX) Fuel Fabrication Facility under 10 CFR Part 70. Under Part 70, the MOX facility is classified as a plutonium processing and fuel fabrication plant. As an applicant for a license to possess and use special nuclear material (SNM) at a plutonium processing and fuel fabrication plant, DCS must obtain the NRC's approval prior to starting to construct the facility. DCS has indicated its intent to submit the license application in two parts, information for a construction permit and information for a possession and use license for SNM. The NRC will first determine if it can grant DCS construction approval. The NRC makes this determination based on contents of the license application that are specifically required by Part 70 for construction approval. The required material is described in detail in 10 CFR 70.22(f).

Following the applicant's second submittal, the NRC will determine if it can grant DCS a possession and use license for SNM. The NRC makes this determination based on the full content of the license application as described in 10 CFR 70.22 and Subpart H to the revised 10 CFR Part 70.

The NRC developed NUREG–1718 to provide guidance to the NRC staff reviewers in the Office of Nuclear Material Safety and Safeguards who will perform safety, safeguards, and environmental reviews of the anticipated application for a license to