Proposed Language: Safety-Conscious Work Environment

(a) Licensees shall establish and maintain a safety-conscious work environment in which employees are encouraged to raise safety and regulatory concerns, and where such concerns are promptly reviewed, given priority based on their potential safety significance, and appropriately resolved with timely feedback to the originator of the concern. Attributes of a safety-conscious work environment include:

(1) A management attitude that promotes employee involvement and confidence in raising and resolving concerns;

(2) A clearly communicated management policy that safety has the utmost priority, overriding, if necessary, the demands of production and project schedules;

(3) A strong, independent quality assurance organization and program;

(4) A training program that encourages a positive attitude toward safety;

(5) A safety ethic at all levels that is characterized by an inherently questioning attitude, attention to detail, prevention of complacency, a commitment to excellence, and personal accountability in safety matters.

(b) When circumstances occur that could adversely impact the safety-conscious environment, or when conditions arise that indicate the potential emergence of an adverse trend in the safety-conscious work environment, the licensee shall take action as required to ensure that the safety-conscious environment is preserved. Indicators that may be considered as possible evidence of an emerging adverse trend include, but are not limited to:

(1) Adverse findings by the Department of Labor or the NRC Office of Investigation (OI) concluding that discrimination has occurred against employees for engaging in protected activity, including a finding of the existence of a hostile work environment;

(2) A significant increase in the rate (or a sustained high number) of allegations made to the NRC that licensee employees are being subjected to harassment and intimidation for engaging in protected activity;

(3) A significant increase in the rate (or a sustained high number) of allegations made to the NRC concerning matters of safety or regulatory concern, particularly if accompanied by low usage or a decrease in use of the licensee's employee concern program (ECP) or other licensee channels for reporting safety and regulatory concerns;

(4) Other indications that the licensee's ECP or other programs for identifying and resolving safety and regulatory concerns are ineffective. Such indications might include: delays in or absence of feedback for concerns raised to the ECP; breaches of confidentiality for concerns raised to the ECP; the lack of effective evaluation, follow-up, or corrective action for concerns raised to the ECP or findings made by the licensee's QA organization; overall licensee ineffectiveness in identifying safety issues; the occurrence of repetitive or willful violations; a licensee emphasis on cost-cutting measures at the expense of safety considerations; and/or poor communication mechanisms within or among licensee groups.

(c) The presence of one or more of the indicators discussed in paragraph (b) of this section may or may not, in isolation, be considered evidence of deterioration in the licensee's safety-conscious work environment. Evaluation of the licensee's safety-conscious work environment should consider these indicators in the context of the overall work environment, including the presence or absence of other indicators, and the presence or absence of related licensee safety and performance issues.

(d) If, based on a review of indicators as discussed in paragraphs (b) and (c) of this section, the Executive Director for Operations determines that the licensee has failed to establish and maintain a safety-conscious work environment as discussed in paragraph (a) of this section, the NRC at its discretion may require the licensee to take action. This action may include (but is not limited to) ordering one or more of the following:

(1) Establishment of a formal employee concerns program (if one does not already exist);

(2) Performance of an independent survey of the licensee's environment for raising safety and regulatory concerns, with periodic follow-up surveys to monitor change;

(3) Establishment of an independent group for oversight of licensee performance in establishing and maintaining a safety-conscious work environment;

(4) Establishment of a ''holding period'' policy, to be applied in cases where an employee of the licensee or its contractor registers a complaint of having been discriminated against for engaging in protected activity. The holding period policy requires that, when such an employee submits to the licensee a complaint that he or she has been discriminated against for engaging in protected activity, the licensee will maintain that employee's pay and benefits until the licensee has investigated the complaint, reconsidered the facts, negotiated with the employee, and informed the employee of a final decision on the matter. After the licensee has informed the employee of its final decision, the holding period of continued pay and benefits will continue for an additional 2 weeks to allow a reasonable time for the employee to file a complaint of discrimination with the DOL. If, by the end of that 2-week period, the employee has filed with the DOL a complaint of discrimination for engaging in protected activity, the licensee will maintain the holding period of continued pay and benefits until the DOL has made a finding based on its initial investigation of the employee's complaint.

(5) Additional enforcement action pursuant to Subpart B of Part 2, including civil penalties.

Dated at Rockville, Maryland, this 19th day of February, 1996.

For the Nuclear Regulatory Commission.

James Lieberman,

Director, Office of Enforcement.

[FR Doc. 97–4702 Filed 2–25–97; 8:45 am]

BILLING CODE 7590–01–P

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 1, 1997, through February 13, 1997. The last biweekly notice was published on February 12, 1997 (62 FR 6567).

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 March 28, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: 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. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (*Project Director*): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts.

Date of amendment request: January 30, 1997.

Description of amendment request: The proposed amendment would change the Updated Final Safety Analysis Report (FSAR) to include the credit for containment overpressure in the Pilgrim Nuclear Power Station net positive suction head (NPSH) analysis for the emergency core cooling pumps.

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 crediting post-LOCA [loss-of-coolant accident] wetwell airspace pressure in ECCS [emergency core cooling system] analyses involve a significant increase in the probability or consequences of an accident previously evaluated?

Chapter 14 of the FSAR contains evaluations of the design basis accidents, which include the refueling accident, the main steam line break outside primary containment, the recirculation line break inside primary containment, and the control rod drop accident. No increase in the probability of the evaluated accidents will result from crediting the post-LOCA wetwell airspace pressure because post-LOCA wetwell airspace pressure does not represent an accident initiator but is rather a byproduct of the conditions which will exist in the containment after the pipe break inside containment.

The worst radiological consequences for the Pilgrim plant are associated with the design basis LOCA which is the double guillotine failure of the recirculation system piping. The radiological analysis of this event, contained in FSAR Chapter 14, uses a TID–14844 source term and assumes a 1.5% per day leakage from the containment, which is greater than the maximum leakage allowed by the Technical Specifications. The results of this analysis are presented in Table 14.5–2 of the FSAR and indicate substantial margin when compared to 10 CFR Part 100 limits.

The radiological consequences of the design basis accident are not increased by taking credit for the post-LOCA wetwell airspace pressure. Assuming containment integrity exists, the mechanism for increasing the consequences of the accident would be an increased leakage rate caused by an increase of the average differential pressure between primary and secondary containment during the accident response. However, the NPSH analyses performed for Pilgrim, which credits the post-LOCA wetwell airspace, does not require that the differential pressure between primary and secondary containment be maintained above the minimum that exists due to the equilibrium conditions based on the suppression pool temperature. Specifically, the wetwell airspace pressure credited in the ECCS pump NPSH analyses is provided by an increase in wetwell vapor pressure and air/nitrogen partial pressure in equilibrium with increasing pool temperature with an accounting for containment initial conditions and leakage.

By crediting the post-LOCA wetwell airspace pressure in the calculation of NPSH, no requirement is created to purposely maintain a higher containment pressure than would otherwise occur; no requirement is incurred to delay operating containment heat removal equipment at the highest rate possible; no requirement is incurred to deliberately continue any condition of high containment pressure to maintain adequate NPSH; and no requirement is incurred for the purposeful addition of air/nitrogen into the containment to increase the available pressure.

Based on these reasons, the probability of accidents previously evaluated is not increased, and the consequences of the design basis accident are not increased.

(2) Will crediting post-LOCA wetwell airspace pressure create the possibility for new or different kinds of accidents?

As stated above, Chapter 14 of the Pilgrim FSAR contains evaluations of design basis accidents that include the refueling accident, the main steam line break outside primary containment, the recirculation line break inside primary containment, and the control rod drop accident. New or different types of accidents are not created by crediting the post-LOCA wetwell airspace pressure because post-LOCA wetwell airspace pressure does not represent an accident initiator but is rather a byproduct of the conditions which will exist in the containment after the pipe break inside containment.

Therefore, crediting post-LOCA wetwell airspace pressure does not create the possibility for new or different kinds of accidents from those previously analyzed.

(3) Will crediting post-LOCA wetwell airspace pressure in ECCS NPSH analyses involve a significant reduction in a margin of safety?

The integrity of the primary containment and the operation of the ECCS systems in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. In order for the ECCS pumps to meet their performance requirements, the NPSH available to the pumps throughout the accident response must meet their specific NPSH requirements. Excess NPSH margin will not improve the performance of the ECCS pumps because NPSH available must only meet NPSH requirements for the pump to operate on its pump curve and meet design expectations.

Crediting post-LOCA wetwell airspace pressure in ECCS NPSH analyses increases the NPSH available to the pumps connected to the suppression pool but limits the increase in NPSH available consistent with the bounding leakage assumptions for the containment system. The amount of post-accident pressure that is utilized in ECCS NPSH analyses is calculated in a manner such that the pressure credited represents a conservative lower bound of the pressure available. Therefore, it is expected that the NPSH margin will exceed that credited in the NPSH analyses.

Credit for wetwell airspace pressure in NPSH analyses is not required under all circumstances. If the suction strainers for the ECCS pumps remain relatively free of post-LOCA debris, adequate NPSH will be available without credit for the wetwell airspace pressure provided by the post-LOCA heatup of the air/nitrogen gas in the containment. If debris accumulates on the pump suction strainers, the NPSH available to the ECCS pumps will be decreased due to the head loss caused by the debris. Credit for the post-LOCA wetwell airspace pressure in the analyses indicates that there is adequate NPSH margin such that NPSH available will remain above NPSH required, and ECCS pump performance will meet applicable requirements. Based on the above discussion, credit for wetwell airspace pressure in ECCS NPSH analyses does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied.

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

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Patrick D. Milano, Acting.

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina.

Date of amendments request: November 1, 1996.

Description of amendments request: The amendments would revise the Technical Specifications (TS) to allow full implementation of the Boiling Water Reactor Owners Group (BWROG) Enhanced Option 1–A Reactor Stability Long Term Solution. In Safety Evaluation Reports (SERs) transmitted to Kevin P. Donovan, Chairman, BWROG, by letters from Robert C. Jones, Office of Nuclear Reactor Regulation, NRC, dated June 21, 1996, and September 20, 1996, the NRC staff concluded that Enhanced Option 1–A generic technical specifications described in Topical Report NEDO–32339, Supplement 4, were acceptable for referencing in license applications.

The characteristics of a reactor system most important in determining stability performance are power, core flow and power distribution. The proposed changes would delete the current limits on power and flow conditions in the technical specifications associated with the implementation of the guidance in General Electric Service Information Letter (SIL) #380, Revision 1 and the power/flow figure (Figure 3.4.1.1–1), add two new specifications on the fraction of core boiling boundary (FCBB) and the Period Based Detection System (PBDS) and relocate certain requirements pertaining to the Average Power Range Monitors (APRM) to the Core Operating Limits Report (COLR).

The current Technical Specifications for Units 1 and 2 permit single loop operation (SLO) only for a 12-hour period and there are no provisions for potential alterations of safety limits or operating limits because of SLO conditions. Approval of the amendment applications discussed above would permit SLO operation subject to the compensatory actions and requirements that address this mode of operation in the revised Technical Specifications. However, Brunswick Unit 2's License currently has a condition, 2.C.(5) that states that the reactor shall not be made critical unless both recirculation loops are in service. This License Condition also requires the plant to be placed in the hot shutdown condition within 24 hours if one recirculation loop becomes out-of-service. The License Condition also allows one or both recirculation loops to be out-of-service for the purposes of testing (not to exceed 24 hours). Whereas the License Condition would permit SLO for up to 24 hours, the current TS limit SLO to 12 hours. The License Condition was added to permit natural circulation testing as required by the startup test program but to preclude long-term SLO or operation in the natural circulation mode. The startup test program was completed many years ago for Brunswick Unit 2 and natural circulation operation is no longer allowed. The License Condition is no longer relevant and if not deleted would negate the objectives of the proposed license amendments discussed above. The licensee has submitted proposed license amendments on the same date of the subject application (i.e., November 1, 1996) to convert the Brunswick Units 1 and 2 Technical Specifications to the Improved Standard Technical Specifications (ISTS) consistent with NUREG–1433, Revision 1, ''Standard Technical Specifications for General Electric Plants, BWR 4.'' Attachment 6 of the later application was a proposed revision of the Brunswick Unit 2 License to delete License Condition 2.C.(5). While the Notice of Consideration of Issuance of the ISTS amendments (62 FR 3719) discussed deletion of License Condition 2.C.(5), the deletion is discussed in this Notice as well, since if the subject amendment applications are approved, the License Condition would thwart the considerable effort represented by the subject amendments to finally resolve the thermal-hydraulic stability issues for Brunswick 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. The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments allow the implementation of the Enhanced Option 1–A (E1A) long term solution to the neutronic/thermal hydraulic instability issue. Current Technical Specification restrictions on power and flow conditions, number of operating recirculation loops and operator actions implemented to reduce the probability of neutronic/thermal hydraulic instability are eliminated and new stability control requirements consistent with NEDO–32339, Supplement 4, are imposed. These requirements include restrictions on power and flow conditions and actions associated with the modified APRM flow biased scram and control rod block functions. These actions include adherence to the boiling boundary limit stability control prior to entry and during operation in the region of the power and flow operating domain which is potentially susceptible to neutronic/thermal hydraulic instability in the absence of the stability control. In addition, the proposed amendments require operator actions based upon a new Period Based Detection System (PBDS). The PBDS is designed to provide alarm indication that conditions consistent with a significant degradation in the stability performance of the reactor has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist.

The proposed amendments will permit operation in regions of the power and flow operating domain postulated to be susceptible to neutron/thermal hydraulic instability (i.e., Restricted and Monitored Regions). Operation in these regions does not increase the probability of occurrence of initiators and precursors of previously analyzed accidents when neutronic/thermal hydraulic instability is not possible. The proposed amendments also permit the implementation of the features of the E1A solution which prevent neutronic/thermal hydraulic instability including pre-emptive reactor scram upon entry into the region of the power and flow operating domain most susceptible to neutronic/thermal hydraulic instability (i.e., Exclusion Region). Furthermore, the E1A solution requires implementation of stability control prior to entry into a region of the power and flow operating domain which is potentially susceptible, in the absence of stability control, to neutronic/thermal hydraulic instability (i.e., Restricted Region). The E1A solution prevents neutronic/thermal hydraulic instability during operation in regions of the power and flow operating domain previously excluded from operation and therefore does not significantly increase the probability of a previously analyzed accident.

Operation in the regions of the power and flow operating domain excluded by current Technical Specification 3/4.4.1.1 and Figure 3.4.1.1–1 can occur as a result of anticipated operational occurrences. The severity of these transients may increase in the absence of operator actions due to the potential occurrence of neutronic/thermal hydraulic instability as a result of operation in these regions. The proposed amendments will permit the implementation of the E1A long term solution to the stability issue. Required features of the E1A solution include adherence to a boiling boundary limit stability control prior to selection by the operator of APRM flow biased scram and control rod block function setpoints which allow operation in a region of the power and flow operating domain potentially susceptible, in the absence of the stability control, to neutronic/thermal hydraulic instability. Upon entry, as a result of an anticipated operational occurrence, into the region most susceptible to neutronic/thermal hydraulic instability during operation with the boiling boundary limit stability control met, the pre-emptive reactor scram prevents neutronic/thermal hydraulic instability. Therefore, the consequences of an accident do not significantly increase while operating with the stability control met. After exiting the region requiring the stability control to be met, the setpoints are automatically returned to the values applicable when anticipated operational occurrences can be initiated from conditions with the stability control not met. This automatic actuation of the more conservative setpoints ensures that the pre-emptive reactor scram will prevent operation as a result of an anticipated operational occurrence in the region most susceptible to neutronic/thermal hydraulic instability should the operator not select the more conservative setpoints appropriate for operation following exit from the region requiring stability control. These required features of the E1A solution prevent operation in the region of the power and flow operating domain most susceptible to postulated neutronic/thermal hydraulic instability by pre-emptive reactor scram regardless of how the region was entered. Therefore, the proposed amendments prevent the occurrence of neutronic/thermal hydraulic instability as a consequence of an anticipated operational occurrence and do not significantly increase the consequences of any previously analyzed accident.

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

The proposed amendments eliminate restrictions on power and flow conditions and impose alternative restrictions which permit the implementation of the E1A long term stability solution. The current restrictions on the power and flow conditions do not prevent the entry into regions of the power and flow operating domain most susceptible to neutronic/thermal hydraulic instability and therefore the possibility of neutronic/thermal hydraulic instability exists in the absence of operator action. The required features of the E1A solution implement a pre-emptive scram upon entry into the region most susceptible, without operator action, to neutronic/thermal hydraulic instability. The accessible operating domain allowed by the proposed amendments is a subset of the power and flow operating domain currently allowed. Current initiators and precursors of accidents and anticipated operational occurrences can not occur with new or different initial conditions. Therefore, the proposed amendments do not create the possibility of a new or different kind of accident from that previously evaluated.

Concurrent with the implementation of the proposed amendments, a modified Flow Control Trip Reference (FCTR) card and a new Period Based Detection System (PBDS) will be installed as required by the E1A solution. The function of the FCTR card is to aid the operator in the identification of entry into regions of the power and flow operating domain potentially susceptible to neutronic/thermal hydraulic instability and to initiate a pre-emptive scram upon entry into the regions most susceptible to neutronic/thermal hydraulic instability. This is accomplished by altering the values of setpoints of the APRM flow biased scram and the control rod block functions generated by the modified FCTR card, which are existing functions of the current FCTR card. The modified FCTR card design includes components which may be susceptible to electromagnetic interference or other environmental effects. The plant specific environmental conditions (temperature, humidity, pressure, seismic, and electromagnetic compatibility) have been confirmed to be enveloped by the PBDS environmental qualification values and will be confirmed to be enveloped by the E1A FCTR card environmental qualification values prior to installation. Therefore, the potential for spurious scrams or common mode failures induced by environmental effects (e.g., electromagnetic interference) is considered negligible. The installation of the modified FCTR card will therefore not create the possibility of a new or different kind of accident from any accident previously evaluated. The function of the PBDS is to provide the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. This is accomplished by the installation of a new PBDS card in the Neutron Monitoring System. The PBDS card takes inputs from individual local power range monitors and provides displays indicating alarm and status conditions to the operator in the control room. These displays can not create the possibility of a new or different kind of accident from any accident previously evaluated. The PBDS card design includes components which may be susceptible to electromagnetic interference or other environmental effects. The plant specific environmental conditions (temperature, humidity, pressure, seismic, and electromagnetic compatibility) have been confirmed to be enveloped by the PBDS environmental qualification values and will be confirmed to be enveloped by the E1A FCTR card environmental qualification values prior to installation. Therefore, the installation of the PBDS card will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendments do not involve a significant reduction in a margin of safety. The proposed amendments permit the implementation of the E1A long term solution to the stability issue. Under certain conditions, existing BWR designs are susceptible to neutronic/thermal hydraulic instability. General Design Criterion (GDC) 12 OF 10 CFR 50, Appendix A, requires thermal hydraulic instability to be prevented by design or be readily and reliably detected and suppressed. When the design of the reactor system does not prevent the occurrence of neutronic/thermal hydraulic instability, instability is an anticipated operational occurrence. GDC 10 of 10 CFR 50, Appendix A, requires that specified acceptable fuel design limits not be exceeded during anticipated operational occurrences.

Analyses performed by the BWROG indicate that neutronic/thermal hydraulic instability induced power oscillations could result in conditions exceeding the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) prior to detection and suppression by the current design of the Neutron Monitoring System and Reactor Protection System. To ensure compliance with GDC 12, the BWROG developed Interim Corrective Actions (ICAs) to enhance the capability of the operator to readily and reliably detect and suppress neutronic/thermal hydraulic instability. The BWROG ICAs also provided additional guidance for monitoring local power range monitors beyond the requirements of current Technical Specification 3/4.4.1.1 to ensure adequate margin to the onset of neutronic/thermal hydraulic instability. Reliance on operator actions to comply with GDC 12 was accepted on an interim basis by the NRC pending final implementation of a long term solution to the stability issue.

The modified design of the Reactor Protection System (APRM flow biased scram) implemented with the E1A solution prevents neutron/thermal hydraulic instability. The E1A solution also requires implementation of the stability control prior to entry into a region of the power and flow operating domain which is potentially susceptible, in the absence of the stability control, to neutronic/thermal hydraulic instability. As a result, the margin to the onset of neutronic/thermal hydraulic instability provided by the existing Technical Specification requirements and BWROG ICAs recommendations is not significantly reduced by the implementation of the E1A solution. The E1A solution assures compliance with GDC 12 by the prevention of neutronic/thermal hydraulic instability and therefore precludes neutronic/thermal hydraulic instability from becoming a credible consequence of an anticipated operational occurrence. The consequences of anticipated operational occurrences and the margin to the MCPR SL will not change upon the implementation of the E1A solution. Therefore, the proposed amendments do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Mark Reinhart (Acting).

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

Date of amendment request: January 20, 1997.

Description of amendment request: The proposed amendments would relocate the surveillance requirements for selected instrumentation from the Technical Specifications to licensee controlled documents because the instrumentation provides indication or an alarm only. The affected surveillance requirements are: 4.1.3.5.b, ''Control Rod Scram Accumulators''; 4.5.1.d.2.c, '' Emergency Core Cooling Systems—Operating''; 4.5.3.1.b, ''ECCS—Suppression Chamber''; and 4.6.2.1.c, ''Containment Systems—Suppression Chamber''. In addition, the proposed amendments would replace TS SR 4.4.3.2.1, ''Reactor Coolant System Leakage'' and SR 4.5.1.d.1, ''ECCS—Operating'' with surveillances more appropriate to the associated LCOs and action statements. Also, the proposed amendments add an action statement to TS 3.5.1, ''ECCS—Operating'' regarding pressure of the ADS accumulator backup compressed gas system bottle, and delete action statements 3.5.3.c, 3.5.3.d, 3.6.2.1.c and 3.6.2.1.d regarding suppression chamber water level instrumentation.

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 because:

The proposed change relocates instrumentation requirements, which provide no post-accident function from the Technical Specifications to the Bases, UFSAR, procedures, or other plant controlled documents. These requirements are part of routine operational monitoring and are not considered in the safety analysis. The Bases, UFSAR, procedures, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the Technical Specifications. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Reactor Coolant Operational Leakage limits monitoring surveillance 4.4.3.2.1 has been modified to eliminate procedural details of what instrumentation/leakage detection systems to use in verifying limits. The proposed surveillance requires verification that the reactor coolant system leakage is within limits at the same frequency as the current surveillance requirement. The reactor coolant leakage detection systems operability requirements are controlled by Technical Specification 3/4.4.3.1. Since any changes to procedures describing the method of monitoring leakage will be evaluated per the requirements of 10 CFR 50.59, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The monitoring action and the surveillance requirements added for the Automatic Depressurization System (ADS) pneumatic supply help assure the continued operability of ADS for the mitigation of accidents involving high reactor vessel pressure and the loss of the high pressure core spray system. The surveillance frequency is reasonable for the ADS supply header pressure due to the redundancy of the instrument nitrogen system, [and] several alarms [that warn] of system trouble. The ADS accumulator backup compressed gas system bottle pressure monitoring surveillance frequency and the proposed action on low bottle pressure is reasonable due to the [presence of the] ADS accumulator check valves and the [availability of the] normal ADS supply header. Therefore, this 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 because:

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the requirements will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Involve a significant reduction in the margin of safety because:

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumption. In addition, the requirements to be transposed from the Technical Specifications to procedures, or other plant controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements in the Bases, UFSAR, procedures, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no significant reduction in a margin of safety will be allowed.

Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions to these requirements proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR Standard Technical Specifications, NUREG–1434, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of instrumentation requirements ensures no significant reduction in the margin of safety.

The Reactor Coolant Operational Leakage limits monitoring surveillance 4.4.3.2.1 has been modified to eliminate procedural details of what instrumentation/leakage detection systems to use in verifying limits. The proposed surveillance requires verification that the reactor coolant system leakage is within limits at the same frequency as the current surveillance requirement. The reactor coolant leakage detection systems operability requirements are controlled by Technical Specification 3/4.4.3.1. Because there are no changes to either the reactor coolant leakage detection systems and the reactor coolant leakage continues to be maintained within the specified limits, at the required frequency, there is no reduction in the margin of safety.

The monitoring action and the surveillance requirements added for the Automatic Depressurization System (ADS) pneumatic supply help assure the continued operability of ADS for the mitigation of accidents involving high reactor vessel pressure and the loss of the high pressure core spray system. This helps assure ADS is maintained in a ready status. The previous TS SRs only tested the instrumentation, and did not verify the parameter remained within limits. Therefore, the margin of safety is not reduced.

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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

Consumers Power Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan.

Date of amendment request: January 10, 1996.

Description of amendment request: The proposed amendment would revise test requirements for the containment emergency escape airlock.

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

The following evaluation supports the finding that operation of the facility in accordance with the proposed change to the Technical Specifications would not:

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

The proposed change does not alter any plant operating conditions, operating practices, equipment design, equipment settings, or equipment capabilities. Therefore, operation of the facility in accordance with the proposed change will not involve an increase in the probability of an accident. This determination is made because the full pressure test and the seal contact check provides reasonable assurance that the Emergency Escape Airlock doors will act as designed to maintain containment integrity. Procedures are established to test seal integrity with full pressure airlock test and to verify the seal contact following the test. Acceptance criteria are established for each evolution. Failure to meet the acceptance criteria would result in corrective action to restore the Emergency Escape Airlock to the intended condition.

The proposed change defines the pressure tests required for the Emergency Escape Airlock and specifies the method used to restore the airlock door seals after full pressure testing. Due to the design of the airlock, the doors must be opened after testing. This change recognizes the practice of verifying the final integrity of the airlock by verifying door seal contact. Since the pressure test does not load the door seals in the same direction as a design basis accident, this seal contact check provides better assurance that the door is sealed than alternative pressure tests. The Emergency Escape Airlock continues to be capable of performing its design function and the consequences of those accidents previously evaluated will not increase.

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

The proposed change does not alter any plant operating conditions, operating practices, equipment design, equipment settings, or equipment capabilities. Therefore, operation of the facility in accordance with the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change requires testing of the Emergency Escape Airlock at full pressure (greater than or equal to P_a) rather than a reduced pressure between-the-seals test. This reduced pressure test is allowed by the existing Technical Specifications when the door is opened during periods when containment integrity is required. The door seal contact check and restoration will provide assurance that the Emergency Escape Airlock is capable of performing its design function after the doors are opened during recovery from full pressure testing. Implementation of these test requirements and meeting the acceptance criteria will ensure that containment integrity with respect to the Emergency Escape Airlock will be maintained. Therefore, there will be no reduction in the margin of safety.

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

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: John N. Hannon.

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

Date of amendment request: February 5, 1997 (TSC 96–11)

Description of amendment request: The proposed changes would reflect replacement of the existing nuclear instrumentation with an enhanced wide range nuclear instrumentation system that provides more channels and continuous coverage from the source to above the power range. As a result: (1) The various references to Intermediate Range of nuclear instrumentation would be eliminated and replaced with reference to Wide Range instrumentation; (2) the minimum number of operable Source and Wide Range Nuclear Instrumentation channels that are available and that are required to be operable in Table 3.5.1–1 would be increased; (3) the minimum power level specified in Note (c) of Table 3.5.1–1 would be changed from 10^{-10} amps on the intermediate range instrument channels to 4 x 10^{-4}% rated power on the wide range instrument channels; and (4) entries that specify the Wide Range Nuclear Instrumentation, the number of Required Operable Channels, reference to a new Action Statement, and Applicability would be added to Table 3.5.6–1.

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

(1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed amendment to the Oconee Technical Specifications is associated with the implementation of an enhanced nuclear instrumentation system. The new Gamma Metrics system provides twice the number of channels of neutron detectors for use during both normal plant operations and post-accident monitoring. The proposed change will make Oconee's Technical Specifications consistent with a nuclear instrumentation system that meets the reliability and redundancy requirements of Regulatory Guide 1.97. Additionally, the new Technical Specifications will be more conservative in terms of stating the minimum number of operable channels required, since there are now a greater number of redundant channels available. Assuring that the nuclear instrumentation at Oconee is more reliable and more redundant, does not affect the probability of an occurrence of an accident, since this system is a monitoring system and not an accident initiator. However, these characteristics (increased reliability and redundancy) could provide additional capability to deal with the consequences of post-accident situations.

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

No. The proposed amendment to Oconee Technical Specifications involves the implementation of an enhanced nuclear instrumentation system. By implementing a nuclear instrumentation system that meets the provisions of Regulatory Guide 1.97, Oconee's ability for neutron monitoring is enhanced during normal operations and post-accident recovery. The Source Range nuclear instrumentation system is utilized for monitoring purposes only, while the Wide Range provides a control rod withdrawal interlock based on high startup rate. The new Gamma Metrics detectors have been shown to be more reliable, accurate, and redundant than Oconee's original detectors. Therefore, changing the Oconee Technical Specifications to be consistent with the current nuclear instrumentation arrangement, as proposed in this amendment request, has no effect on the possibility of any type of accident: new, different, or previously evaluated.

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

No. Margin of safety is associated with confidence in the ability to maintain the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed Technical Specifications amendment will establish operability requirements for an enhanced nuclear instrumentation system at Oconee. By implementing a more reliable and redundant nuclear instrumentation system, Oconee's post-accident monitoring capability is enhanced. Therefore, the ability to protect the public from radiation dose is further assured, and no reduction in any existing margin of safety will occur.

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

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691.

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: October 22, 1996.

Description of amendment request: The proposed amendment would revise Figure 3.4.11–1, ''Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure,'' in Limiting Condition for Operation 3.4.11, ''RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits,'' of the Technical Specifications (TSs). The existing curve is valid only up to 10 Effective Full Power Years (EFPYs) and would be revised to be valid up to 32 EFPYs.

The proposed curves, pages 1 through 5 of Figure 3.4.11–1, have been drawn for five different EFPY periods: 16, 20, 24, 28 and 32. There are two sets of curves attached to the licensee's application. The first set of curves (Attachment 3) would replace the existing curve in TS Figure 3.4.11–1. The second set of curves (Attachment 4) are duplicates of the Attachment 3 curves except that these curves also contain detailed information used in development of the curves and would be included in the next update of the Updated Final Safety Analysis Report (UFSAR) for information.

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 application for the proposed amendment, which is presented below:

(A) The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

Regulatory Guide 1.99, Revision 2 is currently used to prepare the pressure-temperature limit curves and is inherently conservative for Boiling Water Reactors (BWRs). [Grand Gulf Unit 1 is a BWR.] The proposed Technical Specification Figure 3.4.11–1 was prepared in accordance with the requirements of 10CFR50 [10 CFR Part 50], Appendix G [(Fracture Toughness Requirements)], and using NRC approved methodology outlined in NRC Regulatory Guide 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials.'' Operation of the plant within the limitations of the proposed figure will ensure that the Requirements of 10CFR50 [10 CFR Part 50], Appendix G are met up to and including 32 Effective Full Power Years (EFPY) of operation. The proposed changes assure that the existing safety limits are not exceeded due to changing Reactor Vessel conditions by continued incorporation of the effect of neutron radiation embrittlement of vessel materials into the proposed curves.

The curves have also been editorially enhanced by removal of phrases used for validation of the curves. Having the phrases on the TS (Technical Specification) curves distracts from the intended purpose which is to maintain operation of the reactor to the right of the curves. Operators, in performance of their job function, do not need this information to comply with TS Limiting Condition for Operation (LCO) 3.4.11. This change also revises the curve labeling consistent with the terminology used in Table 1 of 10CFR50 [10 CFR Part 50], Appendix G. These enhancements and revisions have no impact on the operation of the plant since they are editorial in nature and do not change the technical content of the curves.

Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The pressure-temperature curves are controlled by the Technical Specifications and are determined using the conservative methodology in NRC Regulatory Guide 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials.'' The proposed pressure-temperature limit curves are inherently conservative, therefore, the possibility of failure of the reactor vessel is not increased. The proposed curves establish new periods of applicability (16, 20, 24, 28, and 32 EFPY) for the current pressure-temperature limitations based on NRC methodology in Regulatory Guide 1.99 and actual fluence measurements. These limitations are appropriate up to and including 32 EFPY exposure and operation of the plant within the figure's limitations will ensure that the requirements of 10CFR50 [10 CFR Part 50], Appendix G are met for that time frame. No physical plant modifications or new operating configurations result from these changes. These changes do not adversely affect the design or operation of any system or component important to safety, rather they establish limits to assure that operations remain within acceptable safety boundaries.

The curves have also been editorially enhanced by removal of phrases used for validation of the curves. Having the phrases on the TS curves distracts from the intended purpose which is to maintain operation of the reactor to the right of the curves. Operators, in performance of their job function, do not need this information to comply with TS Limiting Condition for Operation (LCO) 3.4.11. This change also revises the curve labeling consistent with the terminology used in Table 1 of 10CFR50 [10 CFR Part 50], Appendix G. These enhancements and revisions have no impact on the operation of the plant since they are editorial in nature and do not change the technical content of the curves.

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

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

The proposed curves were developed using the methodology of Regulatory Guide 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials.'' This methodology includes an allowance for margin that is to be included in the upper-bound values of the adjusted reference temperature (ART). The proposed changes maintain the existing margins of safety by modifying the operating limits based on the most limiting of the actual reference temperature shifts. These new limits consider the most limiting pressure vessel material. The revised analysis demonstrates that the existing Technical Specification [TS] pressure-temperature limit curves are applicable for periods of 16, 20, 24, 28, and 32 EFPY. Using the methodology in NRC Regulatory Guide 1.99 Revision 2 and fluence based on actual exposure provides for additional conservatism, and therefore [,] further assures the existence of current margins of safety. The proposed pressure-temperature limit curves are inherently conservative and provide sufficient margin to ensure the integrity of the reactor vessel.

The curves have also been editorially enhanced by removal of phrases used for validation of the curves. Having the phrases on the TS curves distracts from the intended purpose which is to maintain operation of the reactor to the right of the curves. Operators, in performance of their job function, do not need this information to comply with TS Limiting Condition for Operation (LCO) 3.4.11. This change also revises the curve labeling consistent with the terminology used in Table 1 of 10CFR50 [10 CFR Part 50], Appendix G. These enhancements and revisions have no impact on the operation of the plant since they are editorial in nature and do not change the technical content of the curves.

Continuing commitment to the methodology contained in NRC Regulatory Guide 1.99, Rev. 2, will ensure that the most limiting plate or beltline weld material will be utilized in the determination of the pressure-temperature limits for any future curve changes.

Therefore, the proposed change does not result in a significant reduction in a margin of safety.

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

Local Public Document Room Location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Project Director: William D. Beckner.

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana.

Date of amendment request: January 10, 1997.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) for reactor pressure vessel pressure and temperature (P–T) limits to replace the curves for 2 effective full power years (EFPY) with curves for 12 EFPY. The P–T curves are used for heatup, cooldown, and inservice leak and hydrostatic testing.

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.

Pressure-temperature (P–T) limits (RBS Technical Specifications Figure 3.4.11–1) are imposed on the reactor coolant system to ensure that adequate safety margins against nonductile or rapidly propagating failure exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P–T limits are related to the nil-ductility reference temperature, RT_{NDT}, as described in ASME Section III, Appendix G. Changes in the fracture toughness properties of [Reactor Pressure Vessel] RPV beltline materials, resulting from the neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10 CFR [Part] 50, Appendix H. The effect of neutron fluence on the nil-ductility reference temperature of pressure vessel steel is predicted by methods given in Regulatory Guide [RG] 1.99, Rev. 2.

The revised P–T limits of this amendment request were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in Reg. Guide [RG] 1.99, Rev. 2, Regulatory Position C.1. Calculation of adjusted reference temperature by these procedures includes a margin term to ensure conservative, upper-bound values are used for the calculation of the P–T limits. Stress intensity factors used to compute the pressures were calculated in accordance with, and include the required safety factors given in ASME Section III, Appendix G. The limits established by the lower portion of the P–T curves, which cover the discontinuity (non-beltline) regions of the vessel (e.g., flanges, nozzles, etc.), were retained throughout this current analysis. The limits established by the lower portion of these curves do not change as they are not affected significantly by the neutron fluence.

This change is not related to any accidents previously evaluated. The proposed change will provide for approved P–T limit curves which are valid through 12 EFPY. This change will not affect any Safety Limits, Power Distribution Limits, or Limiting Conditions for Operation. The proposed change will not affect reactor pressure vessel [RPV] performance as no physical changes are involved and RBS vessel P–T limits will remain conservative in accordance with Reg. Guide [RG] 1.99, Rev. 2 and ASME Section III, Appendix G requirements. The proposed change will not cause the reactor pressure vessel [RPV] or interfacing systems to be operated outside of their design or testing limits. Also, the proposed change will not alter any assumptions previously made in evaluating the radiological consequences of accidents. The proposed change ensures that adequate margins against brittle fracture of the vessel are maintained through 12 EFPY of reactor operations. Therefore, the probability or consequences of accidents previously evaluated will not be increased by the proposed change.

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

The proposed change is a revision of Technical Specification Figure 3.4.11–1 to show P–T limit curves valid through 12 EFPY. The revised P–T limits have been established in accordance with applicable NRC regulations and the ASME Code. This proposed change does not involve a modification of the design of plant structures, systems, or components. The proposed change will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. The proposed change will not degrade the reliability of structures, systems, or components important to safety (ITS) as equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be downgraded, the frequency of operation of ITS equipment will not be imposed. No new accident types or failure modes will be introduced as a result of the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from that previously evaluated.

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

As stated in the River Bend SER, ''Appendices G and H of 10 CFR 50 describe the conditions that require pressure-temperature [P–T] limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature [P–T] limits must provide safety margins at least as great as those recommended in the ASME Code, Section III, Appendix G. * * * Until the results from the reactor vessel surveillance program become available, the staff will use RG 1.99, Revision 1 [now Revision 2] to predict the amount of neutron irradiation damage. * * * The use of operating limits based on these criteria—as defined by applicable regulations, codes, and standards—will provide reasonable assurance that nonductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of GDC 31.''

Bases for RBS Technical Specification 3.4.11 states: ''The P/T [P–T] limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. * * * Since the P/T [P–T] limits are not derived from any DBA, there are no acceptance limits related to the P/T [P–T] limits. Rather, the P/T [P–T] limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.''

This amendment request proposes P–T limit curves which will be valid through 12 EFPY. The proposed P–T limits were established based on adjusted reference temperatures for vessel beltline material calculated in accordance with Regulatory Position 1 of Reg. Guide [RG] 1.99, Rev. 2 and pressures calculated in accordance with ASME Section III, Appendix G requirements. Required margins and safety factors were included to ensure that conservative, upper-bound values were used in calculation of the P–T limits. The proposed change will not affect any Safety Limits, Power Distribution Limits, or Limiting Conditions for Operation. The proposed change does not represent a change in initial conditions, or in a system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. The proposed P–T limits provide adequate safety margins against brittle failure of the reactor vessel through 12 EFPY of power operations. For these reasons, the proposed changes do not involve a reduction in any margins of safety.

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

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Project Director: William D. Beckner.

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana.

Date of amendment request: January 20, 1997.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to allow the use of flow control spectral shift strategies to increase cycle energy; an estimated additional 30 days at full power. The request is based on a General Electric (GE) Maximum Extended Load Line Limit (MELLL) analysis for the River Bend Station.

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 significantly increase the probability or consequences of an accident previously evaluated.

Abnormal operational transients or accidents analyzed in the SAR have been examined for any impact caused by MELLL operation. The limiting abnormal operation transients, including the Generator Load Rejection with No Bypass (LRNBP) event and the Feedwater Controller Failure (FWCF) maximum demand event, have been evaluated in detail. The LOCA [Loss-of-Coolant Accident], Fuel Loading Error (FLE), rod drop accident, rod withdrawal error, and the Anticipated Transient Without Scram (ATWS) analyses have also been evaluated for the effects of MELLL operation. The flow and power dependent [Minimum Critical Power Ratio] MCPR curves for off-rated and rated conditions and the [Maximum Average Planar Linear Heat Generation Rate] MAPLHGR criteria establish limits on power operation. These limits ensure that the core is operated within the assumptions and initial conditions of the transient or accident analyses. Operation within these limits will ensure that the consequences of a transient or accident remain within the acceptable limits of the analyses.

The [Average Power Range Monitor] APRM scram in the Technical Specifications [TSs] and affected rod block setpoints are revised to ensure that operation remains within the analyzed MELLL region. This restriction ensures the consequences of abnormal operation and accidents are acceptable. The probability of an accident is not affected by the proposed Technical Specification [TS] changes since no systems or equipment which could initiate an accident are affected. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

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

Operation in the MELLL domain expands the current power/flow along the 121% rod line to 100% power at 75% rated core flow and improves flexibility and capacity factor. Abnormal operation transients or accidents have been evaluated and the most limiting cases have been analyzed for applicability for operation in the MELLL region. The proposed Technical Specification [TS] changes prohibit power operation outside the MELLL region and do not constitute or require any system or equipment changes that might create an accident of a different type then previously evaluated. The MAPLHGR, the power and flow dependent MCPR and [Liner Heat Generation Rate] LHGR and the revised Technical Specifications [TSs] will continue to assure that plant operation is consistent with the assumptions, initial conditions and assumed power distribution and therefore will not create a new type of accident. The proposed Technical Specification [TS] changes do not introduce any new modes of plant operation nor involve new system interactions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previous analyzed.

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

The proposed Technical Specifications [TSs] prohibit power operation outside the allowable MELLL region. The transients and accidents described in the SAR are evaluated for operation in the MELLL region. NEDC–32611, ''MELLL Analysis for River Bend Station Reload 6 Cycle 7,'' shows that the OLMCPR for operation in the MELLL region is bounded by the OLMCPR established for current conditions (100% power/107% flow). The thermal limits MCPR and LHGR curves and the MAPLHGR limits establish limits on power operation and thereby ensure that the core is operated within the assumptions and initial conditions of the transient and accident analyses.

As demonstrated in the analysis provided in Attachment 4, [the proposed amendment request] operation within these limits, using the MCPR limits, LHGH limits and MAPLHGR criteria, will ensure that the margin of safety will be maintained to the same level described in the Technical Specifications Bases and the SAR and the consequences of the postulated transient or accidents are not increased. The MCPR safety limit, mechanical performance limits and overpressure limit are not exceeded during any transient or postulated accident. Therefore, the proposed Technical Specifications [TSs] to allow operation in the MELLL region do not involve a significant reduction in margin of safety.

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

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005.

NRC Project Director: William D. Beckner.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine.

Date of amendment request: February 7, 1997.

Description of amendment request: The proposed amendment would modify Technical Specification 3.12 to require both 115 kV incoming lines to be operable when the reactor is critical; allow continued operations for up to 72 hours with one 115 kV incoming line inoperable; allow continued operations for up to 24 hours with both 115 kV incoming lines inoperable; apply the increased operability requirements described above to another affected remedial action; incorporate minor editorial changes to uniformly apply the usage of the term ''operable;'' and change the basis section to be consistent with 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to Specification 3.12.B do not involve a physical change to the plant or the maintenance of the plant. The proposed changes increase the operating requirements associated with the operability of the 115 kV incoming lines beyond that currently required by Technical Specifications. For those accidents previously evaluated, the more restrictive operability requirements associated with maintaining both 115 kV incoming lines operable and the more restrictive remedial action times result in increased assurance that station service power will be available when required. This increased availability will be achieved because elective maintenance on the offsite power system will be significantly restricted and the restoration of inoperable 115 kV incoming lines will be treated with greater urgency. The increased assurance of availability will result in a decrease in the probability or consequences of these postulated accidents.

However, the more restrictive remedial action times decrease the restoration period and consequently increase the possibility that successful restoration may not be achieved, given an outage of the 115 kV power system. A unit shutdown without offsite power would then be commenced. This evolution would involve a unit shutdown without the availability of equipment such as the reactor coolant pumps, condensate pumps and main feedwater pumps. Although none of these components are credited as available for the mitigation of the consequences of accidents previously evaluated, the probability of the occurrence of certain accidents is increased without them.

Although the combination of these considerations could involve an increase in the probability of accidents previously evaluated, the increase would not be significant due to the low probability of independent failures or common cause failures of both of the 115 kV incoming lines. There is no increase in the consequences of any accident previously evaluated as a result of these proposed Technical Specification changes. The proposed Technical Specification changes are consistent with the Standard Technical Specifications approved by the NRC. The proposed changes, therefore, will 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 previously evaluated.

The proposed Technical Specification change does not involve a change to the physical plant or to the physical configuration of the offsite power system. The effect of the proposed change will be to increase the availability of the offsite power system when required. In addition, the proposed change will increase the possibility of a unit shutdown without offsite power operable. However, the accidents previously evaluated assume a simultaneous loss of offsite power, design basis accident and worst case single failure as part of the design basis. The proposed changes do not result in the creation of a unique operating condition or a configuration that has not been previously evaluated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change modifies Technical Specification 3.12 to be consistent with the Standard Technical Specifications. The proposed Technical Specification change maintains the current margin of safety which is based upon supplying power to engineered safeguards. Adequate sources of power remain available for the operation of the engineered safeguards equipment. Therefore, the proposed change 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011.

NRC Project Director: Patrick D. Milano, Acting.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–245, 50–336, and 50–423, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, New London, Connecticut.

Date of amendment request: February 3, 1997.

Description of amendment request: The licensee has proposed to revise Section 6, ''Administrative Controls,'' of the Millstone Unit Nos. 1, 2, and 3 Technical Specifications to reflect organizational changes that have been implemented in the Nuclear Division.

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

* * * The proposed changes do not involve a [significant hazards consideration] because the changes would not:

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

No design basis accidents are affected by these proposed changes. The proposed changes are administrative in nature and are being proposed to reflect the organizational changes which become effective on February 3, 1997. The unit level responsibilities of the Executive Vice President—Nuclear are assigned to the Officers for the individual Millstone units. The site level responsibilities of the Executive Vice President—Nuclear are shared by the Senior Vice President and CNO [Chief Nuclear Officer]—Millstone and the President and Chief Executive Officer. The changes to the SORC [Site Operations Review Committee] and the three unit[s'] PORC [Plant Operations Review Committee] reflect changes in job function or job position titles only.

No safety systems are adversely affected by the proposed changes, and no failure modes are associated with the changes.

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

Because there are no changes in the way plants are operated due to this administrative change, the potential for an unanalyzed accident is not created. There is no impact on plant response, and no new failure modes are introduced. These proposed administrative and editorial changes have no impact on safety limits or design basis accidents, and they have no potential to create a new or unanalyzed event. The changes to the SORC and the three unit[s'] PORC reflect changes in job function or job position titles only.

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

The changes do not directly affect any protective boundaries nor do they impact the safety limits for the protective boundaries. These proposed changes are administrative and editorial in nature. Therefore, there is no reduction in the margin of safety. These changes do not reduce the margin of safety provided by the PORC and the SORC review and approval of changes to the operations of the Millstone Unit Nos. 1, 2, and 3.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: February 5, 1996.

Description of amendment request: The amendment would delete a clause from Technical Specification 4.0.5.a. Specifically, this change would delete the clause ''(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' The amendment would also make the appropriate changes to the Bases section. In addition, NNECO made changes to Bases Section 3/4.7.7 and 3/4.7.8 to add design basis information and provide clarification of system design and 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:

Pursuant to 10 CFR 50.92, NNECO has reviewed the proposed changes to Technical Specification 4.0.5a and Bases Section 3/4.4.10 and has concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

The proposed changes would remove the wording ''* * * (g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' The Inservice Inspection and Testing Programs are described in the technical specifications pursuant to 10 CFR 50.55a. In addition, the proposed changes, in accordance with NUREG–1431 and NUREG–1482, would provide relief to the ASME Code requirement in the interim between the time of submittal of a relief request until the NRC has issued a safety evaluation and granted the relief. The changes being proposed are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect any technical specification that preserves safety analysis assumptions. Any relief from the approved ASME Section XI Code requirements will require a 10 CFR 50.59 evaluation to ensure no technical specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed changes would not affect the probability or consequences of an accident previously analyzed.

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

The proposed changes would remove the wording ''* * * (g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' The Inservice Inspection and Testing Programs are described in the technical specifications pursuant to 10 CFR 50.55a. In addition, the proposed changes, in accordance with NUREG–1431 and NUREG–1482, would provide relief to the ASME Code requirement in the interim between the time of submittal of a relief request until the NRC has issued a safety evaluation and granted relief. The changes being proposed are administrative in nature and will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. Any relief from the approved ASME Section XI Code requirements will require a 10 CFR 50.59 evaluation to ensure no technical specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes would remove the wording ''* * * (g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' The Inservice Inspection and Testing Programs are described in the technical specifications pursuant to 10 CFR 50.55a. In addition, the proposed changes, in accordance with NUREG–1431 and NUREG–1482, would provide relief to the ASME Code requirement in the interim between the time of submittal of a relief request until the NRC has issued a safety evaluation and granted relief. The changes being proposed are administrative in nature and will not alter the bases for assurance that safety-related activities are performed correctly or the basis for any technical specification that is related to the establishment or maintenance of a safety margin. Any relief from the approved ASME Section XI Code requirements will require a 10 CFR 50.59 evaluation to ensure no technical specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed changes 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Deputy Director: Phillip F. McKee.

United States Department of Commerce, National Institute of Standards and Technology, Docket No. 50–184, NIST (formerly known as National Bureau of Standards) Test Reactor or NBSR.

Date of amendment request: January 17, 1997.

Description of amendment request: The National Institute of Standards and Technology (NIST) is planning to change the name of the Reactor Radiation Division to the NIST Center for Neutron Research to be headed by a Director. The requested amendment involves a name change only. All functions, responsibilities, and personnel remain the same. The Technical Specification references to the ''Chief, Reactor Radiation Division'' will be changed to Director, NIST Center for Neutron Research in Sections 7.1, 7.2, and 7.3. The Organization Chart in Figure 7.1 will also reflect this change. The Technical Specification references to the ''Reactor Radiation Division'' will be changed to ''NIST Center for Neutron Research'' in Section 7.2.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A 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 change being proposed is a change in the title of the organization and the title of the head of the organization that directs the operation of the reactor. As noted previously, all functions, responsibilities and personnel remain the same. The staff agrees with the licensee's no significant hazards consideration and finds that the mere title changes render a negative response to the three criteria outlined in 10 CFR 50.92(c).

Local Public Document Room location: N/A.

Attorney for licensee: N/A

NRC Project Director: Seymour H. Weiss.

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

Date of amendment request: December 10, 1996.

Description of amendment request: The proposed amendment would move fire protection requirements from the Vermont Yankee Technical Specifications to the Fire Protection Plan and the final safety analysis report (FSAR), in accordance with the guidance in NRC Generic Letters 86–10 and 88–12.

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

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

The proposed changes are administrative in nature and are consistent with the guidance provided in NRC Generic Letters 86–10 and 88–12. These changes do not affect the initial conditions or precursors assumed in the FSAR safety analyses. These proposed changes also do not decrease the effectiveness of equipment relied upon to mitigate the previously evaluated accidents. Programmatic controls will continue to assure that fire protection program changes do not reduce the effectiveness of the program to achieve and maintain safe shutdown in the event of a fire.

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

The proposed changes do not modify any plant equipment, there is no reduction in fire protection requirements, there is no change in operating procedure and surveillance requirements and no reduction in administrative control or equipment reliability. Therefore, implementation of the proposed change will not affect the design function or configuration of any component, introduce any new operating scenarios, failure modes or accident initiators.

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

The proposed amendment does not involve a reduction to the Fire Protection Program. The fire protection requirements are simply being relocated to other controlled documents. There are no equipment modifications being proposed, only the location of fire protection requirements, which is administrative in nature.

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

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

Attorney for licensee: R. K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110–2624.

NRC Project Director: Patrick D. Milano, Acting Director.

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.

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

Date of amendment request: January 10, 1997.

Brief description of amendment: The proposed change would revise Technical Specification 4.8.1.1.2 to clarify pressure testing requirements for the isolable and non-isolable portions of the diesel fuel oil piping.

Date of publication of individual notice in Federal Register: February 5, 1997 (62 FR 5490).

Expiration date of individual notice: March 6, 1997.

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

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

Date of amendment request: November 6, 1996.

Description of amendment request: The proposed amendments would revise the Technical Specifications governing the cooling water system. The changes are proposed to improve plant operation based on operational experience with the vertical motor-driven cooling water pump. The changes are also proposed to incorporate information gathered by the licensee during its self-assessment Service Water System Operational Performance Inspection (SWSOPI) completed in late 1995.

Date of individual notice in the Federal Register: January 29, 1997 (62 FR 4338).

Expiration date of individual notice: February 28, 1997.

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

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

Date of amendment requests: January 29, 1997.

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

Date of individual notice in the Federal Register: February 7, 1997 (62 FR 5857).

Expiration date of individual notice: March 10, 1997. NSHC comments: February 24, 1997.

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

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

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey Date of amendment request: January 31, 1997.

Brief description of amendment request: The amendment would make changes to Technical Specification (TS) 3.4.3, ''Relief Valves,'' for Salem Unit 1, and TS 3.4.5, ''Relief Valves,'' for Salem Unit 2, to ensure that the automatic capability of the power operated relief valves to relieve pressure is maintained when these valves are isolated by closure of the block valves.

Date of publication of individual notice in Federal Register: February 7, 1997 (62 FR 5861).

Expiration date of individual notice: March 10, 1997.

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

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: October 18, 1996.

Description of amendments request: Amend Technical Specifications to permanently incorporate new requirements associated with steam generator tube inspections and repair. The requirements provide alternate steam generator tube plugging criteria at the tube support plate intersections.

Date of publication of individual notice in the Federal Register: February 11, 1997 (62 FR 6276).

Expiration date of individual notice: March 13, 1997.

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

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

Date of amendment requests: September 19, 1996, as supplemented November 18, 1996, and revised January 13 and January 27, 1997.

Description of amendment requests: The proposed amendments would change Technical Specification requirements related to the low temperature overpressure protection (LTOP) system. Specifically, the reactor coolant system (RCS) temperature below which LTOP is required to be enabled and the temperature below which one high pressure safety injection pump is required to be rendered inoperable would be changed from less than 275 degrees Fahrenheit to less than 355 degrees Fahrenheit. Additionally, the restriction of ''less than the minimum pressurization temperature for the inservice pressure test as defined in Figure 15.3.1–1'' would be deleted and the specific temperature limit of less than 355 degrees Fahrenheit would be specified. The setpoint for the pressurizer power-operated relief valves (PORVs) would be changed from less than or equal to 425 pounds per square inch gage (psig) to less than or equal to 440 psig to allow for instrument inaccuracies and increased margin allowed by the use of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Case N–514. These modified requirements for LTOP ensure that RCS materials meet the requirements of Title 10 of the Code of Federal Regulations, § 50.60, ''Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,'' (10 CFR 50.60) in accordance with 10 CFR Part 50, Appendices G and H, and in accordance with the exemption granted on January 27, 1997, which allows the use of ASME Code Case N–514 as an acceptable alternative. Finally, editorial changes would be made to rename the ''Overpressure Mitigating System'' to the ''Low Temperature Overpressure Protection System.'' The September 19, 1996, application was previously noticed in the Federal Register on October 1, 1996 (61 FR 51308).

Date of individual notice in the Federal Register: February 4, 1997 (62 FR 5256).

Expiration date of individual notice: March 6, 1997. NSHC comments February 19, 1997.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland.

Date of application for amendments: November 26, 1996.

Brief description of amendments: The amendments adopt Option B of 10 CFR Part 50, Appendix J to require Type B and Type C containment leakage testing to be performed on a performance-based testing schedule.

Date of issuance: February 11, 1997.

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

Amendment Nos.: 219 and 196.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 2, 1997 (62 FR 123).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated February 11, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts.

Date of application for amendment: April 25, 1996, as supplemented December 23, 1996.

Brief description of amendment: The amendment will revise the definition of Operable-Operability, revise Technical Specifications (TSs) and associated Bases Section for TS 3.9.B.2 and 3.9.B.3, ''Auxiliary Electrical System,'' TS 3.4.B.1, ''Standby Liquid Control System,'' TSs 3.7.b.1.a, c, and e, and 3.7.b.2.a, c, and e, ''Standby Gas Treatment System and Control Room High Efficiency Air Filtration System,'' and TSs. 4.5.F.1, ''Core and Containment Cooling Systems,'' and delete TS 3.7.b.1.f, ''Standby Gas Treatment System and Control Room High Efficiency Air Filtration System.''

Date of issuance: February 10, 1997.

Effective date: February 10, 1997.

Amendment No.: 170.

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

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31172).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–

455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois.

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

Date of application for amendments: August 2, 1996.

Brief description of amendments: The amendments eliminate License Condition 2.C.(16) from Facility Operating License NPF–37; License Condition 2.C.(5) from Facility Operating License NPF–66; License Condition 2.C.(6) from Facility Operating License NPF–72 and License Condition 2.C.(5) from Facility Operating License NPF–77 that require the licensee to conduct additional corrosion testing of sleeved steam generator tubes.

Date of issuance: February 12, 1997.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 85 to NPF–37, 85 to NPF–66, 77 to NPF–72, and 77 to NPF–77.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revise the licenses.

Date of initial notice in Federal Register: September 25, 1996 (61 FR 50340).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 12, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 26, 1996, as supplemented December 17, 1996

Brief description of amendments: The amendments revise Technical Specification 3.8.2.1 to allow a one-time change to replace the existing 125-volt AT&T high specific gravity round cell battery banks with the conventional low specific gravity cell battery banks.

Date of issuance: February 7, 1997.

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

Amendment Nos.: 172 and 154.

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

Date of initial notice in Federal Register: December 13, 1996 (61 FR 65605).

The December 17, 1996, letter did not change the scope of the November 26, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 7, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina 28223–0001.

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana.

Date of amendment request: August 29, 1996.

Brief description of amendment: The amendment revises the Technical Requirements Manual (TRM) to change the reactor pressure vessel surveillance capsule withdrawal schedule for the River Bend Station. The first capsule will be withdrawn at 10.4 effective full power years (EFPY) rather than at 6 EFPY.

Date of issuance: February 13, 1997.

Effective date: February 13, 1997.

Amendment No.: 92.

Facility Operating License No. NPF–47. The amendment revised the Technical Requirements Manual.

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55034) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 1997.

No significant hazards consideration comments received. No.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

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

Date of amendment request: June 27, 1996.

Brief description of amendment: The amendment modifies TS 3/4.3.3.6, ''Accident Monitoring Instrumentation,'' to reflect the Combution Engineering improved Standard Technical Specification (STS) approved and issued as NUREG–1432. This amendment revises the TS to include Accident Monitoring Instrumentation recommended in Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident,'' Revision 3.

Date of issuance: February 12, 1997.

Effective date: February 12, 1997, to be implemented within 90 days.

Amendment No.: 122.

Facility Operating License No. NPF–38. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 3, 1996 (61 FR 40017).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 12, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 25, 1996, as supplemented by letter dated January 27, 1997.

Brief description of amendment: The amendment changes the Appendix A Technical Specifications by modifying TS 3/4.7.4, ''Ultimate Heat Sink,'' to incorporate more restrictive fan operability requirements and lower the maximum allowed basin temperature.

Date of issuance: February 13, 1997.

Effective date: February 13, 1997.

Amendment No.: 123.

Facility Operating License No. NPF–38. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 19, 1996 (61 FR 58903).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 30, 1996.

Brief description of amendments: These amendments revise the St. Lucie Technical Specifications to remove inconsistencies between the definition of Core Alterations and the Applicability, Action and Surveillance requirements of two specifications relating to water level and containment isolation systems during refueling operations.

Date of Issuance: February 10, 1997.

Effective Date: February 10, 1997.

Amendment Nos.: 148 and 87.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 4, 1996 (61 FR 64386).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 10, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

Date of application for amendment: October 28, 1996.

Brief description of amendment: The amendments consist of changes to the Technical Specifications (TS) in response to your applications, both dated October 28, 1996, regarding containment leakage tests and removal of certain component lists from the TS.

Date of Issuance: February 10, 1997.

Effective Date: February 10, 1997.

Amendment Nos.: 149 and 88.

Facility Operating License No. NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: (61 FR 64386) December 4, 1996. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

Date of application for amendments: December 17, 1996.

Brief description of amendments: Revision to Technical Specification (TS) 4.4.10 regarding reactor coolant pump flywheel inspection intervals.

Date of issuance: February 11, 1997.

Effective date: February 11, 1997.

Amendment Nos.: 193 and 187.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revise the Technical Specifications.

Date of initial notice in Federal Register: January 10, 1997 (62 FR 1476). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 11, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

Date of application for amendment: July 16, 1996.

Brief description of amendment: The amendment changes the Technical Specifications to permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing in accordance with the implementation guidance in NRC's Regulatory Guide 1.163 dated September 1995.

Date of issuance: February 10, 1997.

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

Amendment No.: 159.

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

Date of initial notice in Federal Register: October 9, 1996 (61 FR 52965). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 15, 1996.

Brief description of amendments: The amendments revise the containment cooling systems limiting conditions for operation technical specifications to bring them into conformance with recently completed system analyses by no longer permitting both containment spray pumps to be inoperable at the same time.

Date of issuance: February 10, 1997.

Effective date: February 10, 1997, with full implementation within 30 days.

Amendment Nos.: 125 and 117.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 4, 1996 (61 FR 64388).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 10, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 25, 1996.

Brief description of amendments: These amendments revise the wording in TS Section 4.8.1.1.2.e.2 and the associated TS Bases Section 3/4.8, to remove the specific reference to the Residual Heat Removal (RHR) pump motor and its corresponding kW rating value, and replace it with wording consistent with that specified in the Improved TS (i.e., NUREG–1433, Revision 1, ''Standard Technical Specifications General Electric Plants,'' dated April 1995).

Date of issuance: February 4, 1997.

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

Amendment Nos.: 121 and 85.

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

Date of initial notice in Federal Register: December 18, 1996 (61 FR 66716).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 4, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 27, 1996.

Brief description of amendments: These amendments increase the reactor enclosure secondary containment maximum inleakage rate, and also impact secondary containment drawdown time and system flow rate assumptions, thereby, affecting charcoal filter bed efficiency and post accident dose analysis.

Date of issuance: February 11, 1997.

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

Amendment Nos.: 122 and 86.

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

Date of initial notice in Federal Register: December 4, 1996 (61 FR 64392).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 11, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 6, 1996, as supplemented by letters dated January 15, and 28, 1997.

Brief description of amendment: This amendment modifies Technical Specification (TS) Section 2.1 and its associated TS Bases to reflect the change in the Minimum Critical Power Ratio safety limit due to the use of GE13 fuel product line and the cycle-specific analysis performed by General Electric Company (GE), for LGS, Unit 2, Cycle 5.

Date of issuance: February 12, 1997.

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

Amendment No.: 87.

Facility Operating License No. NPF–85. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 23, 1996 (61 FR 67582).

The January 15, and 28, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 12, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 10, 1996, as supplemented June 24, July 1, August 13, September 20, and October 17, 1996.

Brief description of amendments: The amendments change Technical Specifications 3/4.3.3.1, ''Radiation Monitoring Instrumentation,'' and 3/4.7.6, ''Control Room Emergency Air Conditioning System,'' to reflect a control room design in which the common Unit 1 and Unit 2 control room envelope is supplied by 2 one hundred percent capable Control Room Emergency Air Conditioning System trains.

Date of issuance: February 6, 1997.

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

Amendment Nos.: 190 and 173.

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

Date of initial notice in Federal Register: June 24, 1996 (61 FR 32468) The June 24, July 1, August 13, September 20, and October 17, 1996, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor expand the scope of the initial submittal as described in the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 6, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 31, 1996, as supplemented December 23, 1996.

Brief description of amendments: The amendments change the Technical Specification to (1) Revise the reactor vessel level indication system action statements, (2) revise the channel calibration definition, and (3) delete a requirement to install la jumper in the auxiliary feedwater actuation logic.

Date of issuance: February 6, 1997.

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

Amendment Nos.: 191 and 174.

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

Date of initial notice in Federal Register: June 17, 1996 (61 FR 30641).

The December 23, 1996, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 6, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: November 15, 1996.

Brief Description of amendments: The amendments replace Containment Systems TS 3.6.2.2 for the Spray Additive System, with a new Emergency Core Cooling Systems (ECCS) TS 3.5.6 for the ECCS Recirculation Fluid pH Control System.

Date of issuance: February 3, 1997.

Effective date: As of the date of issuance to be implemented prior to Mode 4 for Unit 1 following the spring 1997 refueling outage; for Unit 2 following the spring 1998 refueling outage.

Amendment Nos.: 123 and 118.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

Date of initial notice in Federal Register: December 18, 1996 (61 FR 66718).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 3, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

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

Date of application for amendment: August 7, 1996.

Brief description of amendment: The amendment revises Technical Specification (TS) 1.0, ''Definitions,'' by defining a refueling interval to be [less than or equal to] 730 days; and revises TS 3/4.0, ''Applicability,'' TS 3/4.6.2.1, ''Containment Systems—Depressurization and Cooling Systems—Containment Spray System,'' and TS 3/4.6.3.1, ''Containment Systems—

Containment Isolation Valves,'' to reflect performing surveillance tests during a refueling interval rather than every 18 months.

Date of issuance: February 10, 1997.

Effective date: February 10, 1997, to be implemented not later than 120 days after issuance.

Amendment No.: 213.

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

Date of initial notice in Federal Register: October 9, 1996 (61 FR 52970).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 12, 1996.

Brief description of amendment: The amendment revised Technical Specifications (TS) 3/4.1.3.4, ''Reactivity Control Systems—Rod Drop Time,'' and TS 3/4.5.2, ''Emergency Core Cooling Systems—Tavg [greater than or equal to] 280°F,'' to change the surveillance test interval from every 18 months to each refueling interval ([less than or equal to] 730 days, nominally 24 months). Additionally, the amendment removed a footnote for TS 4.5.2.b that is no longer applicable.

Date of issuance: February 11, 1997.

Effective date: February 11, 1997, to be implemented not later than 120 days over issuance.

Amendment No.: 214.

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

Date of initial notice in Federal Register: October 9, 1996.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 11, 1997.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 19th day of February 1997.

For the Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 97–4573 Filed 2–25–97; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Request For Public Comment

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

Reapproval:

Rule 24b–2
SEC File No. 270–153
OMB Control No. 3235–0127

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.), the Securities and Exchange Commission (''Commission'') is publishing the following summary of collection for public comment.

Rule 24b–2 (17 CFR 240.25b–2) provides a procedure, whereby persons filing documents with the Commission may request confidential treatment of information contained in such documents, and may request Commission review of adverse staff determinations regarding the confidential treatment request.

Approximately 630 requests for confidential treatment are made per year. Applications pursuant to the rule are generally prepared in conjunction with the document for which confidential treatment is being requested. Based upon our review of the applications we have received, we believe that not more than 30 minutes of the time spent in preparing the entire filing may be attributed to the application required under Rule 24b–2. Thus, the total compliance burden is 315 hours. The approximate cost per hour is $100, resulting in a total cost of compliance for respondents of $31,500 per year (315 hours @$100).

Written comments are invited on: (a) whether the proposed collection of information is necessary for the proper performance of the functions of the agency, including whether the information shall have practical utility; (b) the accuracy of the agency's estimate of the burden of the proposed collection of information; (c) ways to enhance the quality, utility, and clarity of the information to be collected; and (d) ways to minimize the burden of the collection of information on respondents, including through the use of automated collection techniques or other forms of information technology. Consideration will be given to comments and suggestions submitted in writing within 60 days of this publication.

Direct your written comments to Michael E. Bartell, Associate Executive Director, Office of Information Technology, Securities and Exchange Commission, 450 5th Street, NW., Washington, DC 20549.

Dated: February 19, 1997.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 97–4748 Filed 2–25–97; 8:45 am]

BILLING CODE 8010–01–M

[Rel. No. IC–22521; 813–152]

Partners Income Fund; Notice of Application

February 20, 1997.

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

ACTION: Notice of application for exemption under the Investment Company Act of 1940 (the ''Act'').

APPLICANT: Partners Income Fund (the ''Initial Partnership'').

RELEVANT ACT SECTIONS: Order requested under section 6(b).

SUMMARY OF APPLICATION: Applicant requests an order that would amend a prior order to permit the employer of certain employees' securities companies to invest in those companies on terms no more favorable than those available to eligible employees.

FILING DATES: The application was filed on August 6, 1996 and amended on November 26, 1996.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicant with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on March 17, 1997 and should be accompanied by proof of service on the applicant, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the SEC's Secretary.

ADDRESSES: Secretary, SEC, 450 5th Street, N.W., Washington, D.C. 20549. Applicant, c/o McKinsey & Company,