Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting Mr. Sam Duraiswamy (telephone 301/415–7364), between 7:30 a.m. and 4:15 p.m. EDT.

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

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

Dated: May 14, 1999.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 99–12642 Filed 5–18–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 24, 1999, through May 7, 1999. The last biweekly notice was published on May 5, 1999.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed no Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

By June 18, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: March 3, 1999.

Description of amendment request: The proposed amendment would change the reactor vessel (RV) surveillance capsule pull interval from approximately 15 effective full power (EFPY) years to 18 EFPY in Technical Specification (TS) Table 4.6–3.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below: The operation of Pilgrim in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The Pilgrim plant's physical configuration and operational practices are not changed by this proposed change. The licensee is only proposing to change the TS withdrawal schedule for the RV surveillance capsule. This change does not affect any of the current accident mitigation features of the facility or the sequence of any accidents previously analyzed. For the reasons given above, deferral of withdrawal of Pilgrim's second capsule for at least one additional cycle (or 3 EFPY) does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operation of Pilgrim in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. As discussed in the above narrative, the deferral of the second capsule pull at Pilgrim does not change any of the design features or operation of the facility but does defer a TS surveillance. Pilgrim's current TS pressure-temperature (P–T) curves are conservative and will remain so even if the RV surveillance capsule is not pulled this outage. The data from the first RV capsule supports this conclusion. Because the RV capsule pull schedule is being deferred, the P–T curves, which can be modified based on the data from the RV capsule surveillance, will not be changed. The deferral of the withdrawal of Pilgrim's second RV surveillance capsule does not change the design features or operation of the facility and the existing P–T curves have not changed, therefore, the TS change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The capsule pull is a surveillance technique that provides data for modification of the P–T curves. The methods used to develop the temperatures associated with these curves are regarded as conservative. The data from the first RV capsule supported this conclusion. Because the P–T curves have not changed and have been determined to be conservative, the margins of safety that were previously established have not changed. Therefore, deferral of the withdrawal of Pilgrim's second RV surveillance capsule will not involve a significant reduction in the margin of safety.

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, 132 South Street, Plymouth, Massachusetts 02360.

Attorney for licensee: J. Fulton, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: March 30, 1999.

Description of amendment request: The proposed amendment would revise Section 4.0, Surveillance Requirements, of the Technical Specifications (TSs). Specifically, Section 4.0.2 would be added to allow a 24-hour grace period for performing inadvertently missed surveillance.

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

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

Response: No. This proposed change will result in either the plant condition either remaining unchanged (i.e., the system or component is declared operable) or in the plant proceeding to a shutdown condition (i.e., the system or component is declared operable). If at the end of the 24-hour interval, it is necessary to proceed to shutdown, this shutdown is indistinguishable from any shutdown where a system or component is declared inoperable. Allowing an additional 24 hours to perform the surveillance balances the risks associated with an allowance for completing the surveillance within this 24-hour period against the risks associated with the potential for a plant upset and challenge to safety systems when the alternative is a shutdown to comply with the action requirements before the surveillance can be completed. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No. This proposed change will result in either the plant condition either remaining unchanged (i.e., the system or component is declared operable) or in the plant proceeding to a shutdown condition (i.e., the system or component is declared operable). If at the end of the 24-hour interval, it is necessary to proceed to shutdown, this shutdown is indistinguishable from any shutdown where a system or component is declared inoperable. Allowing an additional 24 hours to perform the surveillance balances the risks associated with an allowance for completing the surveillance within this 24-hour period against the risks associated with the potential for a plant upset and challenge to safety systems when the alternative is a shutdown to comply with the action requirements before the surveillance can be completed. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No. This proposed change will result in either the plant condition either remaining unchanged (i.e., the system or component is declared operable) or in the plant proceeding to a shutdown condition (i.e., the system or component is declared operable). If at the end of the 24-hour interval, it is necessary to proceed to shutdown, this shutdown is indistinguishable from any shutdown where a system or component is declared inoperable. Allowing an additional 24 hours to perform the surveillance within this 24-hour period against the risks associated with the potential for a plant upset and challenge to safety systems when the alternative is a shutdown to comply with the action requirements before the surveillance can be completed . Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: July 22 and October 22, 1998; May 6, 1999.

Description of amendment request: The amendments would revise the Technical Specifications (TS) to reflect the licensee's planned use of fuel supplied by Westinghouse. The staff has published a Notice of Consideration of Issuance of Amendments and Proposed No Significant Hazards Consideration Determination on November 18, 1998 (63 FR 64108) covering the July 22 and October 22, 1998, submittals. In the May 6, 1999, submittal the licensee proposed to expand the original amendment request, revising Section 5.6.5 of the Technical Specifications. Section 5.6.5 specifies a list of NRC-approved topical reports that the licensee is required to use to determine reactor core operating limits. The licensee proposed to update this list to show the current approval status of these topical reports.

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 for the proposed changes conveyed by the May 6, 1999, submittal. The NRC staff has reviewed the licensee's analyses against the standards of 10 CFR 50.92(c). The NRC-staff's analysis is presented below.

First Standard

No. The proposed changes to Section 5.6.5 will not affect the safety function and will not involve any change to the design or operation of any plant system or component. The topical reports were previously approved by the NRC staff under separate licensing actions. The use of methodologies in these approved topical reports will ensure that previously evaluated accidents remain bounding. Therefore, no accident probabilities or consequences will be impacted.

Second Standard

No. The proposed changes would not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

Third Standard

No. Margin of safety is associated with confidence in the design and operation of the plant; specifically, the ability of the fission product barriers to perform their design functions during and following an accident. The proposed changes to Section 5.6.5 do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied for the proposed changes to Section 5.6.5. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Ms. Lisa F. Vaughn , Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 5, 1999.

Description of amendment request: The proposed amendments would provide revised spent fuel pool storage configurations, revised spent fuel pool storage criteria, and revised fuel enrichment and burnup requirements which take credit for soluble boron in maintaining acceptable margins of subcriticality in the spent fuel storage pools. Also, the proposed amendments would provide additional criteria for ensuring acceptable levels of subcriticality in the spent fuel storage pools.

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

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

No, based upon the following:

Dropped Fuel Assembly

There is no significant increase in the probability of a fuel assembly drop accident in the spent fuel pools when considering the degradation of the Boraflex panels in the spent fuel pool racks coupled with the presence of soluble boron in the spent fuel pool water for criticality control. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water, and the quantity of Boraflex remaining in the racks has no affect on the probability of such a drop accident.

The criticality analysis showed that the consequences of a fuel assembly drop accident in the spent fuel pools are not affected when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron.

Fuel Misloading

There is no significant increase in the probability of the accidental misloading of spent fuel assemblies into the spent fuel pool racks when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron in the pool water for criticality control. Fuel assembly placement and storage will continue to be controlled pursuant to approved fuel handling procedures to ensure compliance with the Technical Specification requirements. These procedures will be revised as needed to comply with the revised requirements which would be imposed by the proposed Technical Specification changes.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools. A McGuire Station UFSAR change will revise Chapter 16, ''Selected Licensee Commitments'', to provide for adequate monitoring of the remaining Boraflex in the spent fuel pool racks. If that monitoring identifies further reductions in the Boraflex panels which would not support the conclusions of the McGuire Criticality Analysis, then the McGuire TS's and design bases would be revised as needed to ensure that acceptable subcriticality are maintained in the McGuire spent fuel storage pools.

Significant Change in Spent Fuel Pool Temperature

There is no significant increase in the probability of either the loss of normal cooling to the spent fuel pool water or a decrease in pool water temperature from a large emergency makeup when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the spent fuel pool water. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

A loss of normal cooling to the spent fuel pool water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density that would result in a decrease in reactivity when Boraflex neutron absorber panels are present in the racks. However, since a reduction in the amount of Boraflex present in the racks is considered, and the spent fuel pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. However, the additional negative reactivity provided by the current boron concentration limit, above that provided by the concentration required to maintain k_{eff} less than or equal to 0.95 (1170 ppm), will compensate for the increased reactivity which could result from a loss of spent fuel pool cooling event. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a loss of normal cooling to the spent fuel pool will not be increased. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

A decrease in pool water temperature from a large emergency makeup causes an increase in water density that would result in an increase in reactivity when Boraflex neutron absorber panels are present in the racks. However, the additional negative reactivity provided by the current boron concentration limit, above that provided by the concentration required to maintain k_{eff} less than or equal to 0.95 (1170 ppm), will compensate for the increased reactivity which could result from a decrease in spent fuel pool water temperature. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a decrease in pool water temperature will not be increased. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

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

No. Criticality accidents in the spent fuel pool are not new or different types of accidents. They have been analyzed in Section 9.1.2.3 of the Updated Final Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments for fuel enrichments up to 4.75 weight percent U–235. Specific accidents considered and evaluated include fuel assembly drop, accidental misloading of spent fuel assemblies into the spent fuel pool racks, and significant changes in spent fuel pool water temperature. The accident analysis in the Updated Final Safety Analysis Report remains bounding.

The possibility for creating a new or different kind of accident is not credible. The amendment proposes to take credit for the soluble boron in the spent fuel pool water for reactivity control in the spent fuel pool while maintaining the necessary margin of safety. Because soluble boron has always been present in the spent fuel pool, a dilution of the spent fuel pool soluble boron has always been a possibility, however this accident was not considered credible. For the proposed amendment, the spent fuel pool dilution evaluation (Attachment 7) demonstrates that a dilution of the boron concentration in the spent fuel pool water which could increase the rack k_{eff} to greater than 0.95 (constituting a reduction of the required margin to criticality) is not a credible event. The requirement to maintain boron concentration in the spent fuel pool water for reactivity control will have no effect on normal pool operations and maintenance. There are no changes in equipment design or in plant configuration. This new requirement will not result in the installation of any new equipment or modification of any existing equipment. Therefore, the proposed amendment will not result in the possibility of a new or different kind of accident.

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

No. The proposed Technical Specification changes and the resulting spent fuel storage operating limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis (Attachment 6) based on the ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology'' described in Reference 1. The Westinghouse methodology for taking credit for soluble boron in the spent fuel pool has been reviewed and approved by the NRC (Reference 6). This methodology takes partial credit for soluble boron in the spent fuel pool and requires conformance with the following NRC Acceptance criteria for preventing criticality outside the reactor:

(1) k_{eff} shall be less than 1.0 if fully flooded with unborated water which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level; and

(2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

The criticality analysis utilized credit for soluble boron to ensure k_{eff} will be less than or equal to 0.95 under normal circumstances, and storage configurations have been defined using a 95/95 k_{eff} calculation to ensure that the spent fuel rack k_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including uncertainties, tolerances and accident conditions in the presence of spent fuel pool soluble boron. The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to exceeding a k_{eff} of 0.95 has been evaluated (Attachment 7) and shown to be not credible. Accordingly, the required margin to criticality is not reduced.

The evaluations in Attachment 7, which show that the dilution of the spent fuel pool boron concentration from the conservative assumed initial boron concentration (2475 ppm) to the minimum boron concentration required to maintain k_{eff} [less than or equal to] 0.95 (440 ppm) is not credible, combined with the 95/95 calculation which shows that the spent fuel rack k_{eff} will remain less than 1.0 when flooded with unborated water, provide a level of safety comparable to the conservative criticality analysis methodology required by References 2, 3 and 4.

Therefore the proposed changes in this license amendment will not result in a significant reduction in the plant's 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: J. Murray Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 6, 1999.

Description of amendment request: The proposed amendments would expand the allowable values for Interlocks P–6 (Intermediate Range Neutron Flux) and P–10 (Power Range Neutron Flux) in TS 3.3.1, Table 3.3.1–1, Function 16, Reactor Trip System Interlocks, as recommended by Westinghouse.

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; 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.

Criterion 1—Would operation of the facility in accordance with the requested amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system functions are not bypassed during unit conditions under which the safety analysis assumes the functions are not bypassed. The proposed changes involve changing the lower value of the P–10 permissive (power range (PR) neutron flux) allowable values from [greater than or equal to] 9% RTP to [greater than or equal to] 7% RTP, and changing the P–6 permissive (intermediate range (IR) neutron flux) allowable value from [greater than or equal to] 6E11 amp to [greater than or equal to] 4E–11 amp. Changing the P–10 allowable value would allow for tripping and resetting of the permissive at a lower reactor power level. Changing the P–6 allowable value would allow the source range (SR) channels to be blocked at a lower increasing reactor power level and delay resetting of the permissive at a lower decreasing reactor power level.

A review of the UFSAR Chapter 15 accident analyses determined that no credit is taken for the SR reactor trip or the IR reactor trip for any of the UFSAR accidents. Credit is taken for the PR low setpoint trip for a feedwater system malfunction causing an increase in feedwater flow accident (15.1.2), uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition accident (15.4.1), and spectrum of rod cluster control assembly ejection accidents (15.4.8). All three of these accident scenarios are bounded by cases at 0% RTP taking credit for the PR low setpoint trip and cases at [greater than or equal to] 10% RTP taking credit for the PR high setpoint trip. The uncontrolled rod cluster control assembly bank withdrawal from power accident (15.4.2) analyses are performed at initial power levels of 10%, 50%, and 100% RTP to demonstrate that acceptable results are obtained for a range of initial power levels. For this accident, the PR neutron flux high setpoint trip, high pressurizer pressure trip, overpower delta-T (OPDT) trip and overtemperature delta-T (OTDT) trip provide core protection. With the P–10 reset function changed to as low as 7% RTP, the conclusions of Section 15.4.2 analysis would not change. Since the uncontrolled bank withdrawal event is analyzed from both zero power and 10% RTP, all low power initial conditions are adequately bounded. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

Criterion 2—Would operation of the facility in accordance with the requested amendment create the possibility of a new or different kind of accident from any previously evaluated?

The proposed changes to the allowable values will provide adequate deadbands between the trip and reset setpoints as well as adequate margin for instrument drift. The reactor trip system overpower trips continue to perform their safety function as assumed in safety analyses. Only the permissives (P–6 and P–10) for blocking and unblocking of overpower reactor trips are changed. The proposed changes will not invalidate any of the UFSAR accident analyses. The proposed changes will not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—Would operation of the facility in accordance with the requested amendment involve a significant reduction in a margin of safety?

The proposed changes involve lowering the Technical Specification allowable values associated with the P–10 and P–6 permissives for blocking and unblocking of reactor overpower trips. The lowering of these allowable values is not considered a significant reduction since it is just enough to accommodate a deadband recommended by Westinghouse and a margin for instrument drift. The proposed changes will not invalidate any UFSAR Chapter 15 accident analyses. Therefore, the proposed changes will not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 26, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specifications to provide a method for obtaining a Nuclear Regulatory Commission review of (a) the analytical details regarding a revised methodology for determining steam generator tube loads following a main steam line break, and (b) the crediting of the main steam line break detection and feedwater isolation instrumentation as a means for providing runout protection for the turbine-driven emergency feedwater pump.

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

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

No. The proposed changes involve: (a) revising the methodology utilized to determine steam generator tube loads following a main steam line break (MSLB); and (b) utilizing the MSLB detection and feedwater isolation instrumentation as an additional means of providing runout protection of the turbine-driven emergency feedwater (EFW) pump.

The revised methodology utilized to determine steam generator tube loads following a MSLB is consistent with the methodology utilized in the MSLB containment response analysis which has received Nuclear Regulatory Commission (NRC) approval. The revised MSLB analysis reaches the same conclusion as the original analysis (i.e., steam generator tube integrity is maintained). The new analysis takes into consideration the operation of the MSLB detection and feedwater isolation instrumentation to terminate main feedwater (MFW) flow and inhibit the auto-start of or auto-stop the turbine-driven EFW pump. This instrumentation is QA–1, whereas the Integrated Control System (ICS) is non-safety. Furthermore, the revised MSLB analysis results in a greater temperature difference between the steam generator tube and shell, thus, more conservative steam generator tube loads than those identified in the original MSLB analysis.

Also, in the event that the MSLB detection and feedwater isolation instrumentation does not function properly, the non-safety ICS is still available to maintain steam generator water level at the post-trip minimum level as assumed in the original analysis.

Currently, operator action is the only credited means to protect the turbine-driven RFW pump from runout. The MSLB detection and feedwater isolation instrumentation provides an additional method to protect the turbine-driven EFW pump from runout. Crediting the MSLB detection and feedwater isolation instrumentation simply adds defense in depth.

There are no physical changes to the plant structures, systems, or components (SSCs) or operating procedures, nor are there any changes to safety limits or set points. Also, no new radiological release pathways are created.

Thus, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

No. The reanalysis of the steam generator tube loads following a MSLB accident is limited to an accident that is already evaluated in the UFSAR. The methodology is similar to the current analysis for the MSLB containment response. The effects of the MSLB on steam generator tube integrity are the same as in the original analysis—tube integrity is maintained.

The revised analysis takes into consideration the operation of the MSLB detection and feedwater isolation instrumentation, which terminates MFW flow and inhibits the auto-start of or auto-stops the turbine-driven EFW pump following a MSLB. As assumed in the original analysis, the non-safety ICS will remain available to control steam generator water level at the post-trip minimum level should a malfunction occur in the MSLB detection and mitigation circuit. Should this malfunction occur, the resulting tube stresses would decrease relative to the revised analysis.

Crediting the MSLB detection and feedwater isolation instrumentation as a means to protect the turbine-driven EFW pump from runout simply adds defense in depth.

There are no physical changes to the plant SSCs or operating procedures. There are no new hazardous materials or potential missiles. It does not introduce the possibility of any new or different malfunctions. No safety limits or set points are changed.

Thus, the proposed change does not create the possibility of a new or different kind of accident.

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

No. The reanalysis of the steam generator tube loads following a MSLB accident is similar to the current analysis for the previously NRC approved MSLB containment response. The conclusion of the revised MSLB steam generator tube load analysis is the same as the conclusion in the original analysis—steam generator tube integrity is maintained.

Crediting the MSLB detection and feedwater isolation instrumentation as a means to protect the turbine-driven EFW pump from runout simply adds defense in depth.

There are no safety limit, set point, design parameters, or operating procedure changes required. The integrity of the fuel cladding, reactor coolant system, and containment are preserved.

Thus, the proposed change does not involve a significant reduction in a margin of safety.

Duke has concluded based on the above information that there are no significant hazards involved in this LAR.

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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

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

NRC Section Chief: Richard L. Emch, Jr.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: April 9, 1999.

Description of amendment request: The proposed amendment would revise the requirements affecting the surveillance methods for the containment tendons, the conduct of containment visual inspections, and the reporting methods employed in disseminating the results of these inspections to the Nuclear Regulatory Commission.

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

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

The proposed change to the ANO–1 [Arkansas Nuclear One, Unit 1] TS [Technical Specifications] replaces previous requirements and commitments to establish a containment inspection program based on the guidance provided in Regulatory Guide 1.35, Revision 2 in favor of regulations depicted in [Title] 10 [of the] CFR [Code of

Federal Regulations] 50.55a(g)(6)(ii)(B) and 50.55a(b)(2)(ix). ANO–1 is implementing a containment inspection program to comply with these new regulatory requirements. The final rule specifies requirements to assure that the critical areas of the containment structure are routinely inspected to detect and take corrective action for defects that could compromise structural integrity.

Maintaining reactor building structural integrity is independent of the operation of the reactor coolant system (RCS), the reactor protection system (RPS) and emergency core cooling system (ECCS). The reactor building is not considered to be the initiator of any accident previously evaluated. The physical location of inspection details does not prevent or inhibit the reactor building from functioning as designed to provide an acceptable barrier against release of radioactive materials to the environment. Through appropriate inspections and implementation of corrective actions for any degradation discovered during the inspections that might lead to containment structural failures, the probability or consequences of accidents will not be increased.

Therefore, the removal of inspection details from the TS does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Maintaining containment structural integrity is independent of the operation of the RCS, the RPS and ECCS. The proposed changes do not change the design, configuration, or method of operation of the plant. By implementing corrective actions for any degradation discovered during the required inspections of the containment, the possibility of a new or different kind of accident will not be created. Implementation of the requirements of Subsection IWL of the ASME [American Society of Mechanical Engineers] code and those of 10 CFR 50.55a(g)(6)(ii)(B) and 50.55a(b)(2)(ix) provide an equally acceptable containment inspection program.

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

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

The removal of the level of detail currently found in the ANO–1 TS regarding reactor building inspections and incorporating the applicable requirements of Subsection IWL of the ASME code and of 10 CFR 50.55a(g)(6)(ii)(B) and 50.55a(b)(2)(ix) into the ANO–1 containment inspection program has no impact on any safety analysis assumptions. Requirements associated with containment inspections are controlled by safety related procedure 5220.011. Sufficient controls exist under the procedure change process at ANO–1 to ensure current and future regulations and commitments are properly addressed when making revisions to the containment inspection procedure. The addition of structural integrity requirements to ANO–1 TS Specification 3.6.1 imposes consistent requirements with those previously specified in the ANO–1 TSs. The containment inspection program ensures that the containment will function as designed to provide an acceptable barrier against release of radioactive materials to the environment. Through the implementation of the containment inspection program, the existing margin of safety is preserved.

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

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

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: April 9, 1999.

Description of amendment request: The proposed amendment would revise the requirements associated with the station batteries and the direct current (dc) sources to the 125 volt dc switchyard distribution system.

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

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

The switchyard 125V DC control power source requirements do not meet the criteria for inclusion in Technical Specifications (TSs) as evaluated with respect to the selection criteria of [Title] 10 [of the] CFR [*Code of Federal Regulations*] 50.36. These control power sources are not assumed to mitigate accident or transient events. The effects of a loss of these control power sources are enveloped by the Loss of Offsite Power (LOOP) event and relocation is considered to have a non-significant impact on the probability or severity of a LOOP event. These requirements will be relocated from the TSs to an appropriate administratively controlled document and maintained pursuant to 10 CFR 50.59.

Proposed changes incorporating the requirements of TS 3.7.1.D, 3.7.2.E, 3.7.2.F, and 3.7.2.A, as related to the DC electrical power subsystems, in the new TS 3.7.3 results in a more stringent requirement for the ANO–1 [Arkansas Nuclear One, Unit 1] TSs in that reductions to lower conditions of operation in shorter periods of time are now required. These more stringent requirements are not assumed to be initiators of any analyzed events and will not alter assumptions relative to mitigation of accident or transient events.

The proposed addition of TS 3.7.4 allowing continued operation for a limited period of time with battery cell parameters not within limits under certain conditions clarifies an allowance that currently exists in the ANO–1 TS due to the absence of acceptance criteria for the battery cell parameter surveillances.

Proposed changes in Surveillance Requirements and Frequencies reflect current industry guidance on maintenance and testing of the station batteries. These requirements, in themselves, are not considered to be initiators of any analyzed accident condition. Although some frequencies have been extended, continued performance of maintenance activities in accordance with IEEE–450 [Institute of Electrical and Electronic Engineers, ''Recommended Practice for Maintenance Testing and Replacement of Vented Lead-Acid Batteries for Stationary Applications], in addition to the required Surveillance Requirements, ensures that corrective maintenance can be performed prior to a condition challenging an operability limit.

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

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

The proposed changes revise the surveillance requirements, and required actions associated with the 125VDC distribution system and the battery cell parameters. The requirements associated with the ANO–1 switchyard DC sources have been relocated to licensee control. The proposed changes do not change the design, configuration, or method of operation of the plant.

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

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

Relocation of the switchyard 125V DC control power source requirements has no impact on any safety analysis assumptions. In addition, the requirements associated with these control power sources are relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of 10 CFR 50.59.

Proposed changes incorporating the requirements of TS 3.7.1.D, 3.7.2.E, 3.7.2.F, and 3.7.2.A, as related to the DC electrical power subsystems, in the new TS 3.7.3 impose more stringent requirements than previously specified for ANO–1.

The proposed addition of TS 3.7.4 allowing continued operation for a limited period of time with battery cell parameters not within limits under certain conditions clarifies an allowance that currently exists in the ANO–1 TS due to the absence of acceptance criteria for the battery cell parameter surveillances.

Proposed changes in Surveillance Requirements and Frequencies reflect current industry guidance on maintenance and testing of the station batteries. Although some frequencies have been extended, continued performance of maintenance activities in accordance with IEEE–450, in addition to the required Surveillance Requirements, ensures that corrective maintenance can be performed prior to a condition challenging an operability limit.

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

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

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 17, 1999.

Description of amendment request: The proposed amendment changes the Perry Nuclear Power Plant as described in the Updated Safety Analysis Report. The change incorporates a leak-off line in the residual heat removal system. The leak-off line is designed to eliminate an operator work around, which will significantly reduce the collective dose to plant operations personnel.

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 modification has been described, and will be procured and installed in accordance with the original design codes and standards. The safety functions of the RHR [residual heat removal] system have not been impacted by the change. Systems supporting the operation of the RHR system have not been affected by this modification. Though the modification affects the Containment System, the containment remains capable of performing its associated safety functions to the same level as the original design.

The accidents of concern are the Loss-Of-Coolant (LOCA) and the Loss of Shutdown Cooling. The proposed change has been designed in accordance with the original codes and standards. The proposed change will not alter the operation of any plant equipment assumed to function in response to the aforementioned analyzed events or otherwise increase their failure probability. Therefore, the probability of occurrence or the consequences of an accident previously evaluated remains unchanged.

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

The proposed modification has been designed, and will be procured and installed in accordance with the original RHR system design codes and standards. RHR system functions have not been impacted by the change. Systems supporting the operation of the RHR system have not been affected. Failure of the modification to perform its design function due to leak-off line failure or blockage would be identical to the current RHR system performance. Improper operation of the valves associated with the modification have been evaluated and will not prevent or otherwise inhibit the RHR or Containment systems from performing their applicable safety functions.

Missile generation is not a concern since no mechanisms conducive to missile generation have been introduced. Electrical analyses have shown there is no adverse effect upon the diesel generator loadings. A single failure of the new configuration will not result in more than the loss of a single RHR loop which is already analyzed. Therefore, the possibility of a new or different kind of accident from any previously evaluated has not been created.

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

The proposed modification has been designed, and will be procured and installed in accordance with the original RHR system design codes and standards. The RHR and Containment systems remain capable of performing their safety functions. Systems supporting the operation of the RHR system have not been affected. Hence, the RHR system margin of safety with respect to safety classification, protection, redundancy, and seismic classification remains unaffected.

The margins of safety contained in the Technical Specifications and the associated Bases also remain unaffected by this modification. Specifically, Technical Specifications 3.4.6, ''Reactor Coolant System Pressure Isolation Valve Leakage'; 3.4.9, ''RHR Shutdown Cooling System—Hot Shutdown'; 3.4.10, ''RHR Shutdown Cooling System—Cold Shutdown'; 3.6.2.1, ''Suppression Pool Average Temperature'; and 3.6.2.2, ''Suppression Pool Water Level'; and the associated Bases remain unchanged and fully applicable. Hence, the margins of safety defined in the Technical Specifications remains unaffected.

Therefore, the proposed modification does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 31, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications to (1) increase the minimum reactor coolant system (RCS) flow rate limit, (2) delete the reactor coolant flow rate footnote, and (3) change the minimum frequency surveillance for RCS flow rate.

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.

Combustion Engineering (ABB/CE) in Thermal-Hydraulic Report CR–94–19–CSE95–1131, Revision 0 performed a comprehensive evaluation of the effects the removal of the orifice plates would have on steam generator tube degradation. It was concluded that the removal of the orifice plates would increase the primary flow rate by approximately 5%.

The removal of the orifice plates was estimated to increase the probability of tubes requiring repair over the lifetime of the plant. However, the presence of the orifice plates had prevented inspection of approximately 22% of the steam generator tubes for circumferential cracks on the hot-leg side. Therefore, it was concluded that the removal of the orifice plates did not increase the probability of steam generator tube failure, given that the tubes previously covered by the plates are now inspected each outage in accordance with the Electrical Power Research Institute Pressurized Water Reactor (EPRI PWR) steam generator examination guidelines. Fort Calhoun Station is using the eddy current inspection technology to ensure that tubes showing evidence of a crack exceeding the present plugging criteria will be repaired or removed from service. Industry experience has shown that even in cases of severely degraded tubes, the resulting primary to secondary leak rates are insignificant compared to those analyzed in the design basis steam generator tube rupture event.

Calculation of the Reactor Coolant Flow Rate using the heat balance methodology once every refueling outage is consistent with requirements contained in the NUREG 1432, Improved Technical Specifications for Combustion Engineering Plants' surveillance requirement 3.4.1.4.

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

The original orifice plates were installed on each steam generator hot leg tube sheet in the primary inlet plenum as a field modification prior to the initial fuel load in the year 1973. The orifice plates were designed to increase the hydraulic resistance of the primary coolant flow rate in the associated tubes, thereby reducing the primary coolant temperature inside the tubes. Reduction of the primary coolant temperature and flow rate would decrease the heat flux, thus improving the steam quality and reducing the potential for dry-out and surface deposits on the outer surface of the tubes. However, due to inaccessibility, these originally installed orifice plates had prevented tube inspection in the hot leg tube sheet area, even with the latest state-of-the-art eddy current probe technology. The orifice plates also prevented normal repair techniques such as steam generator tube plugging and sleeving.

The original orifice plates were removed during the 1996 refueling outage. However, there were concerns related to Westinghouse fuel failures as a result of flow-induced vibration. To address those concerns, new ''removable'' orifice plates were installed to maintain the RCS flow rate at the previous level. Since then, the remaining batches of the Westinghouse fuel considered most susceptible to flow-induced vibration were replaced during the 1998 refueling outage, thus minimizing the concerns and allowing the permanent removal of the ''removable'' orifice plates.

The removal of the ''removable'' orifice plates returned the steam generators to their original design configuration. RCS flow rate has increased by virtue of decreased hydraulic resistance through the steam generators. No other systems or components other than the steam generators have been affected. The resulting change in operational parameters (decreased reactor coolant T_{hot} temperature and increased flow rate) has been evaluated for the Updated Safety Analysis Report Chapter 14. Potential adverse consequences of the modifications were (1) increase in reactor vessel component vibration, (2) increase in hydraulic loading, and (3) increase in steam generator tube degradation for row 1–18 tubes. The potential adverse consequences were evaluated and found to be acceptable.

Calculation of the Reactor Coolant Flow Rate using the heat balance methodology once every refueling outage is consistent with requirements contained in the NUREG 1432, Improved Technical Specifications for Combustion Engineering Plants' surveillance requirement 3.4.1.4.

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

The removal of the orifice plates has resulted in approximately a 5% increase in the reactor coolant flow rate. This has increased the margin for minimum reactor coolant system flow rate specified in Technical Specifications Section 2.10.4, Power Distribution Limits, Item (5), DNBR Margin During Power Operation Above 15% of Rated Power. Steam Generator tube inspections performed in accordance with Technical Specifications Section 3.17, Steam Generator Tubes, have not been adversely affected.

The increased flow rate has been analyzed for the thermal hydraulic effects on the reactor core and was found acceptable.

Calculation of the Reactor Coolant Flow Rate using the heat balance methodology once every refueling outage is consistent with requirements contained in the NUREG 1432 [Improved Technical Specifications for Combustion Engineering Plants] surveillance requirement 3.4.1.4.

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: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: January 28, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to remove two lists of Containment Isolation Valves (CIVs) in Tables 3.6–1 and 4.4–1 and make related changes to TSs 1.10, 3.6.A.1, and 4.4 and the associated bases.

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

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

Response: No. Operation of Indian Point 3 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The removal of the two component listings (i.e., Tables 3.6–1 and 4.4–1) and the TS references to them from the TS requested by this submittal is performed in accordance with the guidance provided by the NRC in GL 91–08 [Generic Letter 91–08]. As established by the NRC, in the aforementioned GL, such a change will not alter existing TS requirements or those components to which they apply. Required information contained in the two tables being removed is duplicated in the FSAR [final safety analysis report] and other appropriate plant procedures. Any subsequent changes regarding the individual components (i.e., the containment isolation valves) or their operation (e.g., valve positioning under administrative controls) would be addressed in accordance with the requirements specified in the Administrative Controls section of the TS regarding changes to plant procedures and/or changes to the FSAR (i.e., 10 CFR 50.59). These changes will not alter any structure, system, or component and, therefore, will not result in the possibility of an increase in [the] probability or consequence of an accident previously evaluated.

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

Response: No. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The deletion of two component listings (i.e., Tables 3.6–1 and 4.4–1) and the TS references to them from the Technical Specifications and the removal of all references made in the TS regarding these two listings will not alter how the individual components (i.e.—the containment isolation valves) identified in the tables are designed, operated, tested, or maintained. Testing of CIVs will be performed as required by 10 CFR part 50, Appendix J and IP3 TS 6.14.

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

Response: No. The proposed license amendment does not involve a significant reduction in a margin of safety. The proposed changes are in accordance with recommendations provided by NRC in Generic Letter 91–08 and the Standard Technical Specifications, NUREG 1431. These changes will maintain current safety margins while reducing the regulatory/administrative burdens to both the NRC and to the Power Authority. As stated, the changes will not result in changes to the design, operation, or maintenance of the ClVs, and the testing of the CIVs will be in accordance with 10 CFR 50 Appendix J and IP3 TS 6.14.

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

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

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

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: April 12, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to remove the footnote restriction found on page 3.1–36 which states that the departure from nucleate boiling (DNB) analysis contains adequate margin for Cycle 10, but needs to be reviewed/approved prior to Cycle 11.

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

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

Response: The proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed. The removal of the footnote on TS page 3.1–36 is an administrative change in that it does not affect the DNB limits of the current TS. The footnote was added to the TS as part of Amendment 175, which permitted the use of V+ fuel at IP3. The footnote required the Authority to demonstrate that sufficient DNB margin existed for Cycle 11, prior to achieving criticality for that cycle. The NRC requested this DNB limitation because the applicability of the WRB–1 correlation to predict DNB performance for the V+ fuel had not been adequately proven by fuel tests. Westinghouse has completed fuel tests which verify that the use of the WRB–1 correlation with the 15 × 15 V+ fuel is conservative. Therefore, this DNB limitation is no longer applicable and the footnote can be removed.

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

Response: The proposed change does not create the possibility of a new or different kind of accident, as the removal of the footnote on TS page 3.1–36 does not affect the current TS DNB limits, plant equipment, or the way the plant is operated. This footnote was inserted into the TS as part of Amendment 175, which permitted the use of 15 × 15 V+ fuel at IP3. Westinghouse had used scaling techniques to demonstrate that the WRB–1 correlation correctly predicted the critical heat flux performance of the 15 × 15 V+ fuel. Since no fuel tests had been performed on this fuel design, the NRC was concerned that the use of this correlation may be unconservative. Therefore, approval to use the V+ fuel at IP3 was granted based upon the DNB margin available during Cycle 10. This limitation was contained in the footnote on TS page 3.1–36. Westinghouse has recently completed fuel tests on 15 × 15 V+ fuel which verify that the use of the WRB–1 correlation is conservative. Therefore, the use of V+ fuel at IP3 is no longer dependent on the amount of DNB margin available and the footnote can be removed.

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

Response: The proposed deletion of the footnote on TS page 3.1–36 does not involve a significant reduction in a margin of safety. The footnote was introduced as part of Amendment 175, which permitted the use of V+ fuel at IP3. The footnote required the Authority to demonstrate that sufficient DNB margin existed for Cycle 11, prior to achieving criticality for that cycle. The NRC requested this DNB limitation because the applicability of the WRB–1 correlation to predict DNB performance for the V+ fuel had not been adequately proven by fuel tests. Westinghouse has completed fuel tests which verify that the use of the WRB–1 correlation with the 15 × 15 V+ fuel is conservative. Therefore, this DNB limitation is no longer applicable and the footnote can be removed. The removal of the footnote is an administrative change as deleting it does not alter the current DNB margin or future DNB margins.

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

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

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

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: March 29, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) by relocating the procedural details of the Radiological Effluent Technical Specifications (RETS) to the Offsite Dose Calculation Manual (ODCM). The TSs would also be revised to relocate procedural details associated with solid radioactive wastes to the Process Control Program (PCP). In addition, the Administrative Controls section of the TSs would be revised to incorporate programmatic controls for radioactive effluents and environmental monitoring. The proposed changes are consistent with the guidance provided in Generic Letter 89–01, ''Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program.''

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

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

The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents.

Implementation of programmatic controls for RETS in TS will assure that the applicable regulatory requirements pertaining to the control of radioactive effluents will continue to be maintained. Since there are no changes to previous accident analysis, the radiological consequences associated with these analyses remain unchanged, therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes have no impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There is no impact on equipment design or operation and there are no changes being made to the TS required safety limits or safety system settings that would adversely affect plant safety as a result of the proposed changes. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents. A comparable level of administrative control will continue to be applied to those design conditions and associated surveillances being relocated to the ODCM or PCP. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: April 29, 1999 (TS 99–04).

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) for Sequoyah (SQN) Units 1 and 2 by deleting the Auxiliary Feedwater (AFW) suction pressure low channel functional surveillance test. The licensee's analysis of the performance history revealed that the monthly functional test of this instrument channel does not provide an increased assurance of operability that justifies the monthly 7 hours per unit system unavailability that it creates.

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:

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

The probability of occurrence or the consequences for an accident is not increased by this request. The proposal to delete the monthly channel functional test for the auxiliary feedwater (AFW) suction pressure low functions does not alter the way any structure, system or component functions, does not modify the manner in which the plant is operated, and reduces equipment out-of-service time. This request does not degrade the ability of AFW to perform its intended function. Therefore, the pressure switches will be available to perform their intended function.

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

A possibility for an accident or malfunction of a different type than any evaluated previously in SQN's FSAR [Final Safety Analysis Report] is not created. The proposal does not alter the way any structure, system or component functions and does not modify the manner in which the plant is operated. Therefore, the possibility of a new or different kind of accident previously evaluated is not created by the proposed change to delete the monthly functional test of the AFW pressure switches.

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

The margin of safety has not been reduced since the test methodologies are not being changed. Increasing the surveillance interval does not change the results of accident analysis by this request. The proposed change to delete the AFW system pressure low channel functional test does not involve a significant reduction in the margin of safety. The new frequency will not reduce the reliability of the system and increases overall system availability. Therefore, changing the frequency of the surveillance does not reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 3740.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H Knoxville, Tennessee 37902.

NRC Section Chief: Sheri Peterson.

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

Date of amendment request: April 29, 1999 (TS 99–03).

Description of amendment request: The proposed amendment would add new actions to Technical Specification (TS) Limiting Condition for Operations (LCOs) 3.3.3.1 and 3.7.7 to address the situation when one channel of radiation monitoring control room emergency ventilation system actuation equipment is inoperable and would expand the mode of applicability for LCOs 3.3.3.1 and 3.7.7 to include periods when movement of irradiated fuel assemblies are involved and defines actions to take in these instances.

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:

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

The proposed revision does not change any plant functions or equipment operating practices for the radiation monitoring system and control room emergency ventilation system (CREVS). The radiation monitoring instruments and the CREVS are not considered to be the source of any accident evaluated in the Final Safety Analysis Report. These features provide accident mitigation functions that will be utilized in response to postulated accident conditions. The activities and failures that could contribute to the initiation of an accident are not affected by the implementation of this revision. This revision provides for more stringent requirements for operation of the facility (additional limiting condition for operation [LCO] actions and applicability requirements). Therefore the proposed activity will not increase the probability of an accident.

The proposed activity does not affect accident mitigation capabilities or the radiation release amounts for postulated accidents. This TS change will not affect requirements that the radiation monitoring system and CREVS be maintained to support accident mitigation. The functions and testing will remain the same while operability requirements will become more stringent. This TS change enhances the requirements associated with CREVS and the initiation of this system such that inoperabilities are appropriately handled to reduce the safety impact of component inoperabilities. Therefore, the proposed change will not increase the consequences of an accident and could reduce the consequences by limiting operation with inoperable components and requiring the application of appropriate actions for all conditions that could result in a postulated accident that CREVS was designed to mitigate.

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

The proposed change provides more stringent operating requirements for operation of the facility. The proposed activity will not change any plant function or operating practice that could impact accident initiators. Therefore, these more stringent requirements do not result in operation that will increase the probability of any postulated accidents. In addition, CREVS and the associated actuation features are not considered to be the source of an accident. Therefore, the proposed activity will not create the possibility of an accident of a different kind.

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

The proposed activity does not impact plant setpoints designed to maintain the assumptions in the safety analysis or limits for the actuation of systems to mitigate accidents. Plant functions and operating practices will not be altered by the implementation of more stringent requirements for operation of the facility. These requirements, by definition, provide additional restrictions to enhance plant safety. Therefore, the proposed activity will not reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H Knoxville, Tennessee 37902.

NRC Section Chief: Sheri Peterson.

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

Date of amendment request: February 1, 1999, as supplemented on April 19 and April 23, 1999.

Description of amendment request: The amendment request proposes a total replacement of current Technical Specifications Section 6, ''Administrative Controls.'' Administrative changes to certain other sections of Technical Specifications are also being made to conform to the changes resulting from the re-write of Section 6.

The proposed changes represent a comprehensive upgrade of Section 6 of the Vermont Yankee Technical Specifications, incorporating improvements in content and format based on industry standards. In accordance with industry practice some Technical Specifications requirements are being relocated to the recently implemented Vermont Yankee Technical Requirements Manual (TRM), Offsite Dose Calculation Manual (ODCM), or Vermont Yankee Operational Quality Assurance Manual (VOQAM) and will be eliminated from the Technical Specification upon NRC approval.

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

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

The proposed changes have no effect on plant hardware, plant design, safety limit setting, or plant system operation and therefore do not modify or add any initiating parameters that would significantly increase the probability or consequences of an accident previously evaluated.

No new modes of operation are introduced by the proposed changes such that additional adverse consequences would result. Accordingly, the consequences of previously analyzed accidents are not deleteriously affected by this proposed license amendment.

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

The proposed changes do not involve any physical alteration of the plant (no new or different type of equipment will be installed) or any change in the methods governing normal plant operation. These changes do not affect the operation of any systems or components, nor do they involve any potential initiating events that would create any new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated for VYNPS.

3. Involve a significant reduction in a margin of safety, because:

The proposed changes have no impact on any safety analysis assumptions. Consequently, no margin of safety as described in the Final Safety Analysis Report and defined in the basis of any Technical Specification is reduced as a result of these changes.

These proposed changes do not detrimentally affect the ability of structures, systems and components important to safety to fulfill their intended safety functions. Therefore, it is concluded that the proposed changes do no[t] involve a significant reduction in a margin of safety.

Additional Safety Considerations for Specific Changes Deemed to be ''Less Restrictive''

In accordance with the criteria set forth in 10 CFR 50.92, Vermont Yankee has evaluated the proposed changes to the [Vermont Yankee Nuclear Power Station] VYNPS Technical Specifications and determined that they do not involve a significant hazards consideration. Those changes which are deemed to be ''less restrictive'' have been subject to the following additional consideration:

(a) Changes which are deemed to be ''less restrictive'' based solely upon removal from the Technical Specifications and relocated in VYNPC-controlled documents:

NRC's Technical Specifications Branch has conducted reviews of the Administrative Controls section of Standard Technical Specifications and concluded that certain provisions historically contained in Technical Specifications can be relocated to other licensee documents for which changes to those provisions are adequately controlled by other regulatory requirements. In general, Administrative Controls are those requirements not covered by other Technical Specifications, but are considered necessary to assure operation of the facility in a safe manner. Application of this criterion can be based on two categories or requirements: (a) requirements not covered by other regulatory requirements, but are considered necessary to assure the safe operation of the facility or (b) specific requirements that are broadly covered by regulations or other regulatory controls, for which details need to be specified in the Technical Specifications to ensure safe plant operation. In general, however, Technical Specifications need not duplicate other regulatory requirements.

As identified in Attachment A hereto, certain portions of the current Technical Specifications are to be relocated to the Technical Requirements Manual (TRM), Offsite Dose Calculation Manual (ODCM), or the Vermont Yankee Operational Quality Assurance Manual (VOQAM) and removed from the Technical Specifications. As an initial step in this process, the subject requirements are being duplicated in the TRM, ODCM, or VOQAM. Removal from the Technical Specifications will occur upon NRC approval. The ability to relocate these requirements is based on regulations and standards that contain these provisions such that duplication in the Technical Specifications is not necessary.

[1. Involve a significant increase in the probability or consequences of an accident previously evaluated, because:]

The TRM is a[n] FSAR level document and is incorporated by reference into the FSAR. Changes to the TRM will be strictly controlled by the 10 CFR 50.59 process to ensure that proper reviews are conducted. The relocation of requirements to the VYNPC-controlled TRM will not diminish the effectiveness of compliance withthe relocated provisions. Since any changes to the TRM will be evaluated per the requirements of 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously analyzed will be allowed. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Changes to the ODCM are controlled by current Technical Specifications and require the reporting to the NRC of changes to the

ODCM with sufficient information to support the changes together with appropriate analyses or evaluations justifying the changes. The relocation of these details to the ODCM is thus acceptable considering the controls provided by existing regulations and the controls remaining in Technical Specifications for ODCM changes. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Relocation of the Technical Specification Administrative Controls related to quality assurance from the Technical Specifications to the VOQAM is consistent with the guidance provided by the NRC in Administrative Letter 95–06, ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance.'' Changes to the VOQAM are subject to the change control process in 10CFR50.54(a). These provisions are adequate to ensure that quality assurance program commitments are not reduced without prior NRC approval. Therefore, these changes do 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 changes do not involve any 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 information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

[3. Involve a significant reduction in a margin of safety, because:]

The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumption. In addition, the details to be transposed from the Technical Specifications to the TRM, ODCM, and VOQAM are the same as the existing Technical Specifications. Since any future changes to these provisions in the TRM will be evaluated per the requirements of 10CFR50.59 and Technical Specifications already requires supporting information be submitted to the NRC for ODCM changes, no reduction (significant or insignificant) in a margin of safety will be allowed. The provisions of 10CFR50.54(a) are adequate to control changes to the VOQAM and maintain current margins of safety.

Based on 10CFR50.92, the existing requirement for NRC review and approval of revisions (to the Technical Specifications provisions proposed for relocation) does not have a specific margin of safety upon which to evaluate. However, since the proposed changes are consistent with industry standards, approved by the NRC, revising the Technical Specifications to relocate these provisions will not diminish administrative controls necessary to assure the safe operation of the facility.

(b) Change [9] identified in Attachments A and D [of the February 1, 1999, submittal]:

This change proposes to relax the requirement to have an individual qualified in radiation protection procedures onsite at all times. The proposed change will allow the position to be vacant for up to two hours in order to provide for unexpected absence.

[1. Involve a significant increase in the probability or consequences of an accident previously evaluated, because:]

The proposed change does not affect the probability of an accident. The actions of an individual qualified in radiation protection procedures are not assumed to be an initiator of an accident. Also, the consequences of an accident are not affected by the presence of an individual qualified in radiation protection procedures. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event. This change will not have any impact on the safe operation of the plant because the presence of a person qualified in radiation protection procedures is not required for the mitigation of any accident. Therefore, this change will 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:]

This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different type of accident from any accident previously evaluated.

[3. Involve a significant reduction in a margin of safety, because:]

The margin of safety is not affected by the presence or absence onsite of an individual qualified in radiation protection procedures. This proposed change has no effect on the assumptions of any design basis accident. This change has no impact on the safe operation of the plant since the presence onsite of an individual qualified in radiation protection procedures is not required for the mitigation of an accident. This change does not affect any plant equipment or requirements for maintaining plant equipment. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

(c) Change [10] identified in Attachments A and D [of the February 1, 1999, submittal]:

This change proposes to incorporate the allowances of a temporary deviation from the shift staffing levels of 10CFR50.54(m)(2)(i) for up to two hours. In addition, this change proposes to apply these same allowances to the positions of Shift Engineer and non-licensed operators.

[1. Involve a significant increase in the probability or consequences of an accident previously evaluated, because:]

The proposed change does not affect the probability of an accident. The shift staffing level requirements are not assumed to be an initiator or any analyzed event. Also, the consequences of an accident are not affected by these temporary deviations to the shift staffing levels. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event, since 10CFR50.54(m) (ii) and (iii) still maintain the requirements for the presence of licensed operators and senior operators. This change has no impact on the safe operation of the plant. The level of shift staffing will still be maintained as required by 10CFR50.54(m) (ii) and (iii) and does not affect any plant equipment or requirements for maintaining plant equipment. The temporary deviations from the shift staffing level for up to two hours to provide for unexpected absence, provided immediate action is taken to fill the required position is acceptable in terms of staffing requirements for the mitigation of an accident due to the low probability of an accident occurring during these short-term, infrequent deviations and the remaining licensed operators and senior operators. Therefore, this change will 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:]

This change will not physically alter the plant (no new or different type of equipment will be installed). The temporary deviations from shift staffing levels are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different type of accident from any accident previously evaluated.

[3. Involve a significant reduction in a margin of safety, because:]

The margin of safety in not reduced by allowing these temporary deviations from shift staffing levels due to unforeseen events. This proposed change has no effect on the assumptions of any design basis accident. This change has no impact on the safe operation of the plant since 10CFR50.54(m) (ii) and (iii) still maintain the requirements for the minimum number of licensed operators and senior operators necessary to safely operate the plant. This change does not affect any plant equipment or requirements for maintaining plant equipment. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

(d) Changes [38] and [39] identified in Attachments A and D [of the February 1, 1999, submittal]:

In accordance with 10CFR20.1601 (c), these changes propose alternative methods for controlling access to high radiation areas consistent with the intent of 10CFR20.1601 (a) and (b).

[1. Involve a significant increase in the probability or consequences of an accident previously evaluated, because:]

The proposed changes do not affect the probability of an accident. The controls used for access to high radiation areas are not assumed in the initiation of any analyzed event. Also, the consequences of an accident are not affected by these changes. These changes are both consistent with good radiological practices and will provide an adequate level of radiation protection. These proposed changes do not impact the assumptions of any design basis accident. These changes will not alter assumptions relative to the mitigation of an accident or transient event. These changes have no impact on safe operation of the plant. Therefore, these changes will 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 changes will not create the possibility of an accident. These changes will not physically alter the plant (no new or different type of equipment or system will be installed). The changes in methods governing normal plant operations are consistent with the current safety analysis assumptions and deal only with personnel exposure to radiation, not reactor safety. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

[3. Involve a significant reduction in a margin of safety, because:]

The margin of safety is not reduced due to these proposed changes. These changes are both consistent with good radiological safety practice and have been found to provide adequate levels of radiation protection. In addition, these changes provide the benefit of ensuring radiation dose to workers can be minimized by providing the flexibility to select the best means of providing access control to a high radiation area, given the plant area and radiological conditions. These proposed changes have no impact on the safe operation of the plant. No change in analytic limits or setpoints is introduced by these changes. The safety analysis assumptions will still be maintained, thus no question of nuclear safety exits. Therefore, these changes do not involve a significant reduction in a margin of safety.

(e) Change [49] identified in Attachments A and D [of the February 1, 1999, submittal]:

This change proposes to relax the requirement for submitting the (now-named) Occupational Radiation Exposure Report from the currently required date of March 1 to April 30 of each year. April 30 is now the industry standard date for submittal of such reports.

[1. Involve a significant increase in the probability or consequences of an accident previously evaluated, because:]

The proposed change does not affect the probability of an accident. The submittal date of the Occupational Radiation Exposure Report is not assumed to be an initiator of any analyzed event. Also, the consequences of an accident are not affected by the submittal date of this report. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event. This change has no impact on the safe operation of the plant. The report will still be required to be submitted each year and does not affect any plant equipment or requirements for maintaining plant equipment. The submittal date of this report is not required for the mitigation of any accident. Therefore, this change will 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 will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The change in method governing submittal of this report does not affect current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different type of accident from any accident previously evaluated.

[3. Involve a significant reduction in a margin of safety, because:]

The margin of safety i[s] not reduced by allowing the report to be submitted 60 days later. This proposed change has no effect on the assumptions of the design basis accident. This change has no impact on the safe operation of the plant. The report will still be required to be submitted each year and does not affect any plant equipment or requirements for maintaining plant equipment. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

(f) [Change [64] identified in Attachments A and D [of the February 1, 1999, submittal]:

This change proposes to relax the requirement for submitting the (now-named) Annual Radiological Environmental Operating Report from the currently required date of May 1 to May 15 of each year. May 15 is now the industry standard date for submittal of such reports.

[1. Involve a significant increase in the probability or consequences of an accident previously evaluated, because:]

The proposed change does not affect the probability of an accident. The submittal date of this report is not assumed to be an initiator of any analyzed event. Also, the consequences of an accident are not affected by the submittal date of this report. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event. This change has no impact on the safe operation of the plant. The report will still be required to be submitted each year and does not affect any plant equipment or requirements for maintaining plant equipment. The submittal date of this report is not required for the mitigation of any accident. Therefore, this change will 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 will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The change in method governing submittal of this report does not affect current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different type of accident from any accident previously evaluated.

[3. Involve a significant reduction in a margin of safety, because:]

The margin of safety i[s] not reduced by allowing the report to be submitted 14 days later. This proposed change has no effect on the assumptions of the design basis accident. This change has no impact on the safe operation of the plant. The report will still be required to be submitted each year and does not affect any plant equipment or requirements for maintaining plant equipment. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: April 20, 1999.

Description of amendment request: The amendment request proposes changes to the existing requirements associated with the unloading and loading of fuel in the reactor vessel.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

VY has determined that the proposed change to reload the reactor core in a spiral pattern beginning around a Source Range Monitor (SRM) does not involve a significant increase in the probability or consequences of an accident previously evaluated. The design basis accident associated with refueling is the Refueling Accident; i.e., the accidental dropping of a fuel bundle onto the top of the core. There is no assumption as to the core loading pattern in the analysis of this accident. The analyzed abnormal operational transients associated with refueling are: (1) the Control Rod Removal Error During Refueling, and (2) the Fuel Assembly Insertion Error During Refueling. There is no assumption as to the core loading pattern in the analyses of these transients. The Fuel Assembly Insertion Error During Refueling transient involves mislocated and rotated fuel assembly loading errors. However, a change in the approved core loading pattern has no impact on the probability of mislocating or rotating a bundle while following that pattern. Furthermore, the proposed change implements a core loading pattern that provides improved flux monitoring as compared to the pattern prescribed by the current Technical Specifications. When loading the core in accordance with the proposed change, the SRM indication will be indicative of the true flux of the loaded fuel, as the creation of flux traps (moderator filled cavities surrounded on all sides by fuel) is precluded.

The Technical Specification Bases are under the purview of 10CFR50.59. As such, subsequent changes made via 10CFR50.59 to the information relocated to the Bases are not allowed to increase the probability or consequences of an accident previously evaluated. Therefore, relocating the details of the core loading pattern to the Bases does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The SRMs and the core loading pattern are not initiators of any accident previously evaluated. As such, the subject changes cannot affect the probability of an accident previously evaluated. The core loading pattern is not assumed in the mitigation of any accident. Since the proposed change provides improved flux monitoring by the SRMs, operators will have more accurate indication and SRM automatic trip functions will actuate more accurately. As such, any event mitigation function provided by the SRMs is enhanced by this change. Therefore, the associated changes do not involve a significant increase in the consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

VY has determined that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. VY proposes to change the core reloading and offloading patterns to start and stop, respectively, at an SRM versus the geometric center of the core as prescribed by current Technical Specifications. This ensures that flux monitoring instrumentation is always OPERABLE in the fueled region of the vessel. There is no separation of the monitoring device from the fuel by cavities of water as is the case with the pattern prescribed by the current Technical Specifications. As such, flux monitoring is enhanced during core reloading and offloading. This change is conservative relative to the current requirements. Therefore, no new categories or types of accidents are created.

Additionally, the Technical Specification Bases are under the purview of 10CFR50.59. As such, subsequent changes made via 10CFR50.59 to the information relocated to the Bases are not allowed to create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report. Therefore, relocating the details of the core loading pattern to the Bases does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

VY has determined that the proposed change does not involve a significant reduction in a margin of safety. Loading around the geometric center of the core as prescribed by the current Technical Specifications results in cells of moderator separating the fuel from the instrumentation monitoring its flux. This change requires the flux monitoring instrumentation to be in the fueled region, and, in so doing, provides for more accurate monitoring of core flux during core reloading and offloading. As such, the operators will have more accurate indication and SRM automatic trip functions will actuate when the actual flux reaches the trip setpoints. This corrects non-conservatisms that result from cells of moderator separating the fuel from the instrumentation. Therefore, this change will not result in a significant reduction in a margin of safety.

Additionally, the details of the loading pattern are relocated from the Technical Specifications to the Bases. Since any future changes to the Bases will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be allowed. Therefore, relocating the core loading pattern details to the Bases does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of amendment request: April 7, 1999.

Description of amendment request: The proposed amendment would revise the minimum critical power ratio (MCPR) limit in Technical Specification (TS) 2.1.1.2, for the ATRIUM–9X and the SVEA–96 fuel for one and two recirculation loop operation. The proposed amendment would add a new reference in TS 5.6.5, ''Core Operating Limits Report.'' The reference cites ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation-Nuclear Division, July 1998.

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 probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed Technical Specifications amendment uses conservatively established SLMCPR [safety limit minimum critical power ratio] values for WNP–2 such that the fuel is protected during normal operation as well as during plant transients or anticipated operational occurrences.

The probability of an evaluated accident is not increased by the use of the ATRIUM–9X MCPR safety limit of 1.10 (two loop operation) or 1.11 (single loop operation). The ATRIUM–9X fuel was evaluated by SPC (Reference 5) [Letter KVW:98:148 dated July 8, 1998, KV Walters, (Siemens Power Corporation), to RA Vopalensky (Supply System), ''MCPR Safety Limit Reanalysis for WNP–2 Cycle 11''] using the additive constant uncertainty for ATRIUM–9X fuel of 0.0201 which is contained in the NRC safety evaluation approval of Reference 4 [ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF–1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation—Nuclear Division, July 1998]. Based upon the NRC approved additive constant of uncertainty of 0.0201, as documented in Reference 5, at least 99.9% of the SPC ATRIUM–9X fuel rods would be expected to avoid boiling transition with a SLMCPR of 1.10 during two loop operation and 1.11 during single loop operation.

The probability of an evaluated accident is not increased by the use of the ABB SVEA–96 SLMCPRs of 1.10 (two loop operation) or 1.12 (single loop operation). NRC approved methodology documented in CENPD–300–P–A, ''Reference Safety Report for Boiling Water Reactor Reload Fuel'', July 1996 (Reference 3) was used in deriving these ABB SVEA–96 SLMCPR values. The ABB evaluation as a function of cycle exposure established that late in Cycle 15 conservative two loop and single loop SLMCPRs of 1.10 and 1.12, respectively, can be used to represent the entire cycle.

The SLMCPR changes do not require any physical plant modifications, physically affect any plant component, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

Since the operability of plant systems designed to mitigate any consequences of accidents have not changed, the consequences of an accident previously evaluated are not expected to increase.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications of the plant configuration or allowable modes of operation. This Technical Specification change establishes SLMCPRs for SPC fuel based upon the NRC approved additive constant of uncertainty of 0.0201, as documented in Reference 5. At least 99.9% of the SPC ATRIUM–9X fuel rods would be expected to avoid boiling transition with an SLMCPR of 1.10 during two loop operation or 1.11 during single loop operation. Additionally, the ABB SVEA–96 SLMCPRs of 1.10 (two loop operation) or 1.12 (single loop operation) were derived using the NRC approved methodology documented in CENPD–300–P–P, ''Reference Safety Report for Boiling Water Reactor Reload Fuel'', July 1996 (Reference 3). Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

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

Implementation of SLMCPRs derived by proven analytical methods provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated. Because the fuel design safety criteria of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as anticipated operational occurrences is met, there is not 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: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Attorney for licensee: Perry D. Robinson, Esq., Winston & Strawn, 1400 L Street, NW, Washington, DC 20005–3502.

NRC Project Director: Stuart Richards.

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of amendment request: April 20, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.4.11, ''RCS Pressure and Temperature Limits,'' to update the curves that set forth the pressure temperature limit lines. The curves provide the pressure temperature limits for the operation of the reactor coolant system for heatup and cooldown during inservice leak and hydrostatic testing, non-nuclear heating and cooldown, and nuclear heating and cooldown.

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 pressure temperature shift is well within the operating margins of plant equipment. Using the new non-nuclear and nuclear heating and cooldown curves, higher temperature values for corresponding pressures at temperatures which are closest to RT_{NDT}, further reduce the potential for brittle fracture.

The proposed 32 EFPY [effective full power years] curves were developed using methodology that is consistent with the guidance in Regulatory Guide 1.99, Revision 2, Appendix G of the ASME Code and Appendix G of 10 CFR part 50. This methodology is recognized by the NRC and the industry as providing acceptable margin.

Therefore, operation of WNP–2 in accordance with the proposed amendment 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 accident previously evaluated.

The proposed change has no impact on the previously analyzed accidents or transients. The proposed change does not introduce any credible mechanisms for unacceptable radiation release nor does it require physical modification to the plant. The 32 EFPY curves are calculated using a published methodology that was discussed with the NRC.

The proposed change is also within any upper bound limit. The only impact on plant operation is that the plant will be operated with new pressure temperature limits derived from the proposed alternative calculational methodology in place of the previously approved model based on actual plant data.

Therefore, the operation of WNP–2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The results of testing reflected 30 ft-lb shifts and changes in uppershelf energy of the base plate and the weld material. However, the results are well within the values predicted by Regulatory Guide 1.99, Revision 2. Furthermore, the adjusted reference temperature values and the upper shelf energy of the reactor beltline materials are expected to remain within the limits of 10 CFR part 50, Appendix G, for at least 32 effective full power years of reactor operation.

For the non-nuclear and nuclear heating and cooldown curves (with a calculated through wall ΔT), lower temperatures which are closest to RT_{NDT}, have an increased margin of safety due to the higher required temperature values for a given pressure than is required by current curve calculation methodology. Thus additional margin to brittle fracture is achieved for non-nuclear and nuclear heating.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Attorney for licensee: Perry D. Robinson, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: Stuart Richards.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of amendment request: December 31, 1997, as supplemented May 15, September 15, November 25, 1998 and January 28, 1999.

Description of amendment request: Revise the St. Lucie, Unit 2, Technical Specifications to increase the capacity of the spent fuel storage pool, in part, by allowing a credit for a certain soluble boron concentration in the spent fuel pool.

Date of publication of individual notice in the **Federal Register**: April 5, 1999 (64 FR 16502).

Expiration date of individual notice: May 5, 1999.

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

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

Date of application for amendments: April 19, 1999.

Brief description of amendments: The amendments would revise Technical Specification Section 3/4.8.1.2, ''Electrical Power Systems, Shutdown,'' and its associated bases to provide a one-time extension of the 18-month surveillance interval for specific surveillance requirements for Units 1 and 2. This surveillance will be performed prior to the first entry into Mode 4 subsequent to receipt of the requested T/S amendment. In addition, for Unit 2 only, a minor administrative change is included to delete a reference to T/S 4.0.8, which is no longer applicable. For Unit 1 only, an editorial change is made to add the word ''or'' to action statement 3.8.1.2.

Date of publication of individual notice in **Federal Register**: April 29, 1999 (64 FR 23129).

Expiration date of individual notice: June 1, 1999.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

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

Date of amendment request: April 6, 1999.

Description of amendment request: The proposed amendments would allow an increase of 168 fuel assemblies in the storage capacity of Unit 1's Spent Fuel Pool and an increase of 88 fuel assemblies in the storage capacity of Unit 2's Spent Fuel Pool.

Date of publication of individual notice in **Federal Register**: May 4, 1999 (64 FR 23877).

Expiration date of individual notice: June 3, 1999.

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

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.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: December 29, 1998.

Brief description of amendments: The amendments change Technical Specification Tables 3.3.1–1 and 3.3.2–1 to revise the Allowable Values for 12 functions of the Reactor Trip System and Engineered Safety Features Actuation System.

Date of issuance: April 23, 1999.

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

Amendment Nos.: 107, 107, 100 and 100.

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

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9186). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 23, 1999.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN50–455, 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: October 30, 1998.

Brief description of amendments: The amendments revised the Technical Specification (TS) requirements for spent fuel pool inadvertent draindown elevation.

Date of issuance: May 3, 1999.

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

Amendment Nos.: 108, 108 101, and 101.

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

Date of initial notice in **Federal Register**: December 16, 1998 (63 FR 69335). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 3, 1999.

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 0481.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of application of amendment: June 2, 1998, and as supplemented by letters dated January 18 and March 9, 1999.

Brief description of amendment: The amendment relocates requirements related to seismic monitoring instrumentation from the Technical Specifications to the Technical Requirements Manual.

Date of issuance: April 28, 1999.

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

Amendment No.: 194.

Facility Operating License No. DPR–61: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 23, 1998 (63 FR 50936). The January 18 and March 9, 1999, supplements contained revised TS pages to account for TS changes issued by the NRC since the original June 2, 1998, submittal, pages from the Updated Final Safety Analysis Report and TRM, which were revised to support the June 2, 1998, request, and additional clarifications. The supplemental information did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the original notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 28, 1999.

No significant hazards consideration received: No

Local Public Document Room location: Russell Library, 123 Broad Street, Middletown, Connecticut 06457.

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

Date of application for amendment: August 21, 1996, as supplemented May 2, 1997.

Brief description of amendment: The amendment revised Section 3.3.G (Hydrogen Recombiner System and Post-Accident Containment Venting System), the basis for Section 3.3.G, and Section 4.4, Table 4.4–1 (Containment Isolation Valves). This change permits removal of the existing flame-type hydrogen recombiners, its supporting equipment, and replacement with passive autocatalytic recombiners.

Date of issuance: April 27, 1999.

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

Amendment No.: 200.

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

Date of initial notice in **Federal Register**: Janaury 29, 1997 (62 FR 4345). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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

Date of application for amendment: September 3, 1997.

Brief description of amendment: The amendment revises TS 3.14, Control Room Ventilation, to be consistent with NUREG–1432, Standard Technical Specifications, Combustion Engineering Plants.

Date of issuance: May 6, 1999.

Effective date: May 6, 1999.

Amendment No.: 186.

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14281). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 6, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423–3698.

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

Date of application of amendments: March 1, 1999.

Brief description of amendments: The amendments revised the Technical Specifications by adding a Note to Improved Technical Specification (ITS) 3.9, ''Refueling Operations,'' Subsection 3.9.3, ''Containment Penetrations,'' Limiting Condition for Operation 3.9.3.b, to state that the emergency air lock door is not required to be closed when it is sealed with the temporary cover plate.

Date of Issuance: April 28, 1999.

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

Amendment Nos.: Unit 1–303; Unit 2–303; Unit 3–303.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: 64 FR 14282 (March 24, 1999). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 28, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application of amendments: March 1, 1999.

Brief description of amendments: The amendments revised the Technical Specifications by changing the number of required channels shown in TS Table 3.3.8–1, ''Post Accident Monitoring Instrumentation'' for the Reactor Coolant System Hot Leg Temperature function from ''2 per loop'' to ''2.''

Date of Issuance: April 28, 1999.

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

Amendment Nos.: Unit 1–304; Unit 2–304; Unit 3–304

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14281). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 28, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 30, 1998.

Brief description of amendment: The amendment revises the single largest post-accident load capable of being supplied by the diesel generators and relocates this value to the Bases for Technical Specification (TS) Surveillance 4.8.1.1.2.c.3. TS Surveillance 4.8.1.1.2.c.3 has been revised to refer to ''the single largest post-accident load'' rather than a specific numerical value for diesel generator load reject testing. This change is consistent with the guidance provided in NUREG–1432 , ''Improved Standard Technical Specifications for Combustion Engineering Plants.''

Date of issuance: April 21, 1999.

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

Amendment No.: 204.

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

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56241). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: July 21, 1995.

Brief description of amendment: The amendment extends the expiration date of Operating License NPF–29 for Grand Gulf Nuclear Station, Unit 1, from June 16, 2022, to November 1, 2024. The extended date is 40 years from the date the full-power license was issued for the plant on November 1, 1984.

Date of issuance: April 26, 1999.

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

Amendment No: 137.

Facility Operating License No. NPF–29: Amendment revises Operating License No. NPF–29.

Date of initial notice in **Federal Register**: August 16, 1995 (60 FR 42605). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 26, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: December 16, 1998.

Brief description of amendment: The amendment changes Technical Specification (TS) Section 2.1.1.2, ''Reactor Core [Safety Limits],'' by revising the two recirculation loop Minimum Critical Power Ratio (MCPR) limit from 1.13 to 1.12 and the single recirculation loop MCPR limit from 1.14 to 1.13. The revised limits are required to address the River Bend Cycle 9 core design and operation. The proposed TS changes are scheduled to be implemented following refueling outage 8, currently scheduled to begin in April 1999.

Date of issuance: April 27, 1999.

Effective date: As of the date of issuance to be implemented prior to the startup following refueling outage 8.

Amendment No.: 105.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9190). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: October 8, 1998, as supplemented April 15, 1999.

Brief description of amendment: The amendment implements the Boiling Water Reactor Owners Group Enhanced Option I-A for the reactor stability long-term solution to the neutronic and thermal hydraulic instability that is documented in NEDO–32339, Revision 1, ''Reactor Stability Long-Term Solution, Enhanced Option I-A.''

Date of issuance: May 5, 1999.

Effective date: As of the date of issuance and shall be implemented during refueling outage 8.

Amendment No.: 106.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64112). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 1999.

No significant hazards consideration comments received: No.

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

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit 3, Citrus County, Florida

Date of application for amendment: October 30, 1998, as supplemented March 31, 1999.

Brief description of amendment: The amendment proposed to revise the Final Safety Analysis Report (FSAR) and associated Improved Technical Specification (ITS) Bases to reflect changes in the methodology for the B spent fuel pool criticality analysis. The proposed change is necessary due to Boraflex degradation in the B spent fuel pool storage racks.

Date of issuance: April 27, 1999.

Effective date: April 27, 1999.

Amendment No.: 175.

Facility Operating License No. DPR–72: Amendment approves changes to the FSAR and ITS Bases.

Date of initial notice in **Federal Register**: December 30, 1998 (63 FR 71966). The supplemental letter dated March 31, 1999, did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit 3, Citrus County, Florida

Date of application for amendment: January 27, 1999.

Brief description of amendment: The change would allow a one-time extension of approximately 2 months of the steam generator tube inspection interval in order for the inspection to coincide with the next planned refueling outage.

Date of issuance: May 5, 1999.

Effective date: May 5, 1999.

Amendment No.: 176.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 10, 1999 (64 FR 11962). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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

Date of application for amendment: December 31, 1997, as supplemented May 15, 1998, September 15, 1998, November 25, 1998, and January 25, 1998.

Brief description of amendment: This change modified the St. Lucie Unit 2 Technical Specifications to increase the capacity of the spent fuel storage pool, in part, by allowing a credit for a certain soluble boron concentration in the spent fuel pool.

Date of Issuance: May 6, 1999.

Effective Date: Upon issuance of license amendment package with implementation by the end of the next scheduled refueling outage, currently scheduled for April of 2000.

Amendment No.: 101.

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

Date of initial notice in **Federal Register**: February 11, 1998 (63 FR 6985) and December 16, 1998 (63 FR 69340). Following the receipt of the supplement dated November 25, 1998, and the staff's subsequent no significant hazards consideration determination (63 FR 69340), the supplement dated January 28, 1999, contained clarifying information that did not change the no significant hazards consideration determination. An additional notice was required, in accordance with 10 CFR 2.1107, due to an oversight (64 FR 16502, April 5, 1999). An environmental assessment has been published in the **Federal Register** (64 FR 23133, April 29, 1999). In that assessment, the Commission determined that the issuance of this amendment will not result in any environmental impacts other than those evaluated in the Final Environmental Statement.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 6, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981–5596.

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: February 24, 1999.

Brief description of amendments: The amendments changed Technical Specification (TS) 3/4.7.4 to permit the option of monitoring the ultimate heat sink temperature afer the intake cooling water (ICW) pumps but before the component cooling water heat exchangers which is considered to be equivalent to temperature monitoring before the ICW pumps.

Date of issuance: May 5, 1999.

Effective date: May 5, 1999.

Amendment Nos.: 200 and 194.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised the TS.

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14282). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 5, 1999.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: July 14, 1998

Brief description of amendment: The proposed amendment changed the Technical Specifications to revise the liquid and gaseous release rate limits to reflect revisions to 10 CFR Part 20, ''Standards for Protection Against Radiation.''

Date of issuance: May 3, 1999.

Effective date: May 3, 1999, to be implemented within 30 days from the date of issuance.

Amendment No.: 163.

Facility Operating License No. DPR–36: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 13, 1999 (64 FR 2249). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 30, 1997.

Brief description of amendment: The proposed amendment revises portions of Facility Operating License No. DPR–36 to delete License Conditions 2.B.6.c, 2.B.6.e, 2.B.6.f, 2.b.6.g, 2.b.7(a), and 2.B.7(b) which are no longer applicable due to the permanently shutdown and defueled condition of the Maine Yankee Atomic Power Station. Orders dated May 23, 1980, August 29, 1980, and September 19, 1980, are rescinded due to their being superseded by the equipment qualification rule (10 CFR 50.49).

Date of issuance: May 5, 1999.

Effective date: May 5, 1999, and shall be implemented within 30 days from the date of issuance.

Amendment No.: 164.

Facility Operating License No. DPR–36: The amendment revised the Operating License.

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63978). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 1999.

No significant hazards consideration comments received: No.

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

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: January 14, 1998, as supplemented by letters dated May 19, 1998, September 28, 1998, and three letters dated February 5, 1999.

Brief description of amendments: The amendments authorize revisions to the licensing basis as described in the Final Safety Analysis Report (FSAR) Update to incorporate the modification to the 230 kV offsite power system.

Date of issuance: April 29, 1999.

Effective date: April 29, 1999, and shall be implemented in the next periodic update to the FSAR Update in accordance with 10 CFR 50.71(e).

Amendment Nos.: Unit 1–132; Unit 2–130.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Final Safety Analysis Report Update.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53952). The supplemental letters dated September 28, 1998, and the three letters dated February 5, 1999, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: September 3, 1998, as supplemented by letters dated January 22, 1999, February 5, 1999, and March 17, 1999.

Brief description of amendments: The amendments change the Technical Specifications to revise TS 3/4.4.9.1 Figures for heatup and cooldown to extend their applicability to 16 effective full power years.

Date of issuance: May 3, 1999.

Effective date: May 3, 1999, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1–133; Unit 2–131.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 16, 1998 (63 FR69345). The supplemental letters dated January 22, 1999, February 5, 1999, and March 17, 1999 provided additional clarifying information and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 3, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of application for amendment: January 7, 1999.

Brief description of amendment: The amendment allows loading and handling of spent fuel transfer and storage casks in the Trojan fuel building.

Date of issuance: April 23, 1999.

Effective date: April 23, 1999.

Amendment No.: 199.

Facility Operating License No. NPF–1: The amendment changes the Operating License.

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9197). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of application for amendment: January 27, 1999.

Brief description of amendment: This proposed amendment would allow unloading of spent fuel transfer casks in the Trojan Fuel Building.

Date of issuance: April 23, 1999.

Effective date: April 23, 1999.

Amendment No.: 200.

Facility Operating License No. NPF–1: The amendment revises the Operating License.

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9198). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of application for amendment: February 12, 1997.

Brief description of amendment: The amendment deletes the Independent Spent Fuel Storage Installation area from the Permanently Defueled Technical Specifications.

Date of issuance: May 5, 1999.

Effective date: May 5, 1999.

Amendment No.: 201.

Facility Operating License No. NPF–1: The amendment changes the Permanently Defueled Technical Specifications.

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9196). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

Public Service Electric & Gas Company, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of application for amendment: January 15, 1999, as supplemented on March 31, 1999.

Brief description of amendment: The amendment allows a one-time extension of the Technical Specification (TS) surveillance interval to the end of fuel Cycle 13 (IR13) for certain TS surveillance requirements (SRs). Specifically, the amendment extends the surveillance interval in (a) SR 4.3.2.1.3 for the instrumentation response time and sequence testing of each engineered safety features actuation system (ESFAS) function; (b) SRs 4.8.2.3.2.f and 4.8.2.5.2.d for service testing of the 125-volt DC and the 28-volt DC distribution system batteries, respectively; (c) SR 4.8.2.5.2.c.2 for verification of the condition of the 125-volt DC battery connections; (d) SR 4.8.3.1.a.1.a and 4.8.3.1.a.1.b for channel calibration and integrated system functional test for containment penetration conductor protection; (e) SR 4.1.2.2.c for verification that each automatic valve in the reactivity control system flow path actuates on a safety injection (SI) test signal; (f) SRs 4.3.1.1.1,Table 4.3–1, 4.3.2.1.1, Table 4.3–2, 4.3.3.5, Table 4.36, and 4.3.3.7, Table 4.3–11 for the channel calibration of containment water level-wide range, the manual solid-state protection system (SSPS) functional input check, and the ESFAS manual initiation channel functional test; (g) SR 4.5.1.d for verification that each accumulator isolation valve opens automatically on an SI test signal; (h) SR 4.5.2.e.1 for verification that each automatic valve in the ECCS flow path actuates on an SI test signal, (i) SR 4.7.6.1.d.2 for verification that the control room emergency air conditioning system automatically actuates in the pressurization mode on an SI test signal or control room intake high radiation test signal; (j) SR 4.7.10.b for verification that each automatic valve in the chilled water loop actuates on an SI signal; and (k) SR 4.8.1.1.2.d.7 which requires a test to verify that each emergency diesel generator operates for at least 24 hours. The SRs are to be completed during the next refueling outage (1R13), prior to returning the unit to Mode 4 (hot shutdown) upon outage completion. The amendment also makes some administrative and editorial changes on some of the pages that will be affected by the above SR interval extensions.

Date of issuance: May 4, 1999.

Effective date: May 4, 1999.

Amendment No.: 222.

Facility Operating License No. DPR–70: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 10, 1999 (64 FR 6709). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 4, 1999.

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: February 8, 1999.

Brief description of amendments: The amendments revise Technical Specification 4.5.3.2.b to allow the option of using closed and disabled automatic valves to provide the necessary isolation function when performing safety injection and charging pump testing in Modes 4, 5, and 6 (i.e., hot shutdown, cold shutdown, and refueling) for low temperature overpressurization protection.

Date of issuance: April 26, 1999.

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

Amendment Nos.: 220 and 202.

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

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14284). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 26, 1999.

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: March 26, 1998.

Brief description of amendments: The amendments revise Technical Specification 3/4.8.2.1, ''AC Distribution—Operating,'' to add operability conditions and associated action statements for the 115-volt vital instrument bus (VIB) D and inverter. The amendments complete the recommended action from NRC Generic Letter 91–11, Resolution of Generic Issues 48, ''LCOs for Class 1E Vital Instrument Buses,'' and 49, ''Interlocks and LCOs for Class 1E Tie Breakers,'' pursuant to 10 CFR 50.54(f), dated July 18, 1991.

Date of issuance: April 30, 1999.

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

Amendment Nos.: 221 and 203.

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

Date of initial notice in Federal Register: May 6, 1998 (63 FR 25117). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 30, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 1, 1999.

Brief description of amendment: The amendment revises the Ginna Station Improved Technical Specifications battery cell parameters limit for specific gravity (Surveillance Requirement (SR) 3.8.6.3 and SR 3.8.6.6).

Date of issuance: April 23, 1999.

Effective date: April 23, 1999.

Amendment No.: 74.

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

Date of initial notice in Federal Register: March 24, 1999 (64 FR 14284). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

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: January 15,1999 (TS 98–07).

Brief description of amendments: The amendments change the Technical specifications (TS) by adding a new action statement to TS 3.1.3.2, ''Position Indicating Systems—Operating,'' that eliminates the need to enter TS 3.0.3 whenever two or more individual rod position indications per bank may be inoperable. It also allows additional time to determine the position of the non indicating rod(s).

Date of issuance: May 4, 1999.

Effective date: May 4, 1999.

Amendment Nos.: 244 and 235.

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

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR

9201). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 4, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 3, 1998.

Brief description of amendment: The amendment makes changes to the Technical Specifications to more clearly describe the emergency core cooling system actuation instrumentation for the low pressure coolant injection and core spray systems.

Date of Issuance: April 26, 1999.

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

Amendment No.: 170.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 10, 1999 (64 FR 6714). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 26, 1999.

No significant hazards consideration comments received: No.

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

Virginia Electric and Power Company, et al., Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of application for amendments: July 28, 1998.

Brief description of amendments: The amendments revise the Technical Specifications Section 4.6.2.2.1.b for Units 1 and 2 casing cooling and outside recirculation spray pumps surveillance testing criteria.

Date of issuance: April 22, 1999.

Effective date: April 22, 1999.

Amendment Nos.: 219 and 200.

Facility Operating License Nos. NPF–4 and NPF–7. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48272). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 22, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: October 23, 1998.

Brief description of amendment: The amendment revises Technical Specification 3/4.5.1, "Emergency Core Cooling Systems—Accumulators," by increasing the allowed outage time with one accumulator inoperable for reasons other than boron concentration deficiencies from 1 hour to 24 hours. The corresponding Bases section was also revised.

Date of issuance: April 27, 1999.

Effective date: April 27, 1999, to be implemented within 30 days from the date of issuance.

Amendment No.: 124.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64127). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of no Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action.

Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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 room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By June 18, 1999, 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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

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

Southern California Edison Company, et al., Docket No. 50–361, San Onofre Nuclear Generating Station, Unit No. 2, San Diego County, California

Date of application for amendment: April 24, 1999.

Brief description of amendment: This one-time temporary amendment allows the facility to be outside the licensing basis regarding remote shutdown capability of the shutdown cooling system as described in the Updated Safety Analysis Report, Section 5.4.7.1.2, during the period of the repair. The amendment is effective for 7 days from the date of issuance or until the repair of the check valves is completed, whichever occurs first.

Date of issuance: April 26, 1999.

Effective date: April 26, 1999, and is effective for 7 days from the date of issuance or until the check valves repair is completed, whichever occurs first.

Amendment No.: 152.

Facility Operating License Nos. NPF–10: This amendment approved a one-time change to the design basis as described in the Updated Safety Analysis Report.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, consultation with the State of California, and final no significant hazards consideration determination are contained in a Safety Evaluation dated April 26, 1999.

Attorney for Licensee: T.E. Qubre, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

NRC Section Chief: Stephen Dembek.

Dated at Rockville, Maryland, this 12th day of May 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–12494 Filed 5–18–99; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Rel. No. IC–23834; 812–9600]

Morgan Stanley Dean Witter Institutional Fund, Inc., et al.; Notice of Application

May 12, 1999.

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

ACTION: Notice of application for an order pursuant to section 17(d) of the Investment Company Act of 1940 (''Act'') and rule 17d–1 under the Act.

SUMMARY OF APPLICATION: Applicants request an order to permit certain registered management investment companies to deposit their uninvested cash balances into one or more joint accounts for the purpose of investing in short-term repurchase agreements.

Applicants: Morgan Stanley Dean Witter Institutional Fund, Inc. (''MSDWIF''), Morgan Stanley Dean Witter Universal Funds, Inc. (''MSDWUF''), and Van Kampen Series Fund, Inc. (''VKSF'') (each an ''Open-End Fund'' and, collectively, the ''Open-End Funds''); The Latin American Discovery Fund, Inc., The Malaysia Fund, Inc., Morgan Stanley Africa Investment Fund, Inc., Morgan Stanley Asia-Pacific Fund, Inc., Morgan Stanley Emerging Markets Debt Fund, Inc., Morgan Stanley Emerging Markets Fund, Inc., Morgan Stanley Global Opportunity Bond Fund, Inc., The Morgan Stanley High Yield Fund, Inc., Morgan Stanley India Investment Fund, Inc., The Pakistan Investment Fund, Inc., The Thai Fund, Inc., The Turkish Investment Fund, Inc., and Morgan Stanley Russia & New Europe Fund, Inc. (each a ''Closed-End Fund'' and, collectively, the ''Closed-End Funds''); Morgan Stanley Dean Witter Investment Management, Inc. (''MSDW Investment Management''); and Miller Anderson & Sherrerd, LLP (''Miller Anderson'').

Filing Dates: The application was filed on May 10, 1995 and was amended on March 27, 1997, June 11, 1998, and December 4, 1998. Applicants have agreed to file an amendment, the substance of which is included in this notice, during the notice period.

Hearing or Notification of Hearing: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Commission's Secretary and serving applicants with a copy of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on June 7, 1999 and should be accompanied by proof of service on the applicants, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the Commission's Secretary.

ADDRESSES: Secretary, Securities and Exchange Commission, 450 Fifth Street, NW, Washington, DC 20549–0609. Applicants, c/o Richard W. Grant, Esq., Morgan, Lewis & Bockius LLP, 1701 Market Street, Philadelphia, PA 19103.

FOR FURTHER INFORMATION CONTACT: Rachel H. Graham, Senior Counsel, at (202) 942–0583, or Mary Kay Frech, Branch Chief, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

SUPPLEMENTARY INFORMATION: The following is a summary of the application. The complete application may be obtained for a fee from the Commission's Public Reference Branch, 450 Fifth Street, NW, Washington, DC 20549–0102 (tel. (202) 942–8090).

Applicant's Representations

1. The Open-End Funds are open-end management investment companies registered under the Act. Each Open-End Fund currently offers multiple portfolios (''Portfolios''). The Closed-End Funds are closed-end management investment companies registered under the Act. The Portfolios of the Open-End Funds and the Closed-End Funds are referred to collectively as the ''Funds'' and, individually, as a ''Fund.''

2. MSDW Investment Management is registered under the Investment Advisers Act of 1940 (''Advisers Act'') and serves as investment adviser to each Portfolio of MSDWIF, certain Portfolios of MSDWUF, and each Closed-End Fund. Miller Anderson is registered under the Advisers Act and serves as investment adviser to the remaining MSDWUF Portfolios. In addition, MSDW Investment Management serves as investment subadviser to twenty VKSF Portfolios, and Miller Anderson serves as investment subadviser to the remaining two VKSF Portfolios. MSDW Investment Management and Miller Anderson are subsidiaries of Morgan Stanley Dean Witter & Co. MSDW Investment Management, Miller Anderson, and all registered investment advisers now or in the future controlling, controlled by, or under common control and MSDW Investment Management or Miller Anderson are referred to as the ''Advisers'' or, individually, as an ''Adviser.''

3. Applicants request that any relief granted pursuant to the application also apply to (i) future Portfolios of the Open-End Funds and (ii) all other registered management investment companies for which an Adviser may now or in the future act as investment adviser (collectively, the ''Future Funds'').[1]

4. The U.S. assets of each Fund are held by the Chase Manhattan Bank (''Chase'') as custodian. At the end of each trading day, each Fund has, or may have, uninvested cash balances resulting primarily from share purchases that occurred late in the day and cash held in order to assure prompt payment of redemption proceeds (''Cash Balances''). The Cash Balance of each Fund generally is invested by the Fund's Adviser in short-term investments authorized by the Fund's investment policies. Currently, the advisers must make such investments separately on behalf of each Fund. Applicants asserts that these separate purchases result in certain inefficiencies that limit a Fund's return on investment of its Cash Balance.

5. Applicants propose that the Funds establish joint trading accounts or subaccounts (''Joint Accounts'') with Chase or other custodians (collectively, ''Custodians,'' and each a ''Custodian'') into which the Funds may deposit some

[1] Each Fund that currently intends to rely on the requested order is named as an applicant. Any Future Fund that relies on the requested relief will do so only in compliance with the terms and conditions of the application.