As an alternative to the proposed action, the staff considered denial of the proposed action. Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered in the Final Environmental Statements Related to the Operation of Nine Mile Point Nuclear Station, Unit No. 1 dated January 1974 (39 **Federal Register** 3309, dated January 25, 1974), or in the Final Environmental Statements Related to the Operation of Nine Mile Point Nuclear Station, Unit No. 2, (NUREG–1085) dated May 1985.

Agencies and Persons Contacted:

In accordance with its stated policy, on September 10, 1998, the staff consulted with the New York State official, Mr. Jack Spath, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see NMPC's application dated July 21, 1998, 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 located at the Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Dated at Rockville, Maryland, this 14th day of September 1998.

For the Nuclear Regulatory Commission.

S. Singh Bajwa,

Director, Project Directorate I–1, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 98–25415 Filed 9–22–98; 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 August 28, 1998, through September 11, 1998. The last biweekly notice was published on September 9, 1998 (63 FR 48256).

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 October 23, 1998, 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: June 26, 1998.

Description of amendment request: The proposed Technical Specification (TS) amendment would amend various TS pages to correct typographical errors, remove inadvertent replication of information, and update various Bases sections.

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

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

The proposed administrative changes involving typographical errors and updating the Bases reflect plant design, safety limit settings, and plant system operation previously reviewed and approved by the NRC. These changes, therefore, do not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

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

These proposed changes do not involve any potential initiating events that would create a new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

These changes reflect information previously reviewed and approved by the NRC. The proposed changes will make the information in the Technical Specifications consistent with that already approved by the NRC. Therefore, it is concluded that the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: April 25, December 23, 1996, August 8, September 5, 1997, March 26, July 31, and August 24, 1998. The August 24, 1998, supplement supersedes the previous no significant hazards consideration determination included in letters dated April 25, 1996, and March 26, 1998 for the EDG AOT.

Description of amendment request: The proposed Technical Specification (TS) amendment would extend the Emergency Diesel Generator (EDG) allowed outage time (AOT) from 72 hours to 14 days. In support of this change the licensee has proposed various TS changes to decrease the consequences of the extended AOT.

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

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

Operation of Pilgrim Nuclear Power Station in accordance with the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

An Individual Plant Examination (IPE) for Internal Events was submitted to the NRC in response to Generic Letter 88–20 in September 1992. The supporting probabilistic safety analysis (PSA) model was updated as described in BECo letter 95–127, dated December 28, 1995. The updated PSA model was used to quantify the overall impact of the proposed EDG 14-day AOT on core damage frequency. Part III of BECo No. 2.96.040 provides the results of a comprehensive [probabilistic safety assessment] PSA of the impact of the proposed AOTs for the EDGs and [startup transformer] SUT and [shutdown transformer] SDT. As shown in Part III, there is no significant increase in risk due to the proposed change. Thus, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The existing specification 3.9.B.1 is separated into two segments (a and b) because of the proposed different AOTs for the SUT and SDT transformers. As a result of the PSA, the AOT for the SUT (a) is reduced from 7 days to 72 hours, while the AOT for the SDT (b) remains at 7 days. The reduction of the AOT from 7 days to 3 days is based on the relative risk importance of the SUT support to the balance of plant systems. Similarly, an additional reduction from 72 hours to 48 hours is proposed in the AOT for a simultaneous loss of both the SUT or SDT and an EDG (TS 3.9.B.4) based upon the SUT's or SDT's contribution to risk and that two power sources have been removed from the associated bus. The AOT reductions represent a measurable decrease in risk as assessed in the PSA. Thus, the probability or consequences of an accident previously evaluated are not increased.

The current technical specifications allow one EDG to be out of service for three days based on the availability of the SUT and SDT and the fact that each EDG carries sufficient engineered safeguards equipment to cover all design basis accidents. Additionally, the SDT can provide adequate power for one train of ESF equipment for all operating, transient, and accident conditions. With one EDG out of service and a Loss of Offsite Power (LOOP) condition, the capability to power vital and auxiliary system components remains available via the other EDG. Increasing the EDG AOT to 14 days provides flexibility in the maintenance and repair of the EDGs. The EDG unavailability will be monitored and trended in accordance with the Maintenance Rule. The PSA analyses supports the change to a 14 day AOT for the EDGs based on an insignificant increase in overall risk. Implementation of the proposed change is expected to result in less than a one percent increase in the baseline core damage frequency (2.84E–05/yr), which is considered to be insignificant relative to the underlying uncertainties involved with PSA. An additional condition is added requiring the SBO–DG to remain operable for extending the inoperable EDG AOT from 3 days to 14 days, thereby assuring that one EDG and SBO–DG are available during the extended EDG AOT. Thus, the 14-day EDG AOT does not involve an increase in the probability or consequences of an accident previously evaluated.

The proposed addition of the CRMP does not involve a significant increase in the probability or consequences of an accident previously evaluated. Because the changes are administrative in nature and deal only with risk assessment, they have no bearing on accident initiation or mitigation. Therefore, the changes will not affect the probability or consequences of an accident previously evaluated.

The proposed change does not affect the design or performance of the EDGs, and the change will not result in a significant increase in the consequences or probability of an accident previously analyzed. These changes do not involve a increase in the probability or consequences of an accident previously analyzed.

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

The operation of PNPS in accordance with the proposed license amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated because of the following:

The proposed amendment will extend the action completion/allowed outage time for an inoperable EDG from 3 days to 14 days. During this extension, the [station blackout diesel generator] SBO–DG is required to be operable and normal breaker configuration is required to be verified to ensure the SBO–DG is capable of energizing the safety bus associated with the inoperable EDG. These actions assure one EDG and SBO–DG are operable during extended EDG AOTs. The EDGs are designed as backup AC power sources for essential safety systems in the event of loss of offsite power. The SBO–DG is designed to cope with a station black out transient. The proposed AOT does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. The EDGs, SBO–DG and AC equipment are not accident initiators. No change is being made in the manner in which the EDGs provide plant protection. No new modes of plant operation are involved. An extended AOT for one EDG does not create a new or different kind of accident [than] previously evaluated. The PSA results concluded the risk contribution of the EDG AOT extension is insignificant.

Pilgrim has implemented an EDG reliability program to maintain reliability of EDGs. The SBO–DG is included in the reliability program, and the performance of EDGs and SBO–DG are trended for compliance with Maintenance Rule requirements. Thus, the proposed change does not introduce any new mode of plant operation or new accident precursors, involve any physical alterations to plant configurations, or make changes to system set points that could initiate a new or different kind of accident. Therefore, operation in accordance with the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The AOT for an inoperable SUT is reduced from 7 days to 72 hours based upon the PSA that was performed to quantitatively assess the risk impact of the proposed amendment. Additionally, removal of the SUT from service degrades the reliability of the offsite power system and renders the balance of plant unavailable upon a plant shutdown. The proposed reduction in AOT improves overall AC power source availability because the SUT will potentially be inoperable for shorter time periods. Therefore, reducing the AOT does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed addition of the [Configuration Risk Management Program] CRMP does not create the possibility of a new or different kind of accident from any accident previously evaluated because the CRMP will not affect the manner in which [structures, systems, and components] SSCs are designed, operated, or maintained. The administrative changes proposed will only require a risk assessment for specified plant configurations. Any risk assessments performed as a result of this program will only serve to provide plant personnel with risk insights associated with particular plant configurations. Since the changes will not impact SSCs and all accidents involving SSCs, the proposed change does not create a new kind of accident from any previously evaluated.

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

The operation of PNPS in accordance with the proposed license amendment will not involve a significant reduction in a margin of safety. As shown in Part III [of the application dated April 25, 1996), incorporation of the proposed change involves an insignificant reduction in the margin of safety (less than a one percent increase in the baseline core damage frequency (2.84E–05/yr), which is considered to be insignificant relative to the underlying uncertainties involved with PSA).

Also, the proposed changes do not significantly reduce the basis for any technical specification related to the establishment of, or the maintenance of, a safety margin nor do they require physical modifications to the plant. An additional condition is added requiring the SBO–DG to remain operable, in addition to the operable EDG associated with the redundant train while in the 14-day EDG AOT. The PSA results showed that the risk contribution of extending the AOT for an inoperable EDG is insignificant. Also, the reduction in the AOT for the SUT should improve availability thereby reducing overall risk with no reduction of the safety margin. Moreover, the proposed changes affect neither the way in which the EDGs perform their safety function nor the bases for their LCOs.

The proposed change does not involve a significant reduction in a margin of safety. The proposed administrative change to include a risk management program will not impact how plant SSCs are designed, operated, or maintained. The required risk assessments are intended to provide insights that influence decisions on the acceptability of abnormal plant configurations. These insights work in conjunction with existing inputs into the decision-making process rather than as the sole basis for making decisions. Therefore, the changes will not reduce a margin of safety.

As previously stated, implementation of the proposed changes is expected to result in an insignificant increase in: (1) power unavailability to the emergency buses (given that a loss of offsite power has occurred), and (2) core damage frequency. Implementation of the proposed changes does not significantly reduce 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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: September 1, 1998.

Description of amendment request: The licensee’s request proposes to revise Technical Specification 3/4.9.11 ‘‘Water Level—New and Spent Fuel Pools.’’ As a result of the proposed amendment, the licensee has also revised the Fuel Handling Building fuel handling accident analysis and the Containment fuel handling accident analyses.

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

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

Revising the required spent fuel pool water level will not increase the probability of a fuel handling accident. There is no other physical alteration to any plant system, nor is there a change in the method in which any safety related system performs its function. Harris Nuclear Plant (HNP) has revised the fuel handling accident analyses using the conservative assumptions associated with this change. The revised fuel handling accident analyses demonstrate that dose consequences as a result of a fuel handling accident remain below 25% of the 10 CFR 100 guidelines as described in the NRC Standard Review Plan.

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

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

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated because there is no physical alteration to any plant system, other than revising spent fuel pool water level, nor is there a change in the method in which any safety related system performs its function. HNP has design features to mitigate the consequences of a loss of spent fuel pool water level which are unaffected by this change.

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

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

Revising the required spent fuel pool water level does not involve a significant reduction in the margin of safety. There is no other physical alteration to any plant system, other than revising spent fuel pool water level, nor is there a change in the method in which any safety related system performs its function. HNP has revised the fuel handling accident analyses using the conservative assumptions associated with this change. The revised fuel handling accident analyses demonstrate that dose consequences as a result of a fuel handling accident remain below 25% of the 10 CFR 100 guidelines as described in the NRC Standard Review Plan.

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

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

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

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

NRC Project Director: Pao-Tsin Kuo.

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

Date of amendment request: August 31, 1998.

Description of amendment request: The proposed amendment would change the Quad Cities Technical Specifications (TS) to reflect an increase in the maximum allowable Main Steam Isolation Valve (MSIV) leakage from 11.5 standard cubic feet per hour (scfh) to 30 scfh per valve when tested at 25 psig, in accordance with Surveillance Requirement 4.7.D.6

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:

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

The proposed change to Technical Specification Surveillance Requirement 4.7.D.6 increases the maximum allowable leakage rate for a single Main Steam Isolation Valve (MSIV) from 11.5 scfh to 30 scfh. This change has no impact on the automatic or manual closure features of the valve including automatic actuations and response times. Closure of the MSIVs is a postulated transient considered in the design basis of the plant. Since the proposed change does not impact the response characteristics of the MSIVs during a postulated transient condition, the change does not impact the probability of an accident previously evaluated.

The change in allowable MSIV leakage has been evaluated to assess the impact on control room operator dose and offsite dose levels. The radiological assessment was performed with an updated radiological methodology that included significant enhancements, such as credit for suppression pool scrubbing, updated iodine dose conversion factors, and allowance for higher burnup fuel designs. Using this revised methodology, which is consistent with current regulatory requirements, the resulting dose levels from a postulated design basis accident continue to remain below the limits established in 10 CFR 50, Appendix A, General Design Criteria 19 (GDC–19) and 10 CFR 100. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated

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

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

The safety function of the MSIVs is to provide a timely steam line isolation to mitigate the release of radioactive steam and limit reactor inventory loss under certain accident and transient conditions. The MSIVs are designed to automatically close whenever plant conditions warrant a main steam line isolation. The proposed increase in allowable MSIV leakage does not impact the MSIV's ability to perform its underlying safety function, nor does the change involve any physical features of the valves and associated steam lines to create a new or different type of accident.

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

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

The proposed increase in allowable MSIV leakage represents a nominal increase in the release of radioactivity during a design basis event. The radiological assessment was performed with an updated radiological methodology that included significant enhancements, such as credit for suppression pool scrubbing, updated iodine dose conversion factors, and allowance for higher burnup fuel designs. Using this revised methodology, which is consistent with current regulatory requirements, the resulting dose levels from a postulated design basis accident continue to remain below the limits established in 10 CFR 50, Appendix A, GDC–19 and 10 CFR 100.

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

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

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

NRC Project Director: Stuart A. Richards.

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

Date of application of amendments: June 2, 1998.

Description of amendment request: The proposed amendment relocates seismic monitoring equipment requirements from the Technical Specifications to the Technical Requirements Manual (TRM), a document which is controlled under 10 CFR 50.59.

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

CYAPCO has reviewed the proposed changes to the Technical Specifications in accordance with 10 CFR 50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

As a result of the present plant configuration which has the fuel permanently removed from the reactor, the reactor-related accidents previously evaluated (i.e., LOCA, MSLB, etc.) are no longer possible. The accidents previously evaluated that are still applicable to the plant are fuel handling accidents and gaseous and liquid radioactive releases.

There is no significant increase in the probability of a fuel handling accident since refueling operations have ceased. In fact, there is a decrease in probability of a fuel handling accident since the need to move/rearrange fuel assemblies is minimal until they are removed from the spent fuel pool (i.e., for dry cask storage or for transferring to USDOE possession). In addition, the consequences of a fuel handling accident are continuing to decrease since the fuel in the spent fuel pool is continuing to decay.

The radiological consequences of a gaseous or liquid radioactive release are bounded by the fuel handling accident during defueled operation and a spent resin fire during the reactor coolant system decontamination. With the plant defueled and permanently shutdown, the demands on the radwaste systems are lessened since no new radioisotopes are being generated by irradiation or fission. Therefore, there is no increase in the probability or consequences of a gaseous or liquid radioactive release.

The ability of the plant to withstand a seismic event is not affected by this proposed change. The seismic instrumentation does not actuate any protective equipment or serve any direct role in the mitigation of an accident. The equipment will continue to be adequately controlled by the Technical Requirements Manual (TRM) to ensure operability and alert operators to a seismic event, should one occur, so that appropriate actions can be taken. Therefore, there is no increase in the consequences of a seismic event.

This material is being transferred to the TRM. This transfer is in accordance with Generic Letter 95–10, "Relocation of Selected Technical Specifications Requirements Related to Instrumentation," dated December 15, 1995 and is consistent with the NUREG–1431, "Standard Technical Specifications, Westinghouse Plants," Volume 1, Revision 1, dated April, 1995. The removed material included in this category is Technical Specification 3/4.3.3.3 and the related tables.

Based on the above, the proposed changes to the Technical Specifications 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.

There is no change in how spent fuel is stored or moved in the spent fuel pool. Therefore, the postulated fuel handling accidents are still bounding and are still considered as credible postulated accidents.

There is no change in the design and construction of plant systems, structures and components with respect to the capability to withstand a seismic event. Therefore, the currently assumed radioactive releases are still bounding.

This material is being transferred to the TRM. This transfer is in accordance with Generic Letter 95–10 and is consistent with NUREG–1431. The removed material included in this category are Technical Specification 3/4.3.3.3 and the related tables.

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

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

The capability of the plant to withstand a seismic event or other design basis accident is determined by the design and construction of systems, structures, and components. The instrumentation is used to alert operators to the seismic event and evaluate the plant response. The NRC's Final Policy Statement on Technical Specification Improvements (SECY–93–067) stated that instrumentation to detect precursors to reactor coolant pressure boundary leakage, such as seismic instrumentation, is not included in the first criterion. As discussed above, the seismic instrumentation does not serve as a protective design feature or part of a primary success path for events which challenge fission product barriers. The NRC staff, in Generic Letter 95–10, has concluded that the seismic monitoring instrumentation does not satisfy the 10 CFR 50.36 criteria and need not be included in the technical specifications.

This material is being transferred to the TRM. This transfer is in accordance with Generic Letter 95–10 and is consistent with NUREG–1431. The removed material included in this category are Technical Specification 3/4.3.3.3.

The proposed changes to the Technical Specifications do not involve a significant reduction in a margin of safety due to the fact that the capability of the plant to withstand a seismic event or other design bases accident is not affected by this proposed change.

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 amendments request involves no significant hazards consideration.

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

Attorney for the licensee: Mr. John A. Ritsher, Esquire, Ropes & Gray, One International Place, Boston, Massachusetts, 02110.

NRC Project Director: Seymour H. Weiss, Director.

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

Date of amendment request: April 9, 1998 (NRC–98–0071).

Description of amendment request: The proposed amendment would revise the ''**'' footnote to Technical Specification (TS) 3.7.1.2, ''Emergency Equipment Cooling Water System,'' Action ''a'' and add a ''*'' footnote to TS 3.8.1.1, ''A.C. Sources—Operating,'' Action ''c'' to make the actions consistent with TS 3.3.7.5, ''Accident Monitoring Instrumentation,'' for the case of inoperable primary containment oxygen monitoring instrumentation.

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

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

The proposed change will permit operation with both of the primary containment oxygen monitoring instrument channels inoperable for up to 48 hours before requiring entry into a 12 hour shutdown statement, consistent with Technical Specification 3.3.7.5, but less restrictive than the requirements in Technical Specification 3.7.1.2 Action a and Technical Specification 3.8.1.1 Action c, which require entry into the 12 hour shutdown statement immediately if the channel in the remaining division is inoperable, followed by continued shutdown to the COLD SHUTDOWN condition. The shutdown action statement entry conditions for the primary containment oxygen monitoring instrumentation should be no more restrictive in Technical Specification 3.7.1.2 or Technical Specification 3.8.1.1, than they are in Technical Specification 3.3.7.5 for both channels being inoperable. The primary containment oxygen monitoring instrumentation provides the same non-critical function regardless of the reason for the system inoperability. The primary containment oxygen monitors provide the control room operators with indication and alarm of the oxygen concentration in the primary containment, but do not provide any automatic function to mitigate an accident. Because they perform only a monitoring function, the oxygen monitors are not associated with the initiation of any previously evaluated accident; therefore, there is no change in the probability of an accident previously evaluated.

The indication provided by the primary containment oxygen monitors is used by the control room operators to ensure that the oxygen concentration remains within limits and to help make decisions regarding the use of the Combustible Gas Control System, if necessary. Alternate methods using grab samples and laboratory analytical equipment are available for obtaining primary containment oxygen concentration if no primary containment oxygen monitoring instrumentation is available. Additionally, the loss of both oxygen analyzers is not critical for entry into the Emergency Operating Procedures. Entry conditions for the post accident control of hydrogen are based upon the primary containment hydrogen monitor readings, and both channels of primary containment hydrogen monitoring instrumentation are still required to remain operable in accordance with Technical Specification 3.3.7.5. Therefore, this change will not involve a significant increase in the consequences of a previously evaluated accident.

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

As discussed above, the primary containment oxygen monitors are indication and alarm only instruments which provide information to the control room operators. The proposed change does not introduce a new mode of plant operation, nor does it involve a physical modification to the plant. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change involves the length of time that both primary containment oxygen monitoring instrument channels may be out of service. It does not increase the out of service time beyond what is already allowed by Technical Specification 3.3.7.5 for both channels being inoperable. The primary containment oxygen monitors are indication and alarm only instruments which do not affect any parameters or assumptions used in the calculation of any safety margin associated with Technical Specification Safety Limits, Limiting Safety System Settings, Limiting Control Settings or Limiting Conditions for Operation, or other previously defined margins for any structure, system, or component. 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: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: August 24, 1998.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) to clarify, for St. Lucie Units 1 and 2, component operations to be verified in response to a containment sump recirculation signal. For St. Lucie Unit 1, the proposed amendment would modify the list of equipment that comprises an operable control room emergency ventilation system to more accurately reflect installed equipment. For St. Lucie Unit 2, license conditions related to the movement of spent nuclear fuel between units will be deleted and modified as appropriate to reflect the completion of the Unit 1 spent fuel pool re-rack activities.

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

(1) Operation of the facility in accordance with the proposed amendment would not

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

The proposed amendments do not involve accident initiators. The changes to the Unit 1 and Unit 2 Technical Specifications provide additions and clarification to component lists to ensure that explicit terms of the affected specifications are consistent with existing requirements. Other changes to the Unit 2 facility operating license simply delete superseded license conditions that have been previously satisfied and are therefore obsolete. The revisions do not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do the changes alter any assumptions or conditions in the plant safety analyses. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendments are administrative in nature and will not change the physical plant or the modes of operation defined in the facility operating licenses. The changes do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The changes proposed for the Unit 1 and Unit 2 Technical Specifications provide additions and clarification to component lists to ensure that explicit terms of the affected specifications are consistent with existing requirements. Other changes to the Unit 2 facility operating license simply delete superseded license conditions that have been previously satisfied and are therefore obsolete. The revisions do not alter the plant safety analyses or the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. Therefore, operation of either unit in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment requests: September 4, 1998.

Description of amendment requests: The proposed amendments would modify the surveillance requirements and limiting conditions for operation of the technical specifications (TS) for the reactor coolant vent system. Specifically, the proposed amendments would modify the limiting conditions for operation as specified in TS Section 3.1.A.3, Reactor Coolant Vent System, and the surveillance requirements specified in TS Section 4.18, Reactor Coolant Vent System Paths.

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

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

The proposed changes do not affect any system that is a contributor to initiating events for previously evaluated anticipated operational occurrences and design basis accidents. Neither do the changes significantly affect any system that is used to mitigate any previously evaluated anticipated operational occurrences and design basis accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the design, function, or operation of any plant component and does not install any new or different equipment, therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

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

The proposed changes do not alter the initial conditions assumed in deterministic analyses associated with either the RCS [reactor coolant system] boundary or fuel cladding, therefore these changes do not involve a significant reduction in the margins of safety.

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

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

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

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: August 25, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 2.1.2, ''THERMAL POWER, High Pressure and High Flow,'' and the Bases for TS 2.1, ''Safety Limits.'' These changes are being made to implement an appropriately conservative Safety Limit Minimum Critical Power Ratio (SLMCPR) for the upcoming Cycle 9 Hope Creek core and fuel designs.

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 derivation of the revised SLMCPRs for Hope Creek for incorporation into the Technical Specifications, and its use to determine cycle-1 specific thermal limits, have been performed using NRC approved methods. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

There are no significant increases in the consequences of an accident previously evaluated. The basis of the MCPR Safety Limit is to ensure that no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Therefore, the proposed change does not involve an 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 contained in this submittal result from an analysis of the Cycle 9 core reload using the same fuel types as previous cycles. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed Technical Specification changes do not create the possibility of a new or different kind of accident, from any accident previously evaluated.

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

The margin of safety as defined in the Technical Specification bases will remain the same. The new SLMCPRs are calculated using NRC approved methods, which are in accordance with the current fuel design, and licensing criteria. The MCPR Safety Limit remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed Technical Specification 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 Project Director: Robert A. Capra.

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

Date of amendment requests: March 6, 1998.

Description of amendment requests: The proposed amendment would modify the Technical Specifications (TS) to eliminate reference to shutdown cooling (SDC) system isolation bypass valve inverters. The proposed change would allow the licensee to replace the inverters with transfer switches.

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 staff's evaluation of the three criteria are presented below:

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

The SDC system isolation bypass valves are not considered as event initiators in the accidents analyzed in the safety analysis report. Therefore, the proposed change in how the valves are aligned to available power supplies does not affect the probability of an accident previously evaluated.

The SDC system isolation bypass valves are realigned post-accident to place the shutdown cooling system in operation. The proposed change will modify the power supply for these valves from an inverter that is supplied from the safety-related DC buses to the safety-related AC buses through a manual transfer switch. This will allow the power supplies for opposite trains' valves for SDC suction supplies to be powered from opposite trains of electrical power. The operations required to actually place SDC in operation from the control room are unaffected. The proposed change does not affect the course of any accident previously analyzed, and therefore the consequences of any accident previously evaluated are unaffected by the proposed change.

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

The SDC system isolation bypass valves are used during accident mitigations, and are not considered as credible accident initiators. Thus, modifying the manner in which power is supplied to the valves 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.

Current accident analyses assume proper operation of the SDC system to mitigate the consequences of an accident to maintain postulate offsite release below the limits of 10 CFR Part 100. The proposed change only modifies the manner in which power is made available to the valves, while retaining the current design for redundancy and diversity.

The proposed change does not, therefore, affect the current margins of safety.

Based on the above staff analysis, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

Tennessee Valley Authority, Docket No. 50–260 and 50–296, Browns Ferry Nuclear Plant Units 2, 3, Limestone County, Alabama

Date of amendment request: September 4, 1998.

Description of amendment request: The proposed amendment would revise the licensing bases for the Browns Ferry Nuclear Plant (BFN) Units 2 and 3 to credit containment pressure in excess of atmospheric pressure (containment overpressure) in the analysis for Emergency Core Cooling Systems (ECCS) pump required net positive suction head (NPSH) during design basis accident conditions. The proposed licensing bases change would be implemented by a change to the BFN Updated Final Safety Analysis Report (UFSAR).

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

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

NRC Bulletin 96–03 requested BWR [Boiling Water Reactor] owners implement appropriate measures to minimize the potential clogging of the ECCS suppression chamber strainers by potential debris generated by a LOCA [loss-of-coolant-accident]. TVA's [Tennesee Valley Authority's] proposed resolution of this issue for BFN takes credit for containment overpressure to maintain adequate ECCS pump NPSH. Containment overpressure is a result of the conditions which will exist in the containment following the pipe break inside containment. Therefore, the use of containment overpressure in the analysis of the consequences of the LOCA does not affect the precursors for the LOCA, nor does it affect the precursors for any other accident or transient analyzed in Chapter 14 of the BFN Updated Final Safety Analysis Report (UFSAR). Therefore, there is no increase in the probability of any accident previously evaluated.

The worst radiological consequences for the design basis accidents analyzed in UFSAR Chapter 14 are a result of a circumferential break of one of the recirculation loop lines inside containment. The analysis of the radiological consequences of this event assumes a two percent per day leakage from the containment. The results of this analysis are presented in Section 14.6.3 of the UFSAR and indicate substantial margin when compared to 10 CFR Part 100 limits.

The radiological consequences of the design basis accident are not increased by taking credit for the post-LOCA suppression chamber airspace pressure. Without loss of primary containment, no mechanism exists to increase the accident consequences since current leakage bounds this condition. The initial analysis does not assume differential pressure between the drywell and the suppression chamber even though one exists due to the equilibrium conditions caused by the suppression chamber airspace temperature. Specifically, the suppression chamber airspace pressure credited in the ECCS pump NPSH analyses is provided by an increase in suppression chamber vapor pressure due to the increased pool temperature, including an evaluation of the effects of containment initial conditions and leakage.

By crediting the post-LOCA suppression chamber airspace pressure in the calculation of NPSH, no requirement is created to purposely maintain a higher containment pressure than would otherwise occur; no requirement is incurred to delay operating containment heat removal equipment; no requirement is incurred to deliberately continue any condition of high containment pressure in order to maintain adequate NPSH; and no requirement is incurred for the purposeful addition of nitrogen into the containment to increase the available pressure. Therefore, the proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed use of the post-LOCA suppression chamber airspace pressure in the calculation of NPSH for the ECCS pumps does not introduce any new modes of plant operation or make physical changes to plant systems. Rather, the post-LOCA suppression chamber airspace pressure is a byproduct of the conditions that will exist in the containment after a line break inside containment. Therefore, crediting the post-LOCA suppression chamber airspace pressure in the calculation of NPSH does not create the possibility of a new or different accident.

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

The integrity of the primary containment and the operation of the ECCS systems limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a reactor coolant system line break inside primary containment. In order for the ECCS pumps to meet their design basis performance requirements, the NPSH available to the pumps throughout the duration of the accident response must meet their specific NPSH requirements. Excess NPSH margin will not improve the performance of the ECCS pumps.

The post-LOCA suppression chamber airspace pressure is a byproduct of the conditions that will exist in the containment after a line break inside containment. The credit taken for this pressure in ECCS NPSH analyses has been performed in such a manner as to assure that the actual containment overpressure will always exceed the value assumed in the analyses. The NPSH margin will exceed that credited in the NPSH analyses and ECCS pump performance will meet applicable requirements. Therefore, the proposed license amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on its 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: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: August 5, 1998 (TS 98–008).

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant (WBN) Technical Specifications (TS) and associated TS Bases to allow up to 4 hours to make the residual heat removal suction relief valve available as a cold overpressure mitigation (COMS) relief path. This condition will be applicable when entering Mode 4 from Mode 3 during a plant shutdown.

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 4 hour allowance to place the RHR [residual heat removal] relief valve in service in the proposed TS change is bounded by the current COMS TS. The COMS TS currently allows cooldown of the unit while in Mode 4 with only one operable relief path for up to 7 days. Operation in this condition is allowed by Action E.1 of LCO [limiting condition for operation] 3.4.12. The 7 day completion time considers the facts that only one of the RCS [reactor coolant system] relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining relief path during this 7 day time period is very low. Thus a failure of the single available relief path concurrent with an overpressurization event during the proposed 4 hour time period for alignment and preparation of the RHR system for service is more remote. Therefore, the proposed TS change does not increase the probability of an accident previously evaluated. Further, this change does not result in hardware or procedural changes which will affect the probability of the occurrence of an accident. Considering this, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Action E.1 of LCO 3.4.12 addresses a condition where one relief path is inoperable while in Mode 4. The completion time for Action E.1 is 7 days. The 4 hour period of operation in Mode 4 that will be allowed by the addition of Note 4 to the Applicability statement of LCO 3.4.12 is well within the bounds of the analysis for operation allowed by Action E.1. This 4 hour time allowance for placement of the RHR suction relief valve in service therefore, does not cause the initiation of any accident nor create any new [credible] limiting failure for safety-related systems and components. Since the 4 hour period is only a fraction of the 7 day time period previously authorized for operation with only a single relief path, it is not probable that an accident different from those previously evaluated will be created. Therefore, the change has no adverse effect on the ability of the safety-related systems to perform their intended safety functions.

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

The Technical Specifications currently allow one of the two required relief valves to be unavailable for 7 days (Condition E of LCO 3.4.12) while in Mode 4. In this condition (one of the two relief valves inoperable), the proposed change would permit a mode change from Mode 3 to Mode 4 while providing 4 hours to place the RHR system into service. Consequently, this change does not reduce the margin of safety since the probability of an event occurring during the 4 hour period is less than the probability of an event occurring during the 7 days permitted by Action E.1. Considering this, the proposed change does not significantly 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, TN 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET I0H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: August 6, 1998 (TS 98–007).

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant (WBN) Technical Specifications (TS) and associated TS Bases to clarify the intent of the surveillance requirements (SRs) for turbine driven auxiliary feedwater (AFW) pump. The proposed revision would allow three SRs to be performed prior to achieving 1092 psig in the steam generator (SG).

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 license amendment would revise the subject TDAFWP [turbine driven auxiliary feedwater pump] TS surveillance requirements to be consistent with the intent of the current Westinghouse MERITS TS, NUREG 1431, Revision 1. TS 3.3.2 and 3.7.5 would be revised to permit testing of the TDAFWP at SG pressures less than the no-load pressure of 1092 psig [pounds per square inch-gauge]. Under these conditions, the AFW system will continue to satisfy requirements for the analyzed design basis accidents and anticipated operational transients dependent on AFW. The design basis for the AFW system and specifically the TDAFWP will be maintained such that the AFW system and its equipment will continue to perform its safety functions because the TDAFWP test will demonstrate, on recirculation flow near pump shutoff head, the ability to deliver full rated flow to the SGs. The proposed TS change does not result in any modifications to the plant and does not alter any fission barriers or challenge fuel integrity, nor are other safety systems degraded by the subject change. Potential radiological releases are not impacted by this TS change and there are no new release pathways created. Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated for WBN.

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

The proposed TS change does not result in a modification to the plant and has no adverse affect on the ability of any safety-related system to perform its intended function. No new accident scenarios are created and no new failure modes/mechanisms or limiting single failures are created as a result of the proposed change that would prevent the AFW system from performing its safety functions. A lower test pressure than the current value of 1092 psig would have an insignificant impact on the stroke time of the Terry turbine trip and throttle valve, 1–FCV–1–51. Therefore, the proposed TS change will not result in any new or different kind of accident from any accident previously evaluated.

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

This TS change does not change an acceptance limit nor does it reduce a margin of safety associated with the acceptance criteria for any WBN accident. The safety analyses performed for WBN is not based on the SG pressure at which the TDAFWP test is conducted. Specifically, the proposed TS change clarifies requirements for the TDAFW pump testing consistent with industry practice. The capability of the SRs to detect any degradation to the TDAFWP is unaffected. The capability of the SRs to demonstrate automatic start and adequate response time of the TDAFWP is not adversely impacted. The test remains a requirement of the TS, but clarifies that the test may be conducted at a SG pressure less than no-load conditions. The proposed TS change does not reduce the margin of safety limits established to protect any fission product barriers. Therefore, the proposed TS change 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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: May 8, 1998, as supplemented on July 10, 1998.

Description of amendment request: The licensee proposed to change the maximum torus water temperature during normal operation from 100 °F to 90 °F; limit the temperature during testing to 100 °F for no more than 24 hours; and, should temperature exceed 110 °F prevent operation until the temperature is reduced to below 90 °F (changed from 100 °F). *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.

(a) The proposed change to decrease the normal operating suppression pool temperature limit from 100 °F to 90 °F will assure that the consequences of accidents previously evaluated will not be significantly increased.

A reduction in the normal operating suppression pool temperature limit provides more margin for the suppression pool as a heat sink to absorb energy from the reactor vessel following an accident. The effect of higher calculated suppression pool temperatures following an accident as a result of the effect of increased feedwater addition and decreased [residual heat removal] RHR heat exchanger heat removal does not affect the consequences of accidents previously evaluated.

Certain types of Mark I containment loading conditions are increased at lower suppression pool temperatures, but since the analysis of Mark I loads for Vermont Yankee was based on initial suppression pool temperatures between 70 °F and 90 °F, the proposed decrease in the normal operating limit to 90 °F will not affect the consequences of those particular events.

(b) The proposed change to decrease the normal operating suppression pool temperature limit from 100 °F to 90 °F will not affect the probability of accidents occurring. The accidents and transients described in the [final safety analysis report] FSAR are initiated by failures of components which are not in contact with the suppression pool water, therefore a change in the suppression pool temperature will have no affect on the probability of those accidents occurring.

(c) The proposed change to restrict operation during testing that adds heat to the suppression pool to no more than 24 hours while above the normal operating temperature limit will have no affect on the consequences of accidents previously evaluated since accidents are not assumed to be initiated during these modes of operation. This assumption is made in order to assure that plants have testing flexibility at power. In addition to the time limit placed on pool temperature, the plant enters the appropriate limiting condition for operation whenever the RHR system is placed in the suppression pool cooling mode during power operation.

(d) The proposed change to restrict operation during testing that adds heat to the suppression pool to no more than 24 hours while above the normal operating temperature limit will have no affect on the probability of an accident occurring. The accidents and transients described in the FSAR are initiated by failures of components which are not in contact with the suppression pool water, therefore a change in the duration of time at any particular suppression pool temperature will have no affect on the probability of those accidents occurring.

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.

The proposed change to decrease the normal operating suppression pool temperature limit from 100 °F to 90 °F does not change any accident initiators or the types of accidents analyzed. No new modes of equipment operation or physical plant equipment modifications are proposed. The change in predicted peak suppression pool temperature results from more conservatively calculating the effects of currently analyzed accidents. Therefore this change will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change to restrict operation during testing that adds heat to the suppression pool to no more than 24 hours with water temperature above the normal operating temperature limit will allow for appropriate testing of safety related equipment to ensure operability. This testing allowance does not create any new initiating events or transients and does not involve any new modes of operation. Therefore, this change does not create the possibility of a new or different kind of accident from those 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.

The proposed change to decrease the normal operating suppression pool temperature limit from 100 °F to 90 °F assures that the suppression pool can adequately perform its safety function without a significant decrease in the margin of safety. Each of the accidents affected by suppression pool temperature have been evaluated. The evaluation showed that a higher peak suppression pool temperature was predicted based on analysis assumptions that are more conservative tha[n] those used in the current FSAR, but that the increase in peak temperature does not have a[n] impact on containment loads and equipment operability. The principal effect of an increase in peak suppression pool temperature is the reduction of [net positive suction head] NPSH margin for the low pressure [emergency core cooling system] ECCS pumps. Operator action is credited in throttling the ECCS pump flow rates after 10 minutes for the most limiting scenarios in order to assure that available NPSH exceeds required NPSH. Operator action after 10 minutes is consistent with Vermont Yankee's design basis and Emergency Operating Procedures. The proposed reduction in the normal operating suppression pool temperature limit from 100 °F to 90 °F will provide more time for operators to take actions, if required.

Operation of the facility in accordance with the proposed change to restrict operation during testing that adds heat to the suppression pool to no more than 24 hours while above the normal operating temperature limit will not involve a significant reduction in a margin of safety because it restricts the amount of time that the facility can be operated at a suppression pool temperature above the normal operating limit.

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 Project Director: Cecil O. Thomas.

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

Date of amendment request: October 10, 1996.

Description of amendment request: The amendment would add to the WNP–2 Facility Operating License No. NPF–21, the authority to store on the WNP–2 site, byproduct, source, and special nuclear materials currently addressed by the WNP–1 Materials License 46–17694–02.

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

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

The proposed amendment does not remove or modify existing requirements or safety limits. The requirements of the [Atomic Energy] Act and 10 CFR Parts 30, 40, and 70 will govern storage of sealed byproduct and neutron sources. Operation of WNP–2 requires possession and use of similar materials, and control of such materials is currently being exercised pursuant to the requirements of the Act and 10 CFR Parts 30, 40, and 70. The additional inventory of radioactive materials is a very small percentage of that already being controlled under Operating License NPF–21. Stored materials such as those proposed are not assumed as an initiator of, or contributor to, a previously analyzed accident. Consequently, the proposed amendment 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?

The requirements of the Act and 10 CFR Parts 30, 40, and 70 will govern storage of sealed byproduct and neutron sources. These materials will be stored indefinitely, and will not be put to active use. Operation of WNP–2 requires possession and use of similar materials, and control of such materials is currently being exercised pursuant to the requirements of the Act and 10 CFR Parts 30, 40, and 70. Consequently, the proposed amendment 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?

The additional inventory of radioactive materials included in sealed byproduct and neutron sources to be stored is a very small percentage of that already being controlled under Operating License NPF–21. The storage of materials does not impact the normal or emergency operation of the plant. No change to the manner in which the plant is operated is proposed. No modification to the facility is proposed. Consequently the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: M. H. Philips, Jr., Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005–3502.

NRC Project Director: William H. Bateman.

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

Date of amendment request: October 15, 1996, as supplemented by letter dated December 4, 1997.

Description of amendment request: This amendment would modify the secondary containment and standby gas treatment system (SGTS) technical specifications to more accurately reflect the existing design by revising the secondary containment and SGTS surveillance requirements to reflect a revised flow rate, revising the secondary containment integrity surveillance requirements by establishing an acceptable operating region as a function of secondary containment differential pressure and SGTS system flow, and deleting the existing requirement to maintain the secondary containment at greater than or equal to 0.25 inch of vacuum water gauge at all times.

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?

Secondary containment and the Standby Gas Treatment (SGT) system are not initiators or precursors to any accident. The SGT system acts as part of secondary containment to minimize and control airborne radiological releases from the plant following a design basis accident. Therefore, operation of WNP–2 in accordance with the proposed changes will not cause a significant increase in the probability of an accident previously evaluated.

The proposed amendment to the Technical Specifications impacts the capability to demonstrate that the secondary containment and SGT system designs will maintain radioactive releases within 10 CFR 100 guidelines and 10 CFR 50, Appendix A, General Design Criteria 19 limits. As a result, a new (current) design basis accident dose analysis was performed using the source term criteria outlined in Regulatory Guide 1.3, ''Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,'' to evaluate the proposed changes. The new analysis provides a conservative representation of the timing and release of radioactivity during a design basis accident.

The proposed amendment also deletes the normal (nonsafety-related) secondary containment ventilation system surveillance requirement to verify every 24 hours that the pressure within secondary containment is less than or equal to 0.25 inch of vacuum water gauge. This surveillance requirement is not necessary as current Technical Specification Limiting conditions for Operation (LCOs) as well as the WNP–2 Final Safety Analysis Report (FSAR) adequately address secondary containment integrity requirements and ensure secondary containment effluent is monitored. Deletion of the surveillance requirement has no impact on the secondary containment drawdown analysis or the design basis dose analysis. Thus, the analyses assumptions and conclusions remain valid.

The secondary containment and SGT system designs must accommodate a post-accident single failure and remain operable. In addition, certain plant specific parameters, such as SGT capacity, secondary containment in-leakage, outside meteorological conditions, secondary containment heat loads, available cooling capacity, emergency diesel start time and loading sequence, and drawdown time for secondary containment must be considered in the design analyses and dose assessments. The current design in conjunction with an assumed secondary containment leakage of 2240 cfm and a drawdown time of 20 minutes provide assurance that the radiological doses for a design basis accident are maintained below the 10 CFR 100 guidelines and 10 CFR 50, Appendix A, General Design Criteria 19 limits.

The dose analysis supporting the proposed amendment to the Technical Specifications includes analytical changes to the SGT flow rate, secondary containment drawdown time, mixing, and bypass leakage, and established a 95% meteorological basis. These analytical changes, in combination, result in a calculated increase in the offsite thyroid dose values and a decrease in the whole body dose values. Although the calculated offsite thyroid dose values are higher than previously calculated, they remain within the 10 CFR 100 guidelines and 10 CFR 50, Appendix A, General Design Criteria 19 limits. In accordance with Standard Review Plan (NUREG–0800), Section 15.6.5, ''Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary,'' the radiological consequences of a design basis accident are considered acceptable if they are within the guidelines of 10 CFR 100. Since the offsite thyroid dose values remain within these acceptance criteria, and since there is no increase in the control room thyroid dose values or any of the whole body dose value, the changes are considered acceptable and operation of WNP–2 in accordance with the proposed amendment to the Technical Specifications will not cause a significant increase in the 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?

Secondary containment and the SGT system are not initiators or precursors to any accident. The SGT system acts as part of secondary containment to minimize and control airborne radiological releases from the plant following a design basis accident.

The dose analysis supporting the proposed amendment to the Technical Specifications includes analytical changes to the SGT flow rate, secondary containment drawdown time, mixing, and bypass leakage, and establish a 95% meteorological basis. These analytical changes do not alter any safety-related equipment or functions or create any new failure modes. The changes will improve the capability of secondary containment and the SGT system to mitigate the consequences of a design basis accident by ensuring that secondary containment pressure can be drawn down from 0 inches water gauge to at least 0.25 inch of vacuum water gauge during the most adverse environmental conditions. The proposed changes reflect consideration of SGT capacity, secondary containment in-leakage, outside meteorological conditions, secondary containment heat loads, available cooling capacity, emergency diesel start time and loading sequence, and drawdown time for the limiting secondary containment elevation. Required instrumentation have been evaluated to ensure proper operation under normal and accident environmental conditions, including but not limited to pressure, humidity, seismic, temperature, and radiation. The evaluation method is based on American National Standards Institute/Instrument Society of America (ANSI/ISA) Standard S67.04–1988, ''Setpoints for Nuclear Safety-Related Instrumentation,'' and guidelines in ISA draft Recommended Practice RP67.04, ''Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation.''

The proposed amendment to the Technical Specification does not change plant equipment or functions, but serves to clarify and credit existing design features. Fault tree and single failure analyses were performed to ensure that the SGT system design, including the equipment and components, credited in the licensing basis for the proposed amendment meet the single failure criteria for credible failure modes. The proposed amendment also deletes the normal (nonsafety-related) secondary containment ventilation system surveillance requirement to verify every 24 hours that the pressure within secondary containment is less than or equal to 0.25 inch of vacuum water gauge. Deletion of this surveillance requirement does not invalidate existing analyses or change plant equipment or functions. Thus, no new failure modes are created.

Based on equipment failure and qualification analyses performed and the above conclusions, the proposed amendment to the Technical Specifications does not change any safety-related equipment or functions, or create any new failure modes. Therefore, operation of WNP–2 in accordance with the proposed amendment to the Technical Specifications 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?

Consistent with the current Bases for the Technical Specifications and the WNP–2 FSAR, secondary containment and the SGT system act to minimize and control airborne radiological releases from the plant to within 10 CFR 100 guidelines and 10 CFR 50, Appendix A, General Design Criteria 19 limits following a design basis accident.

The proposed amendment to the Technical Specifications impacts the capability to demonstrate that the secondary containment and SGT system designs will maintain radioactive releases within 10 CFR 100 guidelines and 10 CFR 50, Appendix A, General Design Criteria 19 limits. As a result, a new (current) design basis accident dose analysis was performed using the source term criteria outlined in Regulatory Guide 1.3 to evaluate the proposed changes. The new analysis provides a conservative representation of the timing and release of radioactivity during a design basis accident.

The proposed amendment also deletes the normal (nonsafety-related) secondary containment ventilation system surveillance requirement to verify every 24 hours that the pressure within secondary containment is less than or equal to 0.25 inch of vacuum water gauge. This surveillance requirement is not necessary as current Technical Specification LCOs as well as the WNP–2 FSAR adequately address secondary containment integrity requirements and ensure secondary containment effluent is monitored. Deletion of the surveillance requirement has no impact on the secondary containment drawdown analysis or the design basis dose analysis. Thus, it follows that deletion of the surveillance requirement will not impact the offsite and control room dose safety margins established by these analyses.

The dose analysis includes analytical changes which increase SGT system flow rate and secondary containment drawdown time, credit mixing within secondary containment, increase bypass leakage, and establish a 95% meteorological basis. The combined effect of these analytical changes results in an increase in the calculated offsite thyroid dose values. The calculated control room thyroid dose values and all of the whole body dose values are shown to decrease. Although the new thyroid dose values are higher than previously calculated, they remain within the 10 CFR 100 guidelines and 10 CFR 50, Appendix A, General Design Criteria 19 limits. The calculated thyroid dose values at the plant exclusion area boundary (EAB) (1.2 miles) increased from 72 Rem to 114.2 Rem and the calculated thyroid dose at the low population zone (LPZ) (3 miles) increased from 251 Rem to 275.6 Rem.

The LPZ is defined as all land within a 3 mile radius of the plant site and 0 persons reside within this area. The nearest residence is 4.1 miles from the plant site. There are no schools or hospitals within 5 miles of the plant site and the nearest population center is at 12 miles. Considering the low population density in the area immediately surrounding the plant site, the increase in thyroid dose will have a small impact on the health and safety of the public.

Since the offsite thyroid dose values remain within the 10 CFR 100 guidelines and 10 CFR 50, Appendix A, General Design Criteria 19 limits, and since there is a small impact on the health and safety of the public, the increase in the offsite thyroid dose values are considered acceptable and operation of WNP–2 in accordance with the proposed amendnment to the Technical Specifications will not cause 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: M. H. Philips, Jr., Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005–3502.

NRC Project Director: William H. Bateman.

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.

Illinois Power Company, Docket No. 50–461, Clinton Power Station, DeWitt County, Illinois Date of Application for Amendment: August 24, 1998

Brief description of amendment request: The proposed amendment concerns the ''ready-to-load'' requirement for the Division 3 diesel generator (DG). The Division 3 DG requires operator action to reset the mechanical governor to meet the ''ready-to-load'' requirement.

Date of publication of individual notice in **Federal Register**: September 10, 1998 (63 FR 48529).

Expiration date of individual notice: October 13, 1998.

Local Public Document Room location: Vespasian Warner Public Library, 310 N. Quincy Street, Clinton, IL 61727.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: August 28, 1998.

Brief description of amendment request: The proposed amendment would modify Technical Specification 4.0.5 to state that the inservice testing requirement for exercise testing in the closed direction for specified Unit 1 containment isolation valves shall not be required until the next plant shutdown to Mode 5 of sufficient duration to allow the testing or until the next refueling outage scheduled in March 1999.

Date of publication of individual notice in **Federal Register**: September 9, 1998 (63 FR 48254)

Expiration date of individual notice: September 24, 1998.

Local Public Document Room location: Wharton County Junior College, J.M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: June 30, 1998.

Description of amendment request: The proposed amendment would transfer operating authority for the Perry Nuclear Power Plant, Unit No. 1, from The Cleveland Electric Illuminating Company and Centerior Service Company to a new operating company, called the FirstEnergy Nuclear Operating Company. The proposed action has been submitted pursuant to 10 CFR 50.80 and 10 CFR 50.90.

Date of publication of individual notice in **Federal Register**: August 4, 1998 (63 FR 41600).

Expiration date of individual notice: September 3, 1998.

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

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

Date of application for amendment: June 29, 1998, as supplemented July 14, 1998.

Brief description of amendment request: This amendment would reflect the approval of the transfer of the authority to operate Davis-Besse Nuclear Power Station, Unit 1, under the license to a new company, FirstEnergy Nuclear Operating Company.

Date of publication of individual notice in **Federal Register**: August 4, 1998.

Expiration date of individual notice: September 3, 1998.

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

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.

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

Date of application for amendment: October 31, 1996.

Brief description of amendment: This amendment changes Technical Specification 3/4.7.5 by reducing the maximum allowable water temperature for the Ultimate Heat Sink from 95°F to 94°F and increasing the minimum main reservoir level from 205.7 feet mean sea level to 215 feet mean sea level.

Date of issuance: September 8, 1998.

Effective date: September 8, 1998.

Amendment No: 80.

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

Date of initial notice in **Federal Register**: December 4, 1996 (61 FR 64382).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 8, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 16, 1997, as supplemented June 29, 1998. The June 29, 1998, supplemental letter provided clarifying information only, and did not change the initial no significant hazards consideration determination.

Brief description of amendment: This amendment changes Technical Specification 3/4.6.2.3 by reducing the Containment Fan Coolers cooling water flow rate requirement from 1425 gallons per minute (gpm) to 1300 gpm.

Date of issuance: September 8, 1998.

Effective date: September 8, 1998.

Amendment No: 81.

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

Date of initial notice in **Federal Register**: March 25, 1998 (63 FR 14485).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 8, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 14, 1998, as supplemented by letter dated July 17, 1998.

Brief description of amendments: The amendments change the Braidwood, Unit 1, Technical Specification limits on Reactor Coolant System Dose Equivalent Iodine-131 from 0.35 microcuries/gram to 0.05 microcuries/gram for the remainder of Cycle 7.

Date of issuance: September 3, 1998.

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

Amendment Nos.: 95 and 95.

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

Date of initial notice in **Federal Register**: March 11, 1998 (63 FR 11914). The July 17, 1998, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 3, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Duke Energy Corporation, et al., Docket No. 50–413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of application for amendment: August 6, 1998.

Brief description of amendment: The amendment deletes Surveillance Requirement 4.8.1.1.2.i.2, regarding diesel fuel oil system pressure testing, from the unit Technical Specifications for Unit 1 on the basis that the staff had previously approved alternative surveillance based on Code Case N–498–1 of the American Society of Mechanical Engineers.

Date of issuance: September 9, 1998.

Effective date: As of the date of issuance.

Amendment No.: 171.

Facility Operating License No. NPF–35: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes. (63 FR 43962 dated August 17, 1998). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by September 16, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated September 9, 1998.

Attorney for licensee: Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, North Carolina.

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

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

Date of application for amendments: August 14, 1998.

Brief description of amendments: The amendments revise Technical Specification Section 4.6.5.1.b.2 regarding surveillance requirements for the ice condenser. One current requirement specifies that a visual inspection of flow passages be performed once per 9 months to ensure that there is no significant ice and frost accumulation (less than 0.38 inch). DEC proposed to relax the visual inspection frequency of the lower plenum support structures and turning vanes to once per 18 months, while the remaining parts of the ice condenser will continue to be inspected at 9-month intervals.

Date of issuance: September 10, 1998.

Effective date: As of the date of issuance.

Amendment Nos.: Unit 1—172; Unit 2—163.

Facility Operating License Nos. NPF–35 and NPF–52: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes. (63 FR 45872 dated August 27, 1998). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by September 28, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated September 10, 1998.

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

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

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

Date of application for amendments: August 14, 1998.

Brief description of amendments: The amendments revise Surveillance Requirement 4.6.5.1.b.3 of the Technical Specifications, relaxing the visual inspection interval of the ice condenser lower plenum and turning vanes from the current 9-month to 18-month intervals.

Date of issuance: September 10, 1998.

Effective date: As of the date of issuance.

Amendment Nos.: Unit 1–180; Unit 2–162.

Facility Operating License Nos. NPF–2 and NPF–8: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes. (63 FR 45870 dated August 27, 1998). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by September 28, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated September 10, 1998.

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

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North 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 11, 1993, as supplemented August 26, October 26, November 29, and December 6, 1993, October 3, 1995, February 27, May 2, and September 3, 1997, and May 7, 1998.

Brief description of amendments: The amendments completely revise the current Technical Specifications related to the electrical distribution system and incorporate new requirements for system operation, limiting conditions for operation, and surveillance requirements.

Date of Issuance: September 4, 1998.

Effective date: As of the date of issuance, to be implemented coincident with implementation of the Improved Technical Specifications.

Amendment Nos.: Unit 1–232; Unit 2–232; Unit 3–231.

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

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63975).

The May 2, 1997, and May 7, 1998, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 4, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 28, 1998.

Brief description of amendment: The amendment proposed to revise the Improved Technical Specification 5.6.2.8 to change the scope and frequency of volumetric and surface inspections for the reactor coolant pump flywheels. The amendment approves the requested change to reflect the frequency and scope of these inspections as specified in Topical Report WCAP–14535A.

Date of issuance: August 31, 1998.

Effective date: August 31, 1998.

Amendment No.: 170.

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

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40555)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 31, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application foramendment: June 29, 1998, as supplemented July 27, 1998.

Brief description of amendment: The amendment reduces the scope of a previous amendment request dated February 22, 1996. It retains the provision to delete the requirement that the biennial inspection of the emergency diesel generators (EDGs) be performed during shutdown, permits skipping diesel starting battery capacity test for recently installed batteries, and increases the minimum loading during diesel testing from 20% to 80%. In addition, there are wording changes to enhance clarity and a typograhpical error is corrected.

Date of Issuance: September 8, 1998.

Effective date: September 8, 1998, to be implemented within 30 days.

Amendment No.: 197.

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

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40556). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 8, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room Location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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: February 22, 1996.

Brief description of amendments: The amendments revise the Technical Specifications to reference NRC Regulatory Guide 1.9, Revision 3, rather than NRC Regulatory Guide 1.108, Revision 1, for the determination of a valid diesel generator test.

Date of issuance: September 2, 1998.

Effective date: September 2, 1998, with full implementation within 45 days.

Amendment Nos.: 222 and 206.

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

Date of initial notice in **Federal Register**: April 10, 1996 (61 FR 15990).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 2, 1998.

No significant hazards consideration comments received: No.

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

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: June 10, 1998.

Brief description of amendments: The amendments defer the implementation date of Amendments Nos. 216/200 to become effective when modifications are completed but not later than December 31, 2000.

Date of issuance: August 31, 1998.

Effective date: August 31, 1998, with full implementation not later than December 31, 2000.

Amendment Nos.: 221 and 205.

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

Date of initial notice in **Federal Register**: July 31, 1998 (63 FR 40940).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 31, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 22, 1995, as supplemented on May 13, 1998.

Brief description of amendments: The amendments revise Technical Specifications 3.4.1.4 and 3.9.8.2 by deleting footnotes and associated information regarding service water system header operation to allow residual heat removal system operation to be consistent with current regulations and the Standard Technical Specifications—Westinghouse Plants (NUREG–1431).

Date of issuance: September 8, 1998.

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

Amendment Nos.: 214 and 194.

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

Date of initial notice in **Federal Register**: August 30, 1995 (60 FR 45183).

The May 13, 1998, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination, and was within the scope of the original application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 8, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 22, 1998.

Brief description of amendments: The amendments revise the technical specifications to extend the allowed outage time (AOT) for off-site circuits and for the emergency diesel generator.

Date of issuance: September 9, 1998.

Effective date: September 9, 1998, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2–141; Unit 3–133.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 31, 1998 (63 FR 40941).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 9, 1998.

No significant hazards consideration comments received: No.

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

Tennessee Valley Authority, Docket No. 50–296, Browns Ferry Nuclear Plant, Unit 3, Limestone County, Alabama

Date of application for amendment: June 2, 1995, revised March 6, 1997, as supplemented April 11, May 13, and August 20, 1997, and March 13, 1998. (TS–353).

Brief description of amendment: Revises Technical Specifications (TS) to permit implementation of upgrade of power range neutron monitor instrumentation. Other changes also have been incorporated to thermal limits specifications to implement average power range monitor and rod block monitor TS improvements, and maximum extended load line limit analyses.

Date of issuance: September 3, 1998.

Effective date: September 3, 1998.

Amendment No.: 213.

Facility Operating License No. DPR–68: Amendment revises the TS. .

Date of initial notice in **Federal Register**: August 16, 1995 (60 FR

42609). The revision dated March 6, 1997; the proposal for the same changes to be made to the Improved Standard TS format dated April 11, 1997; and the supplemental information dated May 13 and August 20, 1997, and March 13, 1998, did not affect the staff's original finding of no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 3, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611.

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

Date of application for amendments: February 13, 1998 (TS 97–04).

Brief description of amendments: The amendments change the Technical Specifications (TS) by relocating the snubber requirements from Section 3.7.9 of the TS, and its bases, to the Sequoyah Nuclear Plant Technical Requirements Manual. This change does not alter the requirements for operability or surveillance testing of the snubbers. This amendment also deletes License Condition 2.C.(19), for Unit 1 only. This condition is a one-time snubber-related action that was completed and no longer needs to be included in the SQN Operating License.

Date of issuance: August 28, 1998.

Effective date: As of the date of issuance to be implemented no later than 45 days after issuance.

Amendment Nos.: Unit 1–235 ; Unit 2–225.

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

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17235).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

Date of application for amendment: December 23, 1997.

Brief description of amendment: This amendment revised Technical Specification (TS) Section 4.4.5, ''Reactor Coolant System—Steam Generators—Surveillance Requirements (SRs).'' SR 4.4.5.8 was modified to provide flexibility in the scheduling of steam generator inspections during refueling outages.

Date of issuance: September 2, 1998.

Effective date: September 2, 1998.

Amendment No.: 226.

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

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4327).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 2, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 30, 1998.

Brief description of amendment: The licensee proposes to delete the calibration requirements for emergency core cooling actuation instrumentation—core spray (CS) subsystem and low pressure coolant injection (LPCI) system auxiliary power monitor since the relays operate from a switched input and functional testing is sufficient to demonstrate the relay pickup/dropout capability.

Date of Issuance: September 1, 1998.

Effective date: September 1, 1998, to be implemented within 30 days.

Amendment No.: 162.

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

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40563).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 1, 1998.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 17th day of September 1998.

For The Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–25281 Filed 9–22–98; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–23439; 812–10976]

The Austria Fund, Inc., The Spain Fund, Inc., and Alliance Capital Management L.P.; Notice of Application

September 17, 1998.

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

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

SUMMARY OF APPLICATION: Applicants request an order under section 6(c) of the Act granting an exemption from section 19(b) of the Act and rule 19b–1 under the Act to permit certain registered closed-end investment companies to make periodic distributions of long-term capital gains in any one taxable year pursuant to a distribution policy with respect to common stock.

APPLICANTS: The Austria Fund, Inc. (''Austria Fund''), The Spain Fund, Inc. (''Spain Fund''), and Alliance Capital Management L.P. (''Alliance'') on behalf of each other existing and each future closed-end management investment company that is advised by Alliance or by an entity controlling, controlled by or under common control with Alliance (collectively, the ''Funds'').

FILING DATE: The application was filed on January 20, 1998 and amended on September 16, 1998.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicants with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on October 13, 1998, and should be accompanied by proof of service on applicants, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the SEC's Secretary.