PENETRATION SEAL MATERIALS AND OTHER MINOR CHANGES'' (10 CFR PART 50) (WITS 199800128)'' (PUBLIC MEETING) be held on May 25, and on less than one week's notice to the public.

The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: May 25, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–13674 Filed 5–26–00; 12:46 am]

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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 May 6, 2000, through May 19, 2000. The last biweekly notice was published on May 17, 2000 (65 FR 31354).

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

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

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

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

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

By June 30, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of amendment request: April 18, 2000.

Description of amendment request: The amendments would revise Technical Specifications (TS) 3.7.10 and 3.7.12 for Catawba Units 1 and 2. The proposed changes address degraded pressure boundaries on the Auxiliary Building Filtered Ventilation Exhaust System and the Control Room Area Ventilation System. The proposed changes in TS 3.7.10 and 3.7.12 would add Notes which allow the affected ventilation system boundaries to be opened intermittently under administrative controls. Also, it would add a new condition in TS 3.7.10 and 3.7.12. This new condition will require that the boundaries for these two systems be returned to an operable status within 24 hours, when both trains of these systems are inoperable due to an inoperable boundary.

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:

In accordance with the criteria set forth in 10 CFR 50.91 and 50.92, Duke Energy Corporation has evaluated this license amendment request and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

No. The Control Room Area Ventilation System (CRAVS), Control Room pressure boundary, the Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), or the Emergency Core Cooling System (ECCS) pump rooms area pressure boundary are not assumed to be initiators of any analyzed accident. Therefore, the proposed changes contained in this LAR [license amendment request] have no significant impact on the probability of occurrence of any previously analyzed accident.

The proposed new condition for the CRAVS and ABFVES Technical Specifications (TS) would permit a 24-hour period to take action to restore an inoperable pressure boundary to OPERABLE status. The consequences of implementing the 24 hour Completion Time are reasonable based upon: (1) The low probability of a design basis accident occurring during this time period, (2) additional actions that are available to the operator to minimize doses (*e.g.,* self contained breathing apparatus and alternate control room air intakes), and (3) the availability of an operable CRAVS/ABFVES train to provide a filtered environment (albeit with potential unfiltered leakage).

For cases where any of the affected control room or ECCS pump room area/pump rooms pressure boundaries are opened intermittently under administrative controls, appropriate compensatory measures would be required by the proposed TS to ensure the pressure boundary can be rapidly restored. Based on the compensatory measures available to the plant operators and the administrative controls required to rapidly restore an opened pressure boundary, the accident consequences do not cause an increase in dose above the applicable General Design Criteria, Standard Review Plan, or 10 CFR 100 limits. The plant operators will continue to maintain the ability to mitigate a design basis event.

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

No. No changes are being made to actual plant hardware which will result in any new accident causal mechanisms. Also, no changes are being made to the way in which the plant is being operated. Therefore, no new accident causal mechanisms will be generated.

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

No. Margin of safety is related to the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be significantly degraded by the proposed changes. The proposed changes would allow affected pressure boundaries to be degraded for a limited period of time (24 hours).

However, the probability of a design basis event occurring during this time is low and additional actions (*e.g.,* breathing apparatus) would also be taken to minimize dose to the plant operators. When the boundaries are open on an intermittent basis, as permitted by the changes proposed in this LAR, administrative controls would be in place to ensure that the integrity of the pressure boundaries could be rapidly restored. Therefore, it is expected that the plant, and the operators, would maintain the ability to mitigate design basis events and none of the fission product barriers would be affected by this change. Therefore, the proposed change is not considered to result in a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 5, 2000.

Description of amendment request: The proposed amendment implements technical specification (TS) changes associated with thermal-hydraulic stability monitoring. New TS 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' will provide the minimum operability requirements for the OPRM channels, the Required Actions when they become inoperable, and appropriate surveillance requirements. The OPRMs will provide automatic ''detect and suppress'' actions to replace the administrative controls currently in effect through operator training and manual actions. The amendment would remove monitoring guidance from TS 3.4.1, ''Recirculation Loops Operating,'' that will no longer be necessary due to the activation of the automatic OPRM instrumentation. Finally, the amendment would update TS 5.6.5, ''Core Operating Limits Report (COLR),'' to require the applicable setpoints for the OPRMs to be included in the COLR.

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

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

The proposed change specifies limiting conditions for operation, required actions and surveillance requirements for the Oscillation Power Range Monitor (OPRM) system, and allows operation in regions of the power to flow map currently restricted by the requirements of Interim Corrective Actions (ICAs) and certain limiting conditions of operation of Technical Specification (TS) 3.4.1. The restrictions of the ICAs and TS 3.4.1 were imposed to ensure adequate capability to detect and suppress conditions consistent with the onset of thermal-hydraulic (T–H) oscillations that may develop into a T–H instability event. A T–H instability event has the potential to challenge the Minimum Critical Power (MCPR) safety limit. The OPRM system can automatically detect and suppress conditions necessary for T–H instability. With the activation of the OPRM System, the restrictions of the ICAs and TS 3.4.1 will no longer be required.

The probability of a T–H instability event is impacted by power to flow conditions during operation inside specific regions of the power to flow map, in combination with power shape and inlet enthalpy conditions, such that only under such conditions can the occurrence of an instability event be postulated to occur. Operation in these regions may increase the probability that operation with conditions necessary for a T–H instability can occur. However, when the OPRM is OPERABLE with operating limits as specified in the Core Operating Limits Report (COLR), the OPRM can automatically detect the onset of significant local power oscillations and generate a trip signal. Actuation of a Reactor Protection System (RPS) trip will suppress conditions necessary for T–H instability and decrease the probability of a T–H instability event. In the event the trip capability of one or more of the OPRM channels is not maintained, the proposed change includes Required Actions which limit the period of time before the affected OPRM channel (or RPS system) must be placed in the tripped condition. If these actions would result in a trip function such as a scram, or if the OPRM trip capability is not maintained, an alternate method to detect and suppress thermal hydraulic oscillations is required, i.e., the same ICAs as are in place today. In either case the duration of the period of time allowed by the Required Actions is limited, and the probability of a T–H instability event during this limited time is not significantly increased.

Several changes to TS 3.4.1 are made which are more consistent with, or conservative with, respect to the reviewed and approved Standard Technical Specifications for Boiling Water Reactors. These generic changes are considered applicable to the Perry Nuclear Power Plant. They simply provide guidance on the operator actions to be taken and the associated time limits when the Specification is entered, and do not impact the probability of occurrence of an accident. For the above reasons, the proposed change does not result in a significant increase in the probability of an accident previously evaluated.

An unmitigated T–H instability event is postulated to cause a violation of the MCPR safety limit. The proposed change ensures mitigation of T–H instability events prior to challenging the MCPR safety limit if initiated from anticipated conditions, by detection of the onset of oscillations and actuation of an RPS trip signal. The OPRM also provides the capability of an RPS trip being generated for T–H instability events initiated from unanticipated but postulated conditions. These mitigating capabilities of the OPRM system will become available as a result of the proposed change and have the potential to reduce the consequences of anticipated and postulated T–H instability events. The OPRM installation has been evaluated to not adversely impact other installed equipment such as the Average Power Range Monitors (APRMs) or the RPS in a manner that could prevent response to various postulated events, so those events will not have increased consequences due to the OPRMs. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

Therefore, the proposed change, which specifies limiting conditions for operation, required actions and surveillance requirements for the OPRM system, and allows operation in certain regions of the power to flow map, does not significantly increase either the probability or consequences of an accident previously evaluated.

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

The proposed change specifies limiting conditions for operation, required actions and surveillance requirements of the OPRM system, and allows operation in regions of the power to flow map currently restricted by the requirements of ICAs and TS 3.4.1. The OPRM system uses input signals shared with APRM and rod block functions to monitor core conditions and generate an RPS trip when required. Quality requirements for software design, testing, implementation and module self-testing of the OPRM system provide assurance that new equipment malfunctions due to software errors are not created. The design of the OPRM system also ensures that neither operation nor malfunction of the OPRM system will adversely impact the operation of other systems and no accident or equipment malfunction of these other systems could cause the OPRM system to malfunction or cause a different kind of accident. Therefore, operation with the OPRM system does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Operation in regions currently restricted by the requirements of ICAs and TS 3.4.1 is within the nominal operating domain and ranges of plant systems and components, and within the range for which postulated accidents have been evaluated. Therefore operation within these regions does not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes to TS 3.4.1 to be more consistent, or conservative, with respect to the reviewed and approved Standard Technical Specifications, simply provide guidance on the operator actions to be taken and the associated time limits when the Specification is entered, and also do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed change, which specifies limiting conditions for operation, required actions and surveillance requirements of the OPRM system, and allows operation in certain regions of the power to flow map, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change specifies limiting conditions for operation, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power to flow map currently restricted by the requirements of ICAs and TS 3.4.1.

The OPRM system monitors small groups of LPRM [local power range monitor] signals for indication of local variations of core power consistent with T-H oscillations, and generates an RPS trip when conditions consistent with the onset of oscillations are detected. An unmitigated T-H instability event has the potential to result in a challenge to the MCPR safety limit. The OPRM system provides the capability to automatically detect and suppress conditions which might result in a T–H instability event, and thereby maintains the margin of safety by providing automatic protection for the MCPR safety limit while reducing the burden on the control room operators. Therefore, operation with the OPRM system does not involve a significant reduction in a margin of safety. In the event an OPRM channel becomes inoperable, the proposed change includes actions which limit the period of time before the affected OPRM channel (or RPS system) must be placed in the tripp[ed] condition. If these actions would result in a trip function such as a scram (or if the OPRM trip capability is not maintained), the alternate method to detect and suppress thermal hydraulic oscillations (the current ICAs) is required to be put in place. The duration of the period of time allowed by the Required Actions is limited, and the probability of a significant T–H instability event during this limited time is not significantly increased.

Operation in regions currently restricted by the requirements of ICAs and Technical Specification [TS] 3.4.1 is within the nominal operating domain and ranges of plant systems and components, and within the range assumed for initial conditions considered in the analysis of anticipated operational occurrences and postulated accidents. Therefore, operation in these regions does not involve a significant reduction in the margin of safety. The changes to TS 3.4.1 to be more consistent, or conservative, with respect to the reviewed and approved Standard Technical Specifications, simply provide guidance on the operator actions to be taken and the associated time limits when the Specification is entered, and also do not significantly reduce the margin of safety.

Therefore, the proposed change, which specifies limiting conditions for operation, required actions and surveillance requirements of the OPRM system, and allows operation in certain regions of the power to flow map, does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

Florida Power and Light Company, et al., Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: April 23, 2000.

Description of amendment request: The proposed license amendment (PLA) is associated with the required timing for containment hydrogen recombiner post operation insulation resistance testing. This PLA revises Unit 1 Technical Specification 3/4.6.4.2, Electric Hydrogen Recombiners—W, to clarify the requirement for the post-operation insulation resistance test of Surveillance Requirement 4.6.4.2.b.4.

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

1. 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 amendment does not involve an increase in the probability or consequences of any accident previously evaluated. This PLA provides a clarification of the Technical Specification surveillance requirements for verifying hydrogen recombiner operability and reliability. This PLA has no affect on the testing requirements, test frequency, or acceptance criteria for recombiner operability. This change allows vendor recommended guidance and in-house methodology to be established when conducting recombiner heater resistance testing. This will enable consistency in testing and will allow trending for determination of the material condition of the recombiner heaters. The PLA change provides clarification and preserves the intent of the basis to monitor the material condition of the recombiner heaters. Additionally, this change provides consistency and is identical with the Unit 2 Technical Specification surveillance. As such, this change is considered administrative in nature.

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 amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. This PLA is considered administrative in nature and will not alter the way in which the hydrogen recombiner is operated or tested. This PLA allows vendor recommended guidance to be established in order to perform consistent testing and to allow meaningful trending of the results to verify recombiner operability. This PLA has no affect on the testing requirements, test frequency, or acceptance criteria for recombiner operability. This PLA does not result in any plant configuration changes or new failure modes.

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

The proposed amendment does not involve a reduction in the margin of safety. This administrative PLA clarifies the surveillance requirement of the subject Technical Specification by allowing the establishment of vendor recommendations and in-house testing methodology to provide consistent testing conditions and allow meaningful trending of results. This PLA has no affect on the testing requirements, test frequency, or acceptance criteria for recombiner operability. As such, the assumptions and conclusions of the accident analyses in the UFSAR [Updated Final Safety Analysis Report] remain valid and the associated safety limits will continue to be met.

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

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

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company, Docket No. 50–251 and 50–252, Turkey Point Units 3 and 4 in Miami-Dade County

Date of amendment request: April 27, 2000.

Description of amendment request: Florida Power and Light Company (FPL) requests to amend the Turkey Point Unit 3 Facility Operating License DPR–31 Fire Protection license condition 3.G, and to amend the Turkey Point Unit 4 Facility Operating License DPR–41 Fire Protection license condition 3.F. The proposed revisions to the Facility Operating Licenses are required to incorporate references to NRC Safety Evaluations issued in support of 10 CFR 50 Appendix R granted exemptions. In addition, the proposed amendments requests to modify Appendix A of the Facility Operating Licenses DPR–31 and DPR–41 of the Turkey Point Units 3 and 4 Technical Specifications (TS), Section 4.7.6.g. Due to an oversight, the submittal for the request of License Amendments Nos. 201 and 195 for Section 6.0 ''Administrative Controls,'' L–99–056, dated March 8, 1999, discussed revision to TS Section 4.7.6.g on TS Page 3/4 7–21, but inadvertently did not attach the revised marked up Page 3/4 7–21.

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 amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments do not involve an increase in the probability or consequences of an accident previously evaluated because the proposed changes are administrative in nature adding references to exemptions granted by the NRC and to reflect relocation of record retention requirements from the TS to the UFSAR. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

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

The proposed changes to the Facility Operating Licenses and the Technical Specifications are administrative in nature and can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes since the proposed changes do not involve the addition or modification of equipment nor do they alter the design or operation of affected plant systems, structures, or components.

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

The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The proposed changes to the Facility Operating License Conditions and the TS are administrative in nature and do not reduce any of 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 request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: April 28, 2000.

Description of amendment request: The Seabrook Station Technical Specifications (TSs) are proposed to be revised to implement the Relaxed Axial Offset Control (RAOC) operating strategy in support of the use of upgraded Westinghouse fuel with Intermediate Flow Mixers.

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

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

The proposed changes to TS 2.1.1, 3.2.1, 4.2.1.1, 4.2.2.2, 4.2.2.3, 4.2.2.4, 6.8.1.6.b, and changes to the aforementioned TS Bases, are in support of North Atlantic's long-term operating strategy to refuel and operate, commencing with Cycle 8, with Biweekly Notice Coordinator upgraded Westinghouse fuel with Intermediate Flow Mixers (VANTAGE+(w/IFMs)). Evaluations/analyses of accidents which are potentially affected by the parameters and assumptions associated with the fuel upgrade and RAOC strategy have shown that all design standards and applicable safety criteria will continue to be met. The consideration of these changes does not result in a situation where the design, material, and construction standards that were applicable prior to the change are altered. Therefore, the proposed changes occurring with the fuel upgrade will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The proposed changes associated with the fuel upgrade and RAOC strategy do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The actual plant configuration, performance of systems, and initiating event mechanisms are not being changed as a result of the proposed changes. The design standards and applicable safety criteria limits will continue to be met and therefore fission barrier integrity is not challenged. The proposed changes associated with fuel upgrade and RAOC strategy have been shown not to adversely affect the response of the plant to postulated accident scenarios. The proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

The proposed changes to TS Table 2.2–1, TS 3.2.2, TS 3.2.3, and the title on page 3/4 2–6 are editorial changes to correct either typographical errors, simplification of statements, clarification of specific parameters associated with temperature pressure measurements, making some notations consistent with improved Standard Technical Specifications — Westinghouse Plants, NUREG–1431, Rev. 1, and relocating additional cycle-specific values for temperature, pressure and time constants to the [Core Operating Limits Report] COLR, or correcting an erroneous title. These changes do not result in a change to the design basis of any plant structure, system or component or parameters currently specified in the COLR, therefore, operation of the facility within the prescribed limits of TS remains unchanged.

The proposed change to TS 3.2.1, ACTION a.2, to delete the need to reduce the power range neutron flux high trip setpoints subsequent to reducing rated thermal power (RTP) to less than 50% whenever axial flux difference (AFD) is outside of the applicable limits specified in the COLR, does not significantly increase the probability or consequences of an accident previously evaluated.

Therefore, for the reasons stated above, the probability or consequences of an accident previously evaluated are not significantly increased for all the proposed TS changes presented herein.

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

The possibility for a new or different type of accident from any accident previously evaluated is not created since the proposed changes associated with the fuel upgrade and RAOC strategy do not result in a change to the design basis of any plant structure, system or component. These proposed changes do not cause the initiation of any accident nor create any new failure mechanisms. Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The proposed changes do not result in any event previously deemed incredible being made credible.

The proposed changes are not expected to result in conditions that are more adverse and are not expected to result in any increase in the challenges to safety systems.

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

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

The proposed changes will assure continued compliance within the acceptance limits previously reviewed and approved by the NRC for use of upgraded fuel features with RAOC. All of the appropriate acceptance criteria for the various analyses and evaluations will continue to be met.

The proposed editorial changes do not change the current limits specified in Technical Specifications.

Removing the requirement for manually reducing the power range neutron flux high trip setpoint does not result in a significant reduction in a margin of safety. There are other levels of trip protection to terminate a rapid rise in power excursion, such as the overtemperature delta-temperature (OT–T) trip function and previous power range neutron flux high trip setpoint. In addition, a rapid rise in power to greater than 50 percent RTP with AFD outside limits does not immediately create an unacceptable situation. The increased potential for a reactor trip caused by the manual manipulation of the setpoint needlessly exposes the plant to an unnecessary trip with the potential for an undesirable plant transient which may unnecessarily challenge safety systems.

Therefore, the proposed aforementioned TS changes do not involve a signification reduction in a margin of safety.

Based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: February 1, 2000, as supplemented by letter dated April 13, 2000.

Description of amendment request: The proposed amendment proposes changes to the cable spreading room technical specifications to permit pressurizing the cable spreading room to a pressure that exceeds the pressure of the adjacent control room envelope area during testing.

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

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

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

The proposed Technical Specification and Bases changes to exclude the requirements of Surveillance Requirements (SRs) 4.7.7.e.2, 4.7.8.c.2, and 4.7.8.c.3 during pressurization testing of the Cable Spreading Room (CSR) will not increase the probability of an accident previously evaluated. Operation of the Control Room Emergency Air Filtration System and the Control Room Envelope Pressurization System cannot cause an accident to occur.

The proposed Technical Specification and Bases changes to exclude the requirements of SRs 4.7.7.e.2, 4.7.8.c.2, and 4.7.8.c.3 during pressurization testing of the CSR may adversely impact the consequences of previously evaluated accidents. During CSR pressurization testing, the Control Room Emergency Air Filtration and the Control Room Envelope Pressurization Systems may not be able to pressurize and maintain the Control Room envelope at a positive pressure with respect to adjacent areas and the outside atmosphere. As a result, radioactivity released from a design basis accident may enter the Control Room envelope. However, since the CSR area will actually be at a higher pressure than the outside atmosphere (during CSR pressurization testing), radioactive leakage into the CSR area, and subsequently into the Control Room envelope, should not occur after the temporary fan has been stopped. Administrative controls will be established to immediately stop the temporary fan and rapidly depressurize the CSR area in the event Control Building isolation is necessary. Once the CSR area is depressurized, the Control Room Emergency Air Filtration System and the Control Room Envelope Pressurization System will be able to function as designed to mitigate the consequences of the accident. In addition, the probability of a design basis accident (DBA) occurring while the CSR is pressurized is low. Therefore, exempting the requirements of SRs 4.7.7.e.2, 4.7.8.c.2, and 4.7.8.c.3 during CSR pressurization testing will not result in a significant increase in the consequences of an accident previously evaluated.

The proposed Technical Specification and Bases change to clarify the mode of operation of the Control Room Emergency Air Filtration System when the pressurization requirement of SR 4.7.7.e.2 applies, will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed change can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification and Bases changes to exclude the requirements of SRs 4.7.7.e.2, 4.7.8.c.2, and 4.7.8.c.3 during pressurization testing of the CSR, and to clarify the mode of operation of the Control Room Emergency Air Filtration System when the pressurization requirement of SR 4.7.7.e.2 applies, will not alter the plant configuration (no new or different type of permanent equipment will be installed) or require any new or unusual operator actions. Temporary equipment will be utilized to pressurize the CSR, and administrative controls, using additional personnel beyond the normal shift complement, will be implemented to restore the CSR to a configuration that will allow the Control Room Emergency Air Filtration System and the Control Room Envelope Pressurization System to function as designed to mitigate the consequences of an accident. The temporary equipment and administrative controls that will be implemented to perform the CSR pressurization testing will not introduce any new failure modes that could result in a new accident. Also, the response of the plant and the operators following these accidents is unaffected by the changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification and Bases changes to exclude the requirements of SRs 4.7.7.e.2, 4.7.8.c.2, and 4.7.8.c.3 during pressurization testing of the CSR may adversely impact the ability of the Control Room Emergency Air Filtration System and the Control Room Envelope Pressurization System to function as designed to protect the Control Room Operators following a DBA, and to use other accident mitigation equipment contained within the Control Room envelope. However, the administrative controls that will be established to immediately stop the temporary fan and rapidly depressurize the CSR area if Control Building isolation is necessary will provide reasonable assurance that the habitability of the Control Room envelope will be maintained. Therefore, exempting the requirements of SRs 4.7.7.e.2, 4.7.8.c.2, and 4.7.8.c.3 during CSR pressurization testing will not result in a significant reduction in a margin of safety.

The proposed Technical Specification and Bases change to clarify the mode of operation of the Control Room Emergency Air Filtration System when the pressurization requirement of SR 4.7.7.e.2 applies will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. Therefore, there will be no significant reduction in a margin of safety.

The proposed changes do not alter the design, function, or operation of the equipment involved. The impact of the proposed changes has been analyzed, and it has been determined they do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of a new or different kind of accident from any accident previously evaluated, and do not involve a significant reduction in a margin of safety. Therefore, NNECO has concluded the proposed changes do not involve an SHC.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 4, 2000.

Description of amendment request: The proposed amendment would add new sections to the Technical Specifications (TSs) addressing missed surveillance test requirements and establishing a TS Bases control program, revise TS Chapter 6 to allow use of generic titles in lieu of plant-specific titles, allow an alternative when the radiation protection manager does not meet the qualifications of Regulatory Guide 1.8, relocate sections of TS Chapter 6 pertaining to onsite and offsite review and special inspections to the Operational Quality Assurance Plan, and correct typographical errors.

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

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

Operation of the Monticello plant in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. None of the proposed changes involves a physical modification to the plant, a new mode of operation or a change to the Updated Safety Analysis Report transient analysis. These proposed amendments conform to the guidance of NUREG–1433, Revision 1, which was previously issued by the NRC.

The proposed changes do not reduce the level of qualification or training and the aggregate knowledge of the plant staff remains intact. In total, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new mode of plant operation, surveillance test requirement or involve a physical modification to the plant. These proposed amendments generally conform to the guidance of NUREG–1433, Revision 1, which was previously issued by the NRC.

The proposed changes are administrative in nature. The changes propose to relocate specifications from the Technical Specifications to the Operational Quality Assurance Plan through which adequate control is maintained.

The proposed changes do not alter the design, function or operation of any plant component and therefore no new accident scenarios are created. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created by these amendments.

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

The proposed changes do not involve a significant reduction in a margin of safety because the current Technical Specification requirements for safe operation of the Monticello plant are maintained. The proposed changes are administrative in nature and do not involve a physical modification to the plant, a new mode of operation or a change to the Updated Safety Analysis Report transient analyses. The proposed changes do not alter the scope of equipment currently required to be operable or subject to surveillance testing nor does the proposed change affect any instrument setpoints or equipment safety functions.

Therefore, a significant reduction in the margin of safety would not be involved with these proposed changes.

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: May 3, 2000 (PCN–516).

Description of amendment requests: The amendment application proposes to revise the San Onofre Nuclear Generating Station, Units 2 and 3, Technical Specification (TS) 3.4.3, ''RCS Pressure and Temperature (P/T) Limits,'' and the associated Bases. The proposed change would reduce the minimum boltup temperature from 86 °F to 65 °F for the reactor coolant system during the period of time when the reactor vessel head bolts are in tension.

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:

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

Response: No.

This proposed change is a request to revise Technical Specification 3.4.3, ''Pressure Temperature Limits.'' The proposed change reduces the Minimum Boltup Temperature (MBT) from 86°F to 65°F. During operations below 86°F, the plant is in a shutdown mode, open to the atmosphere, and depressurized.

This proposed change does not affect the shape of the Pressure Temperature Limits when Reactor Coolant System (RCS) temperature is above 86°F. Therefore, the probability or consequences of an accident previously evaluated will not be increased by operating the facility in accordance with this proposed change.

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

Response: No.

This proposed change does not change the design or configuration of the plant. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any accident that has been previously evaluated.

(3) Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

This proposed change involves reducing the MBT from 86°F to 65°F. This proposed change meets the American Society of Mechanical Engineers (ASME) Code requirements for establishing the minimum temperature in the reactor pressure vessel flange region when the pressure does not exceed 20% of the pre-operational hydrostatic test pressure. All margins of safety established by the ASME Code requirements are maintained. The operation of the facility in accordance with this proposed 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 requests involve no significant hazards consideration.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: April 10, 2000 (TS 99–013).

Description of amendment request: The proposed amendment requests NRC's approval to use the F* alternate repair criterion in the tubesheet region of the steam generator (SG). The F* criterion addresses the action required when degradation has been detected in the top of the mechanically expanded portion of SG tubes within the SG tubesheet.

The proposed change designates a portion of the tube for which tube degradation of a defined type does not necessitate remedial action, except as dictated for compliance with tube leakage limits as set forth in the WBN Technical Specifications (TS). The proposed amendment would modify the TS which provide tube inspection requirements and acceptance criteria to determine the level of degradation for which the tube may remain in service. The proposed amendment would add definitions required for the F* alternate plugging criterion and prescribe the portion of the tube subject to the criterion.

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

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

The presence of the tubesheet enhances the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. Hardrolling of the tube into the tubesheet results in an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated loss-of-coolant-accident (LOCA) loadings.

The type of degradation for which the alternate plugging criterion, F*, has been developed (cracking with a circumferential orientation) can theoretically lead to a postulated tube rupture event, provided that the postulated through-wall circumferential crack exists near the top of the tubesheet. An evaluation including analysis and testing has been performed to determine the resistive strength of roll expanded tubes within the tubesheet. That evaluation provides the basis for the acceptance criteria for tube degradation subject to the F* criterion.

The F* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The existing technical specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, tube rupture and pullout are not expected for tubes using the F* alternate plugging criterion. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the WBN Unit 1 Final Safety Analysis Report (FSAR). The proposed alternate plugging criterion (F*) does not adversely impact any other previously evaluated design basis accident.

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

Implementation of the proposed F* tubesheet alternate plugging criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to result in an accident initiated outside of the region of the tubesheet expansion. A hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity and leaktightness are expected to be maintained. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The use of the tubesheet alternate plugging criterion has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Regulatory Guide 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' for indications in the free span of tubes and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the F* criterion is any degradation indication in the tubesheet region, more than the F* distance below either the bottom of the transition between the roll expansion and the unexpanded tube, or the top of the tubesheet, whichever is lower. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel Code used in steam generator design. The F* distance has been verified by testing to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inside diameter-based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the technical specifications and the leakage assumptions used in the FSAR accident analyses.

Implementation of the alternate tubesheet plugging criterion will decrease the number of tubes which must be taken out of service with tube plugs or repaired with sleeves. Both plugs and sleeves reduce the RCS flow margin; thus, implementation of the F* alternate plugging criterion will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving. Based on the above, it is concluded that the proposed change does not result in a significant reduction in a loss of margin with respect to plant safety as defined in the FSAR or the bases of the WBN technical specifications.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: April 10, 2000 (TS 99–014).

Description of amendment request: The proposed amendment would revise the WBN Unit 1 Technical Specification (TS) to incorporate new requirements associated with steam generator (SG) tube inspection and repair. The new requirements establish an alternate voltage based SG tube repair criteria at tube support plate and Flow Distribution Baffle plate intersections. This change is consistent with NRC Generic Letter 95–05 ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking.''

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.

Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate. Test data indicates that tube burst cannot occur within the tube support plate (TSP), even for tubes which have 100 percent through-wall electric discharge machining (EDM) notches, 0.75 inches long, provided that the TSP is adjacent to the notched area. Since tube to tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition [main steam line break (MSLB)] differential pressure of 2405 psig. As previously stated, the Regulatory Guide (RG) 1.121 criterion requiring maintenance of a safety factor of 1.43 times the MSLB pressure differential on tube burst is satisfied by ¾-inch diameter tubing with bobbin coil indications with signal amplitudes less than V_{SL} = 6.03 volts, regardless of the indicated depth measurement. At the flow distribution baffle (FDB), a safety factor of 3 against the normal operating condition DP is applied; here a voltage of V_{SL} = 3.81 volts satisfies the burst capability recommendation.

The upper voltage repair limit (V_{URL}) will be determined prior to each outage using the most recently approved NRC database to determine the tube structural limit (V_{SL}). The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{NDE}) and growth (V_G) to establish V_{URL}. As an example, the NDE uncertainty component of 20 percent and a voltage growth allowance of 30 percent per full power year can be utilized to establish a V_{URL} of 3.71 volts for TSP indications, and 2.34 volts for the FDB indications. The 20 percent NDE uncertainty represents a square-root-sum-of-the-squares (SRSS) combination of probe wear uncertainty and analyst variability. The flaw growth allowance should be an average growth rate or 30 percent per effective full power year, whichever is larger. The 30 percent growth allowance used to determine V_{URL} is conservative for the current conditions at WBN Unit 1. The most current NRC approved database, contained in EPRI [Electric Power Research Institute] NP–7480–L, Addendum 2, was used to establish the V_{URL} values for the FDB and TSP intersections. Once approved by NRC, the industry protocol for updating the database will be followed by TVA, ensuring that the most current database is utilized for future applications of the criteria.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated MSLB outside of containment but upstream of the main steam isolation valves (MSIV) represents the most limiting radiological condition relative to the alternate voltage based repair criteria. In support of implementation of the revised repair limit, it will be determined whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary to secondary leakage would result in site boundary doses within a fraction of the 10 CFR 100 guidelines or control room doses within the 10 CFR 50, Appendix A, General Design Criteria (GDC)–19 limit. A separate calculation has determined this allowable MSLB leakage limit to be 10 gallons per minute (gpm) in the faulted loop assuming a reactor coolant system (RCS) dose equivalent iodine concentration of 1.0 mCi/gm. The establishment of the 10 gpm leak rate value is controlled by the 0 to 2 hour offsite doses at the site boundary for the accident initiated iodine spike case, not control room dose.

The methods for calculating the radiological dose consequences for this MSLB are consistent with FSAR Chapter 15 and therefore, the WBN licensing basis. TVA bases the calculated thyroid dose consequences on conversion factors from the International Commission on Radiation Protection (ICRP) Publication 2.

In summary, the calculated radiological consequences of the exclusion area boundary and the low population zone are larger than previously reported for the postulated steamline break event due to the increased leakage and more conservative iodine spiking factors. However, the calculated radiological consequences remain in compliance with the guidelines in the Standard Review Plan, Chapter 15, 10 CFR 50, Appendix A, GDC–19 and 10 CFR 100 reported for the postulated steamline break event. Therefore, it is concluded that the proposed changes do not result in a significant increase in the radiological consequences of an accident previously analyzed.

Consistent with the guidance of Section 2.c of Generic Letter (GL) 95–05, the WBN Unit 1 MSLB leak rate analysis performed prior to returning the SGs to service may be performed based on the projected next end-of-cycle (EOC) voltage distribution or the actual measured distribution at a given outage. The method to be used for the first outage when outside diameter stress corrosion cracking (ODSCC) indication growth rates are available will be based on the indications found during that outage. As noted in GL 95–05, it may not always be practical to complete EOC calculations prior to returning the SGs to service. Under these circumstances, it is acceptable to use the actual measured bobbin voltage distribution instead of the projected EOC voltage distribution to determine whether the reporting criteria is being satisfied.

Therefore, as implementation of the 1.0 volt voltage-based repair criteria at WBN Unit 1 does not adversely affect steam generator tube integrity and implementation is shown to result in acceptable radiological dose consequences, the proposed TS change does not result in a significant increase in the probability or consequences of an accident previously evaluated within the WBN Final Safety Analysis Report (FSAR).

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

Implementation of the proposed steam generator tube alternate voltage based repair criteria (1.0 volts) does not introduce any significant changes to the plant design basis. Neither a single or multiple tube rupture event would be expected in a steam generator in which the repair limit has been applied (during all plant conditions).

The bobbin probe voltage-based tube repair criteria of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100 percent eddy current inspection sample size at the tube support plate elevations, and rotating pancake coil (RPC) inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

TVA will implement a maximum normal operating condition primary to secondary leakage rate limit of 600 gallons per day (gpd) total primary to secondary leakage and 150 gpd primary to secondary leakage per steam generator to help preclude the potential for excessive leakage during all plant conditions. The 150 gpd leakage limit is more restrictive than the current TS operating leakage limit (of 500 gpd) and is intended to provide additional margin to accommodate a stress corrosion crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Leakage trending capability consistent with EPRI Report TR–104788, ''Primary-to-Secondary Leak Guidelines'' has been implemented at WBN Unit 1.

As steam generator tube integrity upon implementation of the 1.0 volt repair limit continues to be maintained through in-service inspection and primary to secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The use of the voltage-based bobbin probe tube support plate elevation repair criteria at WBN Unit 1 maintains steam generator tube integrity commensurate with the criteria of Regulatory Guide (RG) 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by in-service inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the proposed criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the tube support plate elevations is confirmed to result in acceptable primary to secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

As a preventative measure, a total of 214 tubes are excluded from the application of the ODSCC criteria because of the combined effects of loss-of-coolant-accident (LOCA) + safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2). It was determined that tube deformation or through-wall cracks could occur in these tubes.

As noted previously, implementation of the tube support plate intersection voltage-based repair criteria will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, implementation of the 1.0 volt repair limit will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

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

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

NRC Section Chief: Richard P. Correia.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of application for amendments: November 23, 1999.

Brief description of amendments: The amendments changed Technical Specification 5.5.7.c.1, ''Ventilation Filter Testing.'' The testing criteria have been changed to be consistent with the NRC request in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of issuance: May 16, 2000.

Effective date: May 16, 2000.

Amendment Nos.: 209 and 237.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73086) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 16, 2000.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 22, 1999, as supplemented on March 1, 2000.

Brief description of amendments: The amendments relocate Reactor Coolant System (RCS) related cycle-specific parameter limits from the technical specifications to, and thus expanding, the Core Operating Limits Reports (COLRs) for Byron, Units 1 and 2, and Braidwood, Units 1 and 2.

Date of issuance: May 15, 2000.

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

Amendment Nos.: 113 and 106.

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

*Date of initial notice in **Federal Register:*** February 23, 2000 (65 FR 9003). The March 1, 2000, 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 May 15, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 14, 1999, as supplemented on January 21, February 15, February 23, March 10, March 24, March 31 (two letters), April 7 and April 14, 2000.

Brief description of amendments: The amendments increase the maximum reactor core power level to 3489 megawatts thermal; an increase of 5 percent of rated core thermal power, for

LaSalle County Station, Units 1 and 2. In addition, the proposed amendments correct a non-conservative value in the upper limit for drywell and suppression chamber internal pressure that was discovered during the course of the power uprate review.

Date of issuance: May 9, 2000.

Effective date: For Unit 1, immediately, to be implemented within 60 days; for Unit 2, immediately, to be implemented prior to start up of cycle 9.

Amendment Nos.: 140 and 125.

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

*Date of initial notice in **Federal Register:*** August 25, 1999 (64 FR 46427). The letters dated January 21, February 15, February 23, March 10, March 24, two letters on March 31, April 7, and April 14, 2000, contain supplemental, clarifying information that did not change the staff's initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket No. 50–374, LaSalle County Station, Unit 2, LaSalle County, Illinois

Date of application for amendment: February 28, 2000, as supplemented on April 28, 2000

Brief description of amendment: The amendment increases the Technical Specification safety limit for the Minimum Critical Power Ratio from 1.08 for two loop operation and 1.09 for single loop operation to 1.11 and 1.12, respectively.

Date of issuance: May 17, 2000.

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

Amendment No.: 126.

Facility Operating License No. NPF–18: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 22, 2000 (65 FR 15377). The April 28, 2000, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 4, 1999, as supplemented by letter dated April 19, 2000.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Limiting Conditions for Operation for Reactor Coolant System (RCS) Subcooling Margin Monitor in TS Table 3.3.3–1 and revise the functions associated with surveillance requirements for RCS Loops-Test Exceptions in TS 3.4.17. By letter dated April 19, 2000, the licensee withdrew the proposal to relocate the Auxiliary Feedwater Loss of Offsite Power function from TS 3.3.2–1 to TS 3.3.2–1. The other changes requested by August 4,1999, application were addressed under separate correspondence.

Date of issuance: May 19, 2000.

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

Amendment Nos.: 186 and 179.

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

*Date of initial notice in **Federal Register:*** September 8, 1999 (64 FR 48861)

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

No significant hazards consideration comments received: No

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

Date of amendment request: August 18, 1999, as supplemented by letter dated April 20, 2000.

Brief description of amendment: The requested change would revise Technical Specification 3.5.3, ''Safety Feature Actuation System Setpoints,'' and its associated Bases to allow for an increase to the low reactor coolant system pressure setpoint. This setpoint change was requested to account for additional instrument uncertainties associated with cable insulation resistance effects and to allow for the plugging of up to 1200 tubes in each steam generator.

Date of issuance: May 10, 2000.

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

Amendment No.: 207.

Facility Operating License No. DPR–51: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4270). The April 20, 2000, letter provided clarifying information that did not change the scope of the August 18, 1999, application and the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: July 15, 1999, as supplemented by letters dated March 29, 2000, April 13, 2000, April 25, 2000, and May 9, 2000.

Brief description of amendment: The amendment changed the Technical Specifications to institute a Technical Specification Bases Control Program and to provide for record retention as specified in the Quality Assurance Program Manual.

Date of issuance: May 9, 2000.

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

Amendment No.: 161

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4274). The supplements dated March 29, 2000, April 13, 2000, April 25, 2000, and May 9, 2000, did not change the scope of the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 2000. No significant hazards consideration comments received: No.

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

Date of amendment request: July 19, 1999.

Brief description of amendment: The proposed change modifies Technical Specification 4.5.2.f.2 by increasing the performance requirement for the low pressure safety injection pumps.

Date of issuance: May 10, 2000.

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

Amendment No. 162.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4277).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: July 29, 1999.

Brief description of amendment: The amendment changed the Technical Specifications (TS) to extend the allowable outage time to seven days for one containment spray system (CSS) train inoperable. A new ACTION has been added to provide a shutdown requirement for the inoperability of two CSSs. The associated changes to TS Bases are included. However, the licensee requested MODE 4 end state for TS 3.6.2.1 is being deferred.

Date of issuance: May 15, 2000.

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

Amendment No.: 163.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 9, 2000 (65 FR 6406). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 22, 1999, as supplemented January 3, 2000.

Brief description of amendments: The amendments remove the recirculation system motor generator set stop surveillance requirement from the LGS Units 1 and 2 Technical Specifications.

Date of issuance: May 8, 2000.

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

Amendment Nos.: 142 and 104.

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

*Date of initial notice in **Federal Register:*** September 8, 1999 (64 FR 48864). The January 3, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** Notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 8, 2000.

No significant hazards consideration comments received: No

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing

(Exigent Public Announcement or Emergency Circumstances)

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By June 14, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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

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

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

Date of amendment request: May 8, 2000.

Description of amendment request: The amendment revises Technical Specification Surveillance Requirement 3.8.1.9 to increase the limit for the peak transient voltage measured following a full-load rejection by the emergency diesel generator that is being tested.

Date of issuance: May 9, 2000.

Effective date: As of its date of issuance and shall be implemented within 2 days.

Amendment No.: 140.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 9, 2000.

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

NRC Section Chief: Claudia M. Craig.

Dated at Rockville, Maryland, this 24th day of May 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–13518 Filed 5–30–00; 8:45 am]

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OCCUPATIONAL SAFETY AND HEALTH REVIEW COMMISSION

Agency Information Collection Activities: Proposed Collection; Comment Request; Evaluation of the ''E–Z Trial''

ACTION: Notice.

SUMMARY: As part of its effort to reduce paperwork and the burden placed on survey recipients, the Occupational Safety and Health Review Commission (OSHRC) is conducting a preclearance consultation to provide the general public and Federal agencies with an opportunity to comment on a proposed collection of information in accordance with the Paperwork Reduction Act of 1965, Public Law 104–13. OSHRC is soliciting comment concerning an information collection required to evaluate the Review Commission's ''E–Z–Trial'' program.