Dated at Rockville, Maryland, this 31st day of December 1998.

For the Nuclear Regulatory Commission.

William F. Kane,

Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

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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 December 18, 1998, through December 31, 1998. The last biweekly notice was published on December 30, 1998 (63 FR 71962).

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

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

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

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

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

By February 12, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendments request: November 30, 1998.

Description of amendments request: Currently, the Calvert Cliffs Technical Specifications allow defective tubes to be plugged and removed from service, or to be repaired by either the laser-welded sleeving technique developed by Westinghouse Electric Corporation or by using leak-tight, tungsten inert gas-welded sleeving developed by Combustion Engineering, Inc. (ABB–CE). The proposed amendment will revise the appropriate Technical Specifications to permit the use of leak-limiting Alloy 800 repair sleeves developed by ABB–CE to be used at Calvert Cliffs. Combustion Engineering provides two types of leak-limiting Alloy 800 repair sleeves. The first type of repair sleeve spans the expansion transition zone of the tube at the top of the tubesheet. The second type of repair sleeve spans the degraded areas at an eggcrate support elevation or in a free span section.

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

The ABB CE Alloy 800 leak-limiting repair sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applied stresses and fatigue usage for the repair sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Regulatory Guide 1.121. Burst testing of sleeved tubes has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

The Alloy 800 repair sleeve Technical Specification depth-based plugging limit is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of ASME Code, Section III. A bounding tube wall degradation growth rate per cycle and a nondestructive examination uncertainty has been assumed for determining the repair sleeve plugging limit.

Evaluation of the repaired steam generator tubes indicates no detrimental effects on the sleeve or sleeve-tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Calvert Cliffs. Corrosion testing of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant, or the manner in which it is operated. The consequences of a hypothetical failure of the sleeved tube is bounded by the current steam generator tube rupture analysis described in Calvert Cliffs Updated Final Safety Analysis Report, Section 14.15. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis and, therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feed line break will not cause a SGTR [steam generator tube rupture] since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the Calvert Cliffs safety analysis. The minimal repair sleeve leakage that could occur during plant operation is well within the Technical Specification leakage limits.

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

2. Would not create the possibility of a new or different kind of accident from any other accident previously evaluated.

As discussed above, the Alloy 800 repair sleeves are designed using the applicable ASME Code as guidance; therefore, it meets the objectives of the original steam generator tubing. As a result, the functions of the steam generators will not be significantly affected by the installation of the proposed sleeve. The proposed repair sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. The continued integrity of the installed sleeve is periodically verified by the Technical Specification requirements.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant, or the manner in which it is operated. Therefore, BGE [Baltimore Gas and Electric Company] concludes that this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The repair of degraded steam generator tubes with Alloy 80 leak-limiting repair sleeves restores the structural integrity of the degraded tube under normal operating and postulated-accident conditions. The design safety factors utilized for the repair sleeves are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original steam generator design. The portions of the installed sleeve assembly that represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation. Use of the previously identified design criteria and design verification testing assures that the margin to safety is not significantly different from the original steam generator tubes.

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

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

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

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

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: November 9, 1998.

Description of amendment request: The proposed amendments would revise Technical Specification Table 3.3.3–2, ''Emergency Core Cooling System Actuation Instrumentation Setpoints'' to modify the degraded voltage second level undervoltage relay setpoint and allowable value. This change was submitted in response to a concern identified during an Electrical Distribution System Functional Inspection.

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

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

The setpoint change does not change the logic or function of the degraded voltage protection circuits as described in UFSAR [Updated Final Safety Analysis Report] Section 8.2.3. They also do not reduce the reliability of these circuits. The increase in the degraded voltage protection circuit setpoint is conservative compared to the existing setpoint. There is no change as a result of this amendment to the underlying accident and transient analyses that support operations of LaSalle County Station. Inadvertent or spurious operation of the degraded voltage protection function will initiate loading of the safe shutdown loads on the diesel generators and is not assumed to initiate an accident. The proposed degraded voltage setpoints are low enough to prevent spurious actuations given the expected offsite grid voltages. After implementation of this amendment, no operator actions are required for equipment operations in response to degraded voltage conditions.

This change does not affect the initiators or precursors of any accident previously evaluated. This change will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure.

The consequences of accidents previously evaluated are not increased. The proposed change does not affect the required level of availability of systems required to mitigate the accidents considered in the analyses. The proposed changes will ensure that the Class 1E equipment will be capable of starting and operating during a design basis accident with degraded offsite grid voltage. The increase in the level of confidence is the result of more rigorous methodology used to determine limiting Class 1E bus voltages at the minimum expected offsite AC voltage. These calculations demonstrate that the degraded voltage relays will not actuate following a block start of the electrical loads that are automatically actuated by or as a consequence of the LOCA [loss-of-coolant accident] signal if the switchyard voltage remains above 352 kV.

If the grid voltage drops below 352 kV, then the analytical limit of 3814 volts for proper operation of class 1E loads connected to each 4.16 kV Class 1E bus is assured by transfer to the respective onsite power sources (Emergency Diesel Generators (EDGs)) by the degraded voltage logic.

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

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

Setpoint methodology established the bases to ensure that, with known errors, the relays will detect degraded voltage conditions and transfer safety loads to the EDGs at a voltage level adequate to ensure proper safety equipment performance and to prevent equipment damage.

The greater than or equal to 3870 volt setpoint and the greater than or equal to 3814 volt allowable value includes adequate tolerance to calibrate the relay trip units while ensuring that the Class 1E bus voltage will remain above the analytical limits.

These setpoint changes will ensure that adequate voltages will be available for the continuous operation of safety-related equipment required to function during a LOCA. These proposed changes will also ensure that adequate voltages will be available for starting any Class 1E equipment.

The proposed degraded voltage setpoint change does not change the design of the degraded voltage protection system or its function to protect against degraded offsite power. Actuation of the degraded voltage protection system will initiate a sequence of events that will start the EDG for the associated Class 1E bus, strip loads from the Class 1E bus, open all feed breakers to the Class 1E bus, close the Emergency feed breaker (thus energizing the Class 1E bus from the respective EDG), and initiate starting of the Safe Shutdown equipment supplied by the Class 1E bus.

Since the scope of this change does not affect the operation of the auxiliary power system or any actions necessary to mitigate the consequences of accidents or achieve safe shutdown, the change does not involve a new or different accident scenario.

Therefore, these 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 the margin of safety because:

The proposed amendment will allow the degraded voltage setpoint to be conservatively established based on new engineering calculations which consider the lowest expected offsite grid voltage and operation of required Class 1E equipment under design basis accident loading conditions.

The proposed degraded voltage setpoints will ensure that adequate Class 1E bus voltage will be available to support starting and operation of the required Class 1E loads. The proposed setpoint includes instrument error to ensure that the lowest possible voltage will not be lower than the degraded voltage analytical limits. Additionally, the proposed setpoints are low enough to prevent spurious actuations due to expected fluctuations in the grid voltage. The new setpoints are also set with margin to the minimum Class 1E bus voltage, which is based on a minimum grid voltage of 352 kV, which is less than the expected grid voltage of 354 kV. The proposed changes will provide an increase in the level of protection that currently exists and will ensure the margin of safety is adequately maintained.

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

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

Local Public Document Room location: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: November 30, 1998.

Description of amendment request: The Safe Shutdown Makeup Pump (SSMP) allowed outage time (AOT) is being decreased from 67 days to 14 days.

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

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

The change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change to the Technical Specification Allowed Outage Time is conservative with respect to current requirements. This change is being proposed to establish an AOT for the SSMP that is equivalent to that for the reactor core isolation cooling (RCIC) pump (14 day AOT) in order to enhance system performance by assuring maximum SSMP pump availability to a level consistent with RCIC. This is necessary since, pursuant to Paragraph III.G.3 of 10 CFR 50, Appendix R, the SSMP is an alternate system to the RCIC system. By ensuring equipment availability, the probability or consequences of an accident previously evaluated are not increased. In addition, the proposed change has no impact on any accident initiators or initial condition assumptions for accident scenarios. Onsite or offsite dose consequences resulting from an event previously evaluated are not affected by this proposed amendment request.

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

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

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed license amendment provides a reduction to a Technical Specification Allowed Outage Time to enhance system performance by assuring maximum SSMP pump availability to a level consistent with RCIC. The proposed change is conservative with respect to the current requirements. The proposed amendment does not involve any plant physical changes that would create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The proposed change does not involve a significant reduction in a margin of safety. The proposed change enhances system performance by assuring maximum SSMP pump availability to a level consistent with RCIC. Since this is a conservative change that will enhance the performance of the SSMP system, it does not involve a significant reduction in the margin of safety.

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

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

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

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: October 30, 1998.

Description of amendment request: The proposed amendment would change the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS) to delete a note regarding the number of required channels for the Degrees of Subcooling function, and to subdivide the Core Exit Temperature (Backup) function into two new functions in ITS Table 3.3.17–1, Post-Accident Monitoring Instrumentation.

These proposed ITS changes support modifications scheduled for Refueling Outage 11 at CR–3. These modifications are intended to significantly improve the reliability and availability of information to the control room operators for verifying adequate core cooling is maintained following a design basis accident. The proposed ITS change deletes the note describing the use of the SPDS as a backup since the SPDS will be the primary indication of subcooling margin after the planned modifications are implemented.

The planned modifications will separate the sixteen core exit thermocouples into two separate channels of eight core exit thermocouples each. Following the modifications, there will be two core exit thermocouples per channel located in each core quadrant. Each separate channel of eight core exit thermocouples will have an associated core exit temperature recorder on the main control board, instead of the current three recorders, and will provide input into the associated channel of SPDS for calculation of subcooling margin.

The proposed ITS change will subdivide the current Core Exit Temperature (Backup) function into two new functions, Core Exit Temperature (Thermocouple) function and Core Exit Temperature (Recorder) function. For the Core Exit Temperature (Thermocouple) function, the proposed ITS will require at least two OPERABLE core exit thermocouples per core quadrant (at least one per channel) to provide a representative distribution of temperatures across the core to the operator. For the Core Exit Temperature (Recorder) function, both core exit temperature recorders will be required OPERABLE.

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

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

The proposed changes to the Degrees of Subcooling, Core Exit Temperature (Thermocouple), and Core Exit Temperature (Recorder) functions in the CR–3 Improved Technical Specifications (ITS) ensure appropriate post-accident monitoring instrumentation is available for use by the operators during implementation of emergency operating procedures. These emergency operating procedures provide direction to the operators for ensuring that actions required to mitigate the effects of the previously evaluated design basis accidents are performed. The instrumentation is used for monitoring by the operators after an accident occurs, perform no automatic functions, and there are no credible failures of this instrumentation which could initiate any accident previously evaluated. Therefore, the probability of occurrence of any accident previously evaluated is unaffected.

The availability and use of this instrumentation ensures that the prescribed manual operator actions for mitigating the consequences of an accident will be implemented when necessary, and that the operator has sufficient information to verify required automatic actions have occurred when necessary. The availability and use of the instrumentation provides assurance that the consequences of accidents will not be greater than that previously evaluated. The associated modifications that are planned for these post-accident monitoring instruments will enhance the reliability of the required indications to the operators.

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

The proposed changes to this post-accident monitoring instrumentation will ensure appropriate instrumentation is available for use by the operators following a design basis accident. This instrumentation is necessary for performing certain manual actions, or to verify automatic actions have occurred, which are required to mitigate the effects of a design basis accident. The instrumentation is used for monitoring by the operators after an accident occurs, perform no automatic functions, and there are no credible failures of this instrumentation which could initiate a new or different kind of accident. Therefore, the possibility of a new or different kind of accident occurring as a result of this passive instrumentation is not created.

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

The proposed changes to this post-accident monitoring instrumentation provide additional assurance that adequate instrumentation is available for use by the operators to perform manual actions, and to verify that automatic actions that are required to mitigate the effects of a design basis accident have occurred. The instrumentation is used for monitoring by the operators after an accident occurs, and perform no automatic functions. The availability and use of this instrumentation ensures that the prescribed manual operator actions for mitigating the consequences of an accident will be implemented when necessary, and that the operators have sufficient information to verify required automatic actions have occurred when necessary. These required manual and automatic actions are necessary to preserve the margin of safety as defined in the CR–3 ITS. The availability and use of this instrumentation provides assurance that the existing margin of safety will be maintained, and assumptions related to the margin of safety during mitigation of design basis accidents will be preserved. Therefore, the existing margin of safety will not be reduced.

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

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

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC—A5A, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: November 24, 1998.

Description of amendment request: The proposed amendment would change the CR–3 Improved Technical Specifications (ITS) in support of a modification to install a diesel-driven emergency feedwater (EFW) pump (EFP–3) which is intended to resolve capacity limitations of the CR–3 1A Emergency Diesel Generator (EGDG). The licensee has determined that installation of EFP–3 involves an unreviewed safety question and also requires changes and additions to the ITS and Bases.

EFP–3 will be installed as a functional replacement for EFP–1, the motor-driven EFW pump. EFP–3 will start and provide controlled and monitored EFW flow to both steam generators through the same EFW block and control valves as EFP–1 currently uses. The licensee stated that removing the auto-start logic from EFP–1 would eliminate the need to perform EGDG–1A load management to accommodate emergency safeguards (ES) loads required to mitigate design basis accidents. EFP–1 will remain available as a manually started pump. The installation of EFP–3 will also permit other changes in system operation which are intended to reduce reliance on operator actions to perform EGDG load management.

The proposed ITS and Bases changes fall into two categories: (1) new or revised ITS and Bases to account for equipment changes associated with the new EFP–3, and (2) those ITS and Bases requirements being deleted because they were approved until Cycle 12 only.

The new ITS requirements and revisions (category 1 changes) involve revised surveillance requirements (SR) and Bases for EFP–3 (SR 3.7.5), and new ITS and Bases for the diesel fuel oil supply, lube oil and starting air for EFP–3 (3.7.19). Also, the option to use EFP–3 for Once Through Steam Generator (OTSG) cooling is added to the Bases for 3.4.6, RCS Loops—MODE 5, Loops Filled. The Bases for 3.4.6, Background, lists all feedwater pumps that may be available in MODE 5. EFP–3 is added here for completeness.

The Bases for 3.7.5 are revised to describe the new EFP–3 and the new role for EFP–1 as a manual defense-in-depth pump. The Bases are also revised to indicate that EFP–3 cannot directly access the condenser hotwell. The phrase ''with the exception of the loss of all AC power (Ref. 3)'' is deleted from the Applicable Safety Analysis because with the addition of EFP–3, the EFW system is able to maintain its function on a loss of off-site power (LOOP) with a single failure.

In Section 3.7.5, EFW System, one SR is being revised and one new SR is being added. These changes are intended to provide SRs that demonstrate OPERABILITY of EFP–3 and essential subsystems. SR 3.7.5.1 and Bases are revised to add verification of proper valve position for starting air and fuel oil flow paths for EFP–3 on a 45-day frequency.

SR 3.7.5.6 is added to provide assurance that the DC electrical support system will be available to support OPERABILITY of EFP–3. This SR is based on a similar SR currently approved for the station DC system required by ITS 3.8.4, DC Sources—Operating. SR 3.7.5.6 was determined necessary because DC power is essential for starting EFP–3.

ITS 3.7.19 was added to ensure essential subsystems are within limits needed to maintain EFP–3 OPERABLE. The specification includes requirements for fuel oil, lube oil, and starting air. This specification has an allowed outage time (AOT) for these parameters if they are less than the limit but above a minimum value. Below the minimum allowed value, EFP–3 must be declared inoperable.

A number of ITS and Bases are being revised to remove the requirements that permitted operation of CR–3 until Cycle 12 only (category 2 changes). All text marked with the footnote ''Note—Valid until Cycle 12 only,'' and the note itself, is being deleted except for a few instances which are discussed in the licensee's submittal.

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.

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

This change involves the addition of a new safety-related Diesel-Driven Emergency Feedwater Pump (EFP–3). The Emergency Feedwater (EFW) System is not an initiator for any design basis accident except for those accidents associated with an increase in primary to secondary cooling and a loss of heat sink. The new EFP–3 functionally replaces the Motor-Driven Emergency Feedwater Pump (EFP–1) and is no more likely to cause an inadvertent cooldown than the existing EFP–1. The starting logic of the Emergency Feedwater Initiation and Control system is the same for EFP–3 as it was for EFP–1 before. No other control or logic changes are being made that would make EFP–3 more likely to cause a cooldown transient.

EFP–3 has a slightly greater probability of failing to start compared to EFP–1 with offsite power available. Therefore, there is a slight increase in the probability of an event that involves a loss of heat sink when considering only the Improved Technical Specifications (ITS) required EFW Pumps. The new EFP–3 will be highly reliable and therefore this increase in risk is not significant. Loss of EFP–3 alone does not cause a total loss of heat sink without the loss of the Turbine-Driven Emergency Feedwater Pump (EFP–2) and the remaining feedwater pumps. The most important of these feedwater pumps is EFP–1, which will be maintained as a safety-grade backup. EFP–3 is less reliable than EFP–1 with offsite power available. However, if offsite power is not lost, EFP–1 should be available for use. Therefore, the overall EFW system reliability is enhanced.

The consequences of the failure of EFP–3 to start or inadvertently actuate were considered. Failure of EFP–3 to start will have the same impact as failure of EFP–1. Therefore, the consequences of evaluated accidents are the same. EFP–3 will be designed to have minimum and maximum flows equivalent to EFP–1. No changes to the system will cause a decrease in the ability of the EFW system to remove heat from the Once Through Steam Generators (OTSGs). Similarly, the heat removal capability of EFP–3 will not be different than EFP–1. Therefore, there will not be the potential of a significantly greater overcooling event due to inadvertent start of EFP–3.

The license changes associated with the addition of EFP–3 remove a number of ITS Actions that established compensatory measures due to the possibility of overloading the Emergency Diesel Generators (EGDGs) and cross-train dependencies with EFP–2. These compensatory actions are no longer required. The changes to the EFW system eliminate EGDG limitations and reliance of the ''A'' train EFW pump on EFP–2. The revised ITS Actions ensure the equipment required to mitigate an accident is restored to OPERABLE status in accordance with previously approved limits. In addition, replacing required operator actions with automatic functions provides greater assurance that mitigating actions will occur. Therefore, these changes will not adversely affect the probability or consequences of evaluated accidents.

Based on the above, the addition of EFP–3 and the associated license changes do not involve a significant increase in the probability or consequences of a previously evaluated accident.

5. Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

EFP–3 performs the same functions as the existing EFP–1. No plant conditions are changed to cause new or different accidents. Although a diesel engine has different failure modes than a motor-driven pump, the consequences of a pump failure are the same. An interlock and administrative controls are provided to ensure that both EFP–3 and EFP–1 do not run at the same time. The interlock and administrative controls prevent any new interactive failure modes that could be caused by having both ''A'' train pumps (or all three EFW pumps) operating at the same time.

The revised ITS Actions ensure equipment is restored to OPERABLE status in accordance with previously approved timeframes. No new plant configurations or conditions are created by these Actions.

Therefore, these changes cannot create the possibility of an accident of a different type than previously evaluated in the SAR [Safety Analysis Report].

6. Does not involve a significant reduction in the margin of safety.

EFP–3 is designed to meet the same performance criteria as EFP–1. EFP–3 will replace EFP–1 in the ITS. The pump will perform the same functions, will be reliable and meet the same design criteria. There are no functions performed by EFP–1 that will be significantly different with EFP–3. The margin of safety provided by the specification relates to the ability to provide a heat sink. EFP–3 will provide the same margin of safety. In addition, EFP–1 will be available as a safety-grade backup and can deliver EFW to the OTSGs if offsite power is available or if the ''A'' train EGDG has adequate load margin.

The cooling capability of EFP–3 will be equivalent to EFP–1. Therefore, EFP–3 provides the same protection to the fuel cladding from temperature excursions as EFP–1. The EFP–3 modifications will be done without making penetrations through reactor coolant system (RCS) or containment boundaries. Therefore, the integrity of these fission product barriers remains unchanged.

The proposed changes to the ITS delete temporary restrictions placed on systems due to the potential to overload the EGDGs and cross-train dependencies with EFP–2. These compensatory actions are no longer required. The changes to the EFW system eliminate EGDG limitations and reliance of the ''A'' train EFW pump on EFP–2. The revised ITS Actions ensure the equipment required to mitigate an accident is restored to OPERABLE status in accordance with previously approved limits. In addition, replacing required operator actions with automatic functions provides greater assurance that mitigating actions will occur.

Based on the above evaluation, there is no reduction in the margin of safety associated with the proposed equipment, system and license changes.

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

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

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC—A5A, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: July 14, 1998.

Description of amendment request: The proposed amendment would amend the Technical Specifications to revise the liquid and gaseous release rates to reflect the replacement of the former 10 CFR 20.106 requirements with the existing 10 CFR 20.1302 requirements.

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 change does not:

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

The likelihood that an accident will occur is neither increased nor decreased by these Technical Specification changes. These Technical Specifications changes will not impact the function or method of operation of plant equipment. No systems, equipment, or components are affected by the proposed changes. The proposed revisions to the liquid and gaseous release rate limits will not result in any change or increase in the types or amounts of effluents other than that which has historically been deemed acceptable for release, nor will there be an increase in individual or cumulative occupational radiation exposures other than that which has historically been deemed acceptable. Therefore, the proposed changes to the Technical Specifications do not involve any increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes do not involve changes to the physical plant or operations. The proposed changes are administrative in nature and will not change the types and amounts of effluents from that which has historically been deemed acceptable. Since these administrative changes do not contribute to accident initiation, they do not produce a new accident scenario nor do they alter any existing accident scenarios. Therefore, the proposed changes to the Technical Specifications would 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 not reduce the margin of safety because compliance with the limits of the existing 10 CFR 20.1301 will be demonstrated by operating within the limits of 10 CFR Part 50, Appendix I and 40 CFR Part 190. For the liquid effluent releases the annual dose of 500 mrem, upon which the concentrations in the previous 10 CFR Part 20, Appendix B, Table II, Column 2, are based, is a factor of 10 higher than the annual dose of 50 mrem, upon which the concentrations in the existing 10 CFR 20, Appendix B, Table 2, Column 2, are based. Also, for gaseous effluent releases, the limits associated with the gaseous release Technical Specifications will be revised to the previously acceptable instantaneous dose rate limits.

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

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

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 321 Ferry Road, Wiscasset, ME 04578.

NRC Project Director: Seymour H. Weiss.

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

Date of amendment request: December 10, 1998.

Description of amendment request: The proposed amendment would allow NNECO to implement plant modifications that would ensure that proper flow paths can be established for boron precipitation control after a loss-of-coolant accident.

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 [Northeast Nuclear Energy Company] 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 plant modifications will ensure proper flow paths can be established for boron precipitation control after a Loss of Coolant Accident

(LOCA). This will be accomplished by the following plant modifications:

a. Provide an alternate AC source of power for 2–SI–651, ''Shutdown Cooling Header Containment Isolation Valve,'' a Facility Z1 component, from Facility Z2.

b. Provide an alternate DC source of power for 2–CH–517, ''Auxiliary Spray Charging Header Supply Valve,'' a Facility Z2 component, from Facility Z1.

c. Provide an alternate DC source of power for 2–CH–519, ''Loop 1A Charging Header Supply Valve,'' a Facility Z2 component, from Facility Z1.

d. Provide test jacks to determine valve position for LPSI [low-pressure safety injection] injection valves 2–SI–615, 2–SI–625, 2–SI–635, and 2–SI–645 at the respective motor control center (MCC [motor control center] B51 for 2–SI–615 and 2–SI–625 (MCC B61 for 2–SI–635 and 2–SI–645).

e. Provide bypass capability of the low pressure open permissive for 2–SI–651.

The alternate power supply to valves 2–SI–651, 2–CH–517 and 2–CH–519, and the position indication for valves 2–SI–615, 2–SI–625, 2–SI–635, and 2–SI–645 cannot initiate an accident. The proposed modifications will not change the design parameters, failure positions or design requirements of the valves. The proposed plant modifications will ensure valves 2–SI–651, 2–CH–517 and 2–CH–519 can operate after a LOCA to perform their accident mitigating functions. Therefore, providing an additional power source to 2–SI–651, 2–CH–517 and 2–CH–519, and a local means of determining the position of valves 2–SI–615, 2–SI–625, 2–SI–635, and 2–SI–645 cannot initiate an accident and will not adversely affect the function of these components to mitigate the consequences of an accident.

The proposed plant modifications will also bypass the open permissive for 2–SI–651. This pressure permissive, which protects the low pressure Shutdown Cooling (SDC) System from the high pressure Reactor Coolant System (RCS), allows 2–SI–651 to be opened only when pressurizer pressure is below 280 psia [pounds per square inch absolute]. This pressure permissive would be disabled upon a loss of Facility Z1 power. This would prevent the opening of 2–SI–651. The new local control switch for 2–SI–651, which bypasses this pressurizer pressure permissive, is isolated by normally open relay contacts. When aligned to its alternate power, local control is enabled, and remote control in the Main Control Room is isolated. Multiple operator errors would be required to align 2–SI–651 to the alternate power source during normal operation. To misalign these valves, an equipment operator would have to perform steps located only in an Emergency Operating Procedure. Additionally, control room operators would have to disregard annunciators that indicate the valves are being transferred to their alternate power source. Therefore, the only time the valve is expected to be opened by the local control switch is after a LOCA with a Facility Z1 failure. Currently, the potential exists for an operator to open the valve when pressure is above 280 psia. An undetectable single failure of the contact which provides the permissive would allow an operator to open the valve even when pressure is above 280 psia. During normal operation, this condition would be annunciated in the Main Control Room. During accident conditions, this annunciator may be disabled. Therefore, 2–SI–651 could be opened with pressurizer pressure above 280 psia without annunciation. Although 2–SI–651 could be opened, the pressure permissive for 2–SI–652, the upstream isolation valve (Attachment 1 Figure 1), would prevent 2–SI–652 from opening. This would protect the shutdown cooling suction line from overpressurization. During accident conditions, if both valves were opened and pressure increased above 280 psia, annunciation of 2–SI–652 being open would be available to provide indication of the potential overpressure condition. Therefore, the installation of the capability to bypass the open permissive for 2–SI–651 will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed plant modifications have no adverse effect on how any of the associated systems or components function to prevent or mitigate the consequences of design basis accidents. Also, the proposed changes have no adverse effect on any design basis accident previously evaluated since the modifications will ensure that accident mitigation equipment will be available to function as assumed in the LOCA analysis. Therefore, the proposed plant modifications do not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed plant modifications will provide the capability of powering 2–SI–651, 2–CH–517 and 2–CH–519 from either Facility Z1 or Facility Z2, and will add test jacks to MCC B51 and B61 to determine the position of valves 2–SI–615, 2–SI–625, 2–SI–635, and 2–SI–645. Additionally, this activity adds a local control switch which will bypass the open permissive for 2–SI–651 when aligned to the alternate power source. A single failure in any of the breakers or disconnect switches which allow 2–SI–651, 2–CH–517 and 2–CH–519 to be powered from either facility is bounded by the failure of the valve. A failure of any of the test jacks may result in a loss of control power to the associated valves. This failure is also bounded by the failure of the valve. During normal operation the local control switch which bypasses the pressure permissive is isolated by normally open contacts. A single failure of the local control switch or isolating relay during normal operation cannot disable the pressure permissive.

Since a single failure of any component added by this activity is bounded by existing component failures, a failure of these components cannot create a new accident. Therefore, the proposed plant modifications 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 plant modifications will ensure boron precipitation control can be established. This will be accomplished by providing an alternate power source for 2–SI–651, 2–CH–517 and 2–CH–519, and adding test jacks to determine the position of valves 2–SI–615, 2–SI–625, 2–SI–635, and 2–SI–645. Additionally, this activity adds a local control switch which will bypass the open permissive for 2–SI–651 when aligned to the alternate power source. Although the potential exists to route redundant power trains in the same cable trays, conduits and cable, the design of the modifications ensures that a single failure will not compromise the redundant power distribution system. The installation of the connection jacks and local control switch will not alter the failure analysis for the valves, and will not change the design parameters of the valves (i.e. pressure rating). Therefore, the proposed plant modifications will not compromise RCS pressure boundaries, containment integrity, or fuel cladding. In addition, the new disconnect switches, breakers, cabling, and auxiliary components are all designed for the rated voltages and currents, and are QA [quality assurance] Category I seismically and environmentally qualified, as required.

Based on the above, the proposed plant modifications will not reduce the integrity of the plant protective boundaries, or adversely affect the LOCA analysis. These modifications will have no adverse effect on equipment important to safety. The equipment will continue to function as assumed in the design basis accident analysis. This will ensure that the acceptance criteria of 10 CFR 50.46(b)(5) for long term core cooling will be met. Therefore, there will be no significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: December 16, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirements 4.8.1.1.2 and 4.8.1.1.3, Table 4.8.1.1.2–1, and the associated Bases. The proposed changes would remove the Emergency Diesel Generator accelerated testing and special reporting requirements from the TSs in accordance with the guidance provided in Generic Letter 94–01.

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 TS changes do not involve any physical changes to plant structures, systems, or components. [Public Service Electric & Gas Company] PSE&G has implemented the provisions of the Maintenance Rule for diesel generators, including the associated regulatory guidance, thereby establishing a program that assures diesel generator performance. The elements of the program include the performance of detailed root cause analysis of individual failures, effective corrective actions taken in response to individual failures, and implementation of preventive maintenance consistent with the Maintenance Rule. Monitoring the effectiveness of diesel generator maintenance and continuing surveillance testing in accordance with the proposed TS changes will ensure that the diesel generators will perform their intended functions and will minimize failures. The accelerated testing requirements are therefore no longer considered to be necessary and are deleted. The requirements of 10 CFR 50.72 and 10 CFR 50.73 ensure that diesel generator failures are properly reported. The special reporting requirements are therefore unnecessary and are deleted. Based on the above information, the changes will not adversely affect the assurance of diesel generator reliability or operability, and there is no significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed TS changes do not involve any physical changes to the design of plant systems, structures or components, nor do the changes involve a change in plant operation. The diesel generators will continue to function as designed to mitigate the consequences of an accident. Eliminating the accelerated testing requirements and special reporting requirements does not permit plant operation in a configuration that would create a different type of malfunction to the diesel generators than any previously evaluated. In addition, the proposed TS changes do not alter the conclusions described in the [Updated Final Safety Analysis Report] UFSAR regarding the safety related functions of the diesel generators or their support systems. No new failure modes will be introduced. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This request does not involve an adverse impact on diesel generator design, operation, or reliability. Since monitoring and maintenance is being performed in conformance with 10 CFR 50.65, modifying the surveillance testing frequency requirements does not adversely affect the reliability of the diesel generators. Deletion of the special reporting requirements does not impact operability or reliability of the diesel generator. Since the diesel generator function is not affected by the proposed change, this request does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

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

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

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

Date of application for amendments: December 3, 1998.

Brief description of amendments: The amendments would revise Technical Specification Section 4.6.5.1, ''Ice Condenser, Ice Bed,'' and the associated bases to reflect the maximum ice condenser flow channel blockage assumed in the accident analyses.

Date of publication of individual notice in **Federal Register**: December 28, 1998.

Expiration date of individual notice: January 27, 1999.

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

Notice of Issuance of Amendments To Facility Operating Licenses

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

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

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

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

Arizona Public Service Company, et al., Docket Nos. STN 50–528 and STN 50–529, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona

Date of application for amendment: October 6, 1998.

Brief description of amendment: The amendments revise Technical Specifications (TS) 3.3.1, ''Reactor Protective System (RPS) Instrumentation—Operating,'' and TS 3.3.2, ''Reactor Protective System (RPS) Instrumentation—Shutdown.'' The amendments clarify the power level threshold at which certain RPS instrumentation trips must be enabled and may be bypassed, and clarify that this level is a percentage of the neutron flux at rated thermal power (RTP). The bypass power level, 1E–4% RTP, is specified as logarithmic power instead of thermal power. The NRC approved these changes for Palo Verde Unit 3 on an exigent basis in its letter dated October 19, 1998. The exigent TS amendment resulted in TS pages with notes specifying different requirements between Unit 3 and Units 1 and 2. These amendments remove these notes regarding Unit 3 from the affected TS pages so that all Units now have the same TS.

Date of issuance: December 23, 1998.

Effective date: December 23, 1998.

Amendment No.: Unit 1–119; Unit 2–119.

Facility Operating License Nos. NPF–41 and NPF–51: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 4, 1998 (63 FR 59586).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004.

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

Date of application for amendment: February 11, 1998.

Brief description of amendment: The Updated Final Analysis Report (UFSAR) describes the response of the salt service water (SSW) system to a complete loss of AC power by assuming that the system would be divided by the closure of one of the two division isolation valves. Boston Edison Company (BECo) has discovered single failures involving a partial loss of AC power could place the SSW system in a configuration of one pump supplying both trains of heat exchangers for the first 10 minutes of the worst case design basis accident. BECo has determined that these single failures are an unreviewed safety question. The amendment authorizes BECo to change UFSAR Section 10.7, ''Salt Service Water System,'' to address this single fauilure vulnerability.

Date of issuance: December 21, 1998.

Effective date: December 21, 1998.

Amendment No.: 180.

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

Date of initial notice in **Federal Register**: April 8, 1998, (63 FR 17220)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 10, 1997, as supplemented May 23, 1997, and October 15, 1998.

Brief description of amendment: This amendment revises Technical Specification (TS) 3.5.1, ''Emergency Core Cooling System (ECCS) Accumulators,'' by (1) increasing the allowed outage time (from 1 hour to 72 hours) that one ECCS accumulator can be inoperable as a result of the boron concentration being outside of TS limits, and (2) modifying surveillance requirement 4.5.1 consistent with the guidance provided in NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements,'' December 1992, and the Standard Technical Specifications (STS) for Westinghouse Plants, NUREG-1431, Revision 1.

Date of issuance: December 31, 1998.

Effective date: December 31, 1998.

Amendment No.: 86.

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17226).

The May 23, 1997, and October 15, 1998, submittals contained clarifying information only, and did not change the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Consumers Energy Company, Docket No. 50—155, Big Rock Point (BRP) Plant, Charlevoix County, Michigan

Date of application for amendment: September 19, 1997.

Brief Description of amendment: This amendment changes the DPR—6 License and revises its Technical Specifications to reflect the permanently shutdown and defueled condition of the BRP plant.

Date of issuance: December 24, 1998.

Effective date: No later than 45 days from date of issuance.

Amendment No.: 120.

Facility Operating License No. DPR–6: The amendment revises the DPR—6 License and Appendix A Technical Specifications to the licensee.

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63974).

No significant hazards consideration comments received: No.

Local Public Document Room location: North Central Michigan College Library, 1515 Howard Street, Petoskey, MI 49770.

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: July 13, 1998.

Brief description of amendments: These amendments change the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and BVPS–2) Updated Final Safety Analysis Reports (UFSAR) descriptions of the Intake Structure main entrance and interconnecting cubicle doors. The changes approved by these amendments address a new failure mode of safety-related equipment that had not been previously considered for BVPS–1. The changes state that the cubicle interconnecting flood protection doors are normally closed with their inflatable seals depressurized and that the associated security/fire doors are normally closed. This door closure arrangement provides protection for the safety-related equipment in the interconnecting cubicles from the consequences of potential internal flooding.

Date of issuance: December 16, 1998.

Effective date: December 16, 1998.

Amendment Nos.: 218 and 96.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments approve changes to the Updated Final Safety Analysis Reports.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43202)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 16, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: B.F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: June 18, 1996, as supplemented September 8 and 30, 1998.

Brief description of amendment: The amendment (1) makes editorial changes to Technical Specification (TS) 4.4.5 and associated Bases; (2) revises the Bases for TS 3.4.6.2 to provide consistency with the Beaver Valley Power Station, Unit No. 1, Updated Final Safety Analysis Report (UFSAR); and (3) revises Index Page XVII to reflect the revision of page numbers due to shifting of text by License Amendment No. 198.

Date of issuance: December 21, 1998.

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

Amendment No: 219.

Facility Operating License No. DPR–66. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64109).

The September 8 and 30, 1998, letters did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the November 18, 1998, **Federal Register** notice; these letters only provided updated TS pages to be consistent with the UFSAR.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

Date of amendment request: May 31, 1996.

Brief description of amendment: The amendment revises the surveillance test interval for the reactor protection system reactor trip breakers, reactor trip modules, and electronic trip relays from a monthly interval to a quarterly interval.

Date of issuance: December 31, 1998.

Effective date: December 31, 1998.

Amendment No.: 194.

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

Date of initial notice in **Federal Register**: August 28, 1996 (61 FR 44356).

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

No significant hazards consideration comments received: No.

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

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

Date of amendment request: May 18, 1998, as supplemented on December 8, 1998.

Brief description of amendment: The amendment allows the use of trisodium phosphate stored in three baskets on the containment floor as a replacement to the sodium hydroxide addition system for the control of sump pH during long term core cooling in recirculation phase.

Date of issuance: December 23, 1998.

Effective date: The license amendment is effective as of its date of issuance to be implemented prior to the facility's restart from refueling outage 2R13.

Amendment No.: 194.

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

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56241)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 28, 1997.

Brief description of amendment: The amendment modifies the actions associated with Technical Specification (TS) Table 3.3–1 for the Reactor Protective Instrumentation and TS Table 3.3–3 for the Engineered Safety Feature Actuation System Instrumentation.

Date of issuance: December 29, 1998.

Effective date: December 29, 1998, to be implemented within 30 days.

Amendment No.: 195.

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

Date of initial notice in **Federal Register**: August 27, 1997 (62 FR 45456).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 29, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 29, 1998, as supplemented by letters dated December 17, 1998 and December 22, 1998.

Brief description of amendment: This amendment revises the as-found lift setting tolerance for the ANO–2 main steam safety valves and the pressurizer safety valves, revises the maximum allowable linear power level-high trip setpoint with inoperable steam line safety valves, and relocates part of the specifications for steam line safety valves to the ANO–2 Safety Analysis Report. Administrative and bases changes have also been made.

Date of issuance: December 31, 1998.

Effective date: The license amendment is effective as of its date of issuance to be implemented within 30 days.

Amendment No.: 197.

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

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56242).

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

No significant hazards consideration comments received: No.

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

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

Date of amendment request: August 12, 1998.

Brief description of amendment: The amendment changes the Technical Specifications by increasing the maximum boron concentration in the Safety Injection Tanks (SITs) and the Refueling Water Storage Pool (RWSP) from 2300 ppm to 2900 ppm.

Date of issuance: December 21, 1998.

Effective date: December 21, 1998, to be implemented within 60 days.

Amendment No.: 147.

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

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56249).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 11, 1996, and supplemented March 26, 1997.

Brief description of amendments: The amendments relocate certain quality assurance related requirements from the TS to the licensee's Quality Assurance Program Description.

Date of issuance: December 28, 1998.

Effective date: December 28, 1998, with full implementation within 120 days.

Amendment Nos.: 226 and 210.

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

Date of initial notice in **Federal Register**: July 31, 1996 (61 FR 40022).

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

No significant hazards consideration comments received: No.

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

Indiana Michigan Power Company, Docket No. 50–315 , Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: August 28, 1998, as supplemented November 4, 1998.

Brief description of amendment: The amendment grants relief from the steam generator surveillance requirement in Section 4.4.5.3 of the Technical Specifications (TS). The surveillance requirement is associated with non-destructive examination of the steam generator tubes which is required every 24 months. The relief allows the examination to be deferred from April 8, 1999, until the next refueling outage for D.C. Cook, Unit 1.

Date of issuance: December 30, 1998.

Effective date: December 30, 1998, with full implementation within 45 days.

Amendment No.: 227.

Facility Operating License No. DPR–58: Amendment adds paragraph 2.C.(9) to the License.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53950).

The November 4, 1998, submittal provided additional information that did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 30, 1997, as supplemented November 12, 1998.

Brief description of amendment: The amendment deletes TSs requirements associated with meterological monitoring instrumentation which have been relocated to the Updated Safety Analysis Report in accordance with 10 CFR 50.36 and the guidance in NRC Generic Letter 95–10, ''Relocation of Selected Technical Specification Requirements Related to Instrumentation.''

Date of issuance: December 22, 1998.

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

Amendment No.: 85.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: June 18, 1997 (62 FR 33126).

The November 12, 1998, letter 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 December 22, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents

Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

Date of application for amendment: September 19, 1995.

Brief description of amendment: The amendment reduces the frequency of the Technical Specification (TS) 4.5.1.d surveillance interval for boron concentration of the safety injection tasks from once per 31 days to once every 6 months. Initially, the change was requested for TS Section 4.5.1.b. However, TS Section 4.5.1.b was subsequently changed to TS Section 4.5.1.d by Amendment No. 220 to Facility Operating License No. DPR–65 dated September 3, 1998, in response to NNECO's application dated August 23, 1995.

Date of issuance: December 17, 1998.

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

Amendment No.: 221.

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

Date of initial notice in **Federal Register**: October 25, 1995 (60 FR 54722).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 17, 1998

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 2, 1998.

Brief description of amendment: The amendment revises the Updated Final Safety Analysis Report (UFSAR) by changing UFSAR Sections 9.7.2, ''Service Water,'' and 9.4, Reactor Building Closed Cooling Water,'' to include in the discussions the use of various types of internal protective coatings and liners used in the piping and components of the systems. The change also indicates that periodic maintenance, surveillance, and inspections will be conducted to ensure that coating or liner degradation will be promptly detected and corrected to provide reasonable assurance that the systems can perform their safety-related functions.

Date of issuance: December 18, 1998.

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

Amendment No.: 222.

Facility Operating License No. DPR–65: Amendment revised the Updated Final Safety Analysis Report and Appendix B to Operating License.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43206).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 18, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 5, 1995, as supplemented October 9, 1998.

Brief description of amendment: The amendment extends surveillance test intervals and allowable out-of-service times for instrumentation in the Emergency Core Cooling (ECCS), Rod Block, Isolation Group 4 (High Pressure Coolant Injection, or HPCI) and Isolation Group 5 (Reactor Core Isolation Cooling, or RCIC), Reactor Building Ventilation & Standby Gas Treatment, Recirculation Pump Trip and Alternate Rod Injection, and Shutdown Cooling Supply Isolation Systems.

Date of issuance: December 23, 1998.

Effective date: December 23, 1998, with full implementation within 30 days.

Amendment No.: 103.

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

Date of initial notice in **Federal Register**: August 30, 1995 (60 FR 45182). The October 9, 1998, submittal withdrew a portion of the original request, made additional editorial changes, and provided updated Technical Specification pages. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 15, 1996, as supplemented March 19 and October 12, 1998.

Brief description of amendment: The amendment revises the Technical Specifications so that either 8 or 12 hour shifts will be considered ''normal'' and 40 hours will be considered a ''nominal'' week, changes the wording for surveillances required ''once per shift'' to ''once per 12 hours,'' clarifies the ''once per hour'' wording related to fire watch patrols, and makes a number of other clarifications and typographical corrections.

Date of issuance: December 24, 1998.

Effective date: December 24, 1998, with full implementation within 30 days.

Amendment No.: 104.

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53951). The October 12, 1998, submittal provided additional clarifications and new TS pages. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 24, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998.

Brief description of amendments: The amendments revise Technical Specifications (TS) Table of Contents; TS 3.1, ''Reactor Coolant System;'' TS 4.0, ''Surveillance Requirements;'' TS 5.0, ''Design Features;'' and associated Bases by removing or relocating requirements that are adequately controlled by existing regulations other than 10 CFR 50.36 and the TS and by modifying TS 6.0 to more closely meet the format and content of the standard technical specifications.

Date of issuance: December 7, 1998.

Effective date: December 7, 1998, with full implementation of the TS and License Condition 7 by September 1, 1999. License Condition 6 shall be implemented by the next USAR update, but no later than June 1, 1999. Implementation shall also include the relocation of TS requirements to the appropriate licensee-controlled documents as identified in the licensee's application dated December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, and evaluated in the staff's safety evaluation attached to these amendments.

Amendment Nos.: 141 and 132.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Licenses and TS.

Date of initial notice in **Federal Register**: June 5, 1996 (61 FR 28618). The November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, submittals provided additional clarifying information, revised implementation dates, and updated TS pages. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 7, 1998. '

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 4, 1998.

Brief description of amendments: The amendments revise Technical Specifications (TS) 3.1.A.3.b, 4.18, and Bases for TS 4.18 to clarify the surveillance requirements and limiting conditions for operation of the reactor coolant vent system.

Date of issuance: December 17, 1998.

Effective date: December 17, 1998, with full implementation within 30 days.

Amendment Nos.: 142 and 133.

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

Date of initial notice in **Federal Register**: September 23, 1998 (63 FR 50938).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: April 17, 1997.

Brief description of amendment: The amendment revises Technical Specifications (TS) 2.12, ''Control Room Systems,'' to delete the limiting condition for operation (LCO) and surveillance for control room temperature and replace it with an associated LCO and surveillance for the control room air conditioning system. In addition, the amendment revises TS 2.1, ''Reactor Coolant System,'' TS 2.6, ''Containment System,'' and TS 2.8, ''Refueling Operations,'' and the associated surveillance requirements to incorporate the design basis requirements for refueling operations and to correspond to NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants.''

Date of issuance: December 31, 1998.

Effective date: December 31, 1998, to be implemented within 60 days from the date of issuance.

Amendment No.: 188.

Facility Operating License No. DPR–40. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 4, 1997 (62 FR 30639).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

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

Date of application for amendment: April 16, 1998, as supplemented August 20, 1998.

Brief description of amendment: The amendment will extend the surveillance interval for five instrument channels from the current 18 months to 24 months. The proposed amendment also revises Section 6 of the Technical Specifications to reflect updated analyses.

Date of issuance: December 16, 1998.

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

Amendment No.: 185.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56256).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 16, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 6, 1998.

Brief description of amendment: The amendment revises Appendix B Technical Specification 3.5, Main Condenser Steam Jet Air Ejector and Table 3.10–1, Radiation Monitoring Systems that Initiate and /or Isolate Systems including the associated Bases to provide Allowable Outage Times for selected instrumentation.

Date of issuance: December 28, 1998.

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

Amendment No.: 249.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43211).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 1, 1997, as supplemented on October 6, 1997, February 18 and July 7, 1998.

Brief description of amendments: The amendments revise Technical Specification Section 4.2.1 of Appendix B to require that Public Service Electric & Gas Company (PSE&G) adhere to the Incidental Take Statement, approved by the National Marine Fisheries Service (NMFS), but remove the specific requirements. Removing the specific requirements of Section 4.2.1 enables PSE&G to utilize relief granted by the NMFS on a case-by-case basis.

Date of issuance: December 18, 1998.

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

Amendment Nos: 216 and 196.

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

Date of initial notice in **Federal Register**: September 10, 1997 (62 FR 47698).

The October 6, 1997, February 18 and July 7, 1998 submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 18, 1998.

No significant hazards consideration comments received: No.

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

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: May 21, 1997, as supplemented on December 4, 1998. The December 4, 1998, submittal contained clarifying information only, and did not change the initial no significant hazards consideration determination.

Brief description of amendment: The amendment revises the Virgil C. Summer Nuclear Station Technical Specifications to change the methods for testing the control room and spent fuel pool ventilation system charcoal adsorbers from American National Standards Institute Standard N509–1980 to American Society for Testing and Materials Standard D3803–1989.

Date of issuance: December 23, 1998.

Effective date: December 23, 1998.

Amendment No.: 140.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: June 18, 1997 (62 FR 33133).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

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

Date of application for amendments: May 7, 1998.

Brief description of amendments: The amendments revise the reference for obtaining the thyroid dose conversion factors used in the definition of Dose Equivalent Iodine 131 (I–131) in Technical Specification Section 1.1, ''Definitions.''

Date of issuance: December 16, 1998.

Effective date: December 16, 1998, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2—145; Unit 3—137.

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

Date of initial notice in **Federal Register**: November 4, 1998 (63 FR 59595).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 16, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 12, 1998, as supplemented by letters dated September 18, 1998, October 29, 1998, and November 23, 1998.

Brief description of amendments: The amendments authorize revision of the San Onofre Nuclear Generating Station Updated Final Safety Analysis Report to incorporate a new turbine missile protection calculation methodology.

Date of issuance: December 21, 1998.

Effective date: December 21, 1998, to be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e) that occurs after 60 days of the date of issuance.

Amendment Nos.: Unit 2–146; Unit 3–138.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments authorize revisions to the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: November 9, 1998 (63 FR 60412).

The November 23, 1998, supplemental letter provided additional information and did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 21, 1998.

No significant hazards consideration comments received: No.

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

Tennessee Valley Authority, Docket No. 50–259, Browns Ferry Nuclear Plant, Unit No. 1, Docket No. 50–260, Browns Ferry Nuclear Plant, Unit No. 2, and Docket No. 50–296, Browns Ferry Nuclear Plant, Unit No. 3, Limestone County, Alabama

Date of amendment request: June 2, 1997 as supplemented November 19, 1998.

Description of amendment request: Presently Technical Specification (TSs) require both the recirculation loops to be operable and provide a 12-hour allowable outage time (AOT) for single loop operation (SLO) mode. The amendments modify TS to allow indefinite SLO instead of the 12-hour AOT.

Date of issuance: December 23, 1998.

Effective date: December 23, 1998.

Amendment No.: 236, 256 and 216.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 13, 1997 (62 FR 43377). The licensee's letter of November 19, 1998, did not expand the scope of the application or affect 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 December 23, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 18, 1998.

Brief description of amendment: Changes Technical Specification (TS) 3.7.4, Steam Generator Atmospheric Dump Valves (ADVs), and its associated bases by adding a new TS CONDITION, REQUIRED ACTION, and COMPLETION TIME to address a potential condition where two ADVs are made technically inoperable when one train of the safety-related auxiliary control air system is taken out of service.

Date of issuance: December 17, 1998.

Effective date: December 17, 1998.

Amendment No.: 16.

Facility Operating License No. NPF–90: Amendment revises the TSs.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43213) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 17, 1998.

No significant hazards consideration comments received: None.

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

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

Date of application for amendment: February 28, 1996, as supplemented October 2 and December 12, 1997, March 30 and December 11, 1998.

Brief description of amendment: The February 28, 1996 letter proposed to extend the surveillance interval for Westinghouse type AR relays with alternating current and direct current coils from quarterly to an 18 month interval. The letter of December 11, 1998 revised the scope of the application such that it now applies only to Westinghouse type AR relays which use alternating current coils. Accordingly, this amendment approves the extension of the surveillance interval only for Westinghouse type AR relays which use alternating current coils.

Date of issuance: December 30, 1998.

Effective date: December 30, 1998.

Amendment No.: 17.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: April 10, 1996 (61 FR 15998). The October 2 and December 12, 1997, March 30 and December 11, 1998 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 30, 1998, as supplemented by submittals dated October 27, November 30, and December 3, 1998. The supplemental submittals did not expand the scope of the original application or change the staff's proposed no significant hazards considerations determination.

Brief description of amendment: This amendment reflects the approval of the transfer of the authority to operate the Perry Nuclear Power Plant, Unit 1, under the license to a new company, FirstEnergy Nuclear Operating Company. In addition, several administrative changes unrelated to the transfer are being made to delete certain sections of the license relating solely to one-time historical events that have occurred.

Date of issuance: December 21, 1998.

Effective date: December 21, 1998.

Amendment No.: 96.

Facility Operating License No. NPF–58: This amendment revised the operating license.

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 29, 1998, as supplemented by submittals dated July 14, October 26, and November 30, 1998.

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

Date of issuance: December 21, 1998.

Effective date: December 21, 1998.

Amendment No.: 228.

Facility Operating License No. NPF–3: Amendment revised the operating license.

Date of initial notice in **Federal Register**: August 4, 1998 (63 FR 41602) The additional information provided in the supplemental submissions provided clarifying information only which did not affect the staff's proposed no significant hazards consideration or expand the scope of the application as noticed initially.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1998.

No significant hazards consideration comments received: No.

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

TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: August 2, 1996 (TXX–96434), as supplemented by letters dated October 2, 1998 (TXX–98215), and November 13, 1998 (TXX–98241 and TXX–98244).

Brief description of amendments: The amendment increases the allowed outage time (AOT) for a centrifugal charging pump from 72 hours to 7 days and adds a Configuration Risk Management Program.

Date of issuance: December 29, 1998.

Effective date: December 29, 1998, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 62; Unit 2—Amendment No. 48.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 27, 1998, (63 FR 65617) supersedes FR notice dated September 24, 1997.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 29, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

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

Date of application for amendment: May 8, 1998, as supplemented on July 10 and October 2, 1998.

Brief description of amendment: The amendment reduces the normal operating suppression pool water temperature limit and adds a time restriction for the temperature limit allowed during surveillances that add heat to the suppression pool.

Date of Issuance: December 28, 1998.

Effective date: December 28, 1998, to be implemented within 30 days.

Amendment No.: 163.

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

Date of initial notice in **Federal Register**: September 23, 1998 (63 FR 50941).

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

No significant hazards consideration comments received: No.

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

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: September 12, 1996, as supplemented April 24, 1997, and September 24, 1998

Brief Description of amendments: The amendments revise License Condition 3.I, Fire Protection, and relocate fire protection requirements from the Technical Specifications to the Updated Final Safety Analysis Report.

Date of issuance: December 16, 1998.

Effective date: December 16, 1998.

Amendment Nos.: 217 and 217.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Licenses and Technical Specifications.

Date of initial notice in **Federal Register**: November 4, 1998 (63 FR 59598).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 16, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

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

Date of application for amendment: October 10, 1996, as supplemented by letter dated November 9, 1998.

Brief description of amendment: The amendment changes Facility Operating License No. NPF–21 to authorize the storage of byproduct, source, and special nuclear materials at the WNP–2 site. These materials had been originally stored at the WNP–1 site and are not intended for use at WNP–2.

Date of issuance: December 29, 1998.

Effective date: December 29, 1998, to be implemented within 45 days from the date of issuance.

Amendment No.: 155.

Facility Operating License No. NPF–21: The amendment revised the operating license.

Date of initial notice in **Federal Register**: September 23, 1998 (63 FR 50942).

The November 9, 1998, supplemental letter provided additional clarifying information that did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 29, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Dated at Rockville, Maryland, this 6th day of January 1999.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 99–660 Filed 1–12–99; 8:45 am]

BILLING CODE 7590–01–P

THE PRESIDENT'S COUNCIL ON SUSTAINABLE DEVELOPMENT

Twenty-First Meeting of the President's Council on Sustainable Development (PCSD) To Take Public Comment on the Council's Recommendations and Draft Report to the President

ACTION: Notice.

SUMMARY: The President's Council on Sustainable Development (PCSD), a Presidential advisory council with representation from industry, government, environmental, and Native American organizations, will convene its twenty-first meeting in Washington, D.C. on Wednesday, February 10, 1999 to take public comment and finalize recommendations for its report to the President. A draft of the executive summary for this report is included below for public review. If you would like to read the entire report please visit our website at "http://www.whitehouse.gov/PCSD" or contact the PCSD office at the address or phone number below. The Council will consider all comments received.

The Council's current charter from the President is to forge consensus on policy, demonstrate implementation, get the word out about sustainable development, and evaluate progress. The Council is advising the President in four specific areas: (1) Domestic implementation of policy options to reduce greenhouse gas emissions; (2) next steps in building the new environmental management system of the 21st century; (3) promoting multi-jurisdictional and community cooperation in metropolitan and rural areas; and (4) policies that foster the United States' leadership role in sustainable development internationally. The final report to the President will fulfill this charter and culminate work in all four areas.

At the Council's last few meetings, the members have deliberated among themselves, listened to experts, and