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

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Point Beach Nuclear Plant, Units 1 and 2.

Agencies and Persons Consulted

In accordance with its stated policy, on July 29, 1997, the staff consulted with the Wisconsin State official, Ms. Sarah Jenkins of the Wisconsin Public Service Commission, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

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

For further details with respect to the proposed action, see the licensee's letter dated January 24, 1997, as supplemented by letter dated May 15, 1997, which are available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at The Lester Public Library, 1001 Adams Street, Two Rivers, WI 54241.

Dated at Rockville, Maryland, this 7th day of August 1997.

For the Nuclear Regulatory Commission.

Linda L. Gundrum,

Project Manager, Project Directorate III–1, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 97–21360 Filed 8–12–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving 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 July 19, 1997, through August 1, 1997. The last biweekly notice was published on July 30, 1997, (62 FR 40843).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

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

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

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

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

By September 12, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

Date of amendments request: July 8, 1997

Description of amendments request: The proposed amendments remove the suppression chamber water volume band from Technical Specification (TS) 3.6.2.1.a.1 while retaining the equivalent water level band. The values for the suppression chamber water volume corresponding to the low and high suppression chamber water levels will be retained in the Bases section of the TS and will be revised by the proposed amendments to account for the displacement of water due to the planned installation of larger emergency core cooling system suction strainers. The revised relationship between the high and low suppression chamber water levels and suppression chamber water volume will also be described in the Updated Final Safety Analysis Report.

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

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

The proposed change revises the values of the minimum and maximum suppression chamber pool water volume limits. The water inventory of the suppression chamber pool is not a precursor of an accident and, therefore, cannot increase the probability of an accident previously evaluated. The pressure suppression chamber water pool mitigates the consequences of loss-of-coolant accidents (LOCAs) transients [sic], and other events by providing a heat sink for reactor primary system energy releases. The proposed minimum and maximum pool water volume values will be consistent with the current suppression chamber pool water level limits. No changes to setpoints will be made as a result of the proposed change. The impact of the proposed change to the minimum and maximum suppression chamber pool volume limits on the suppression chamber pool temperatures and pressures following a design basis LOCA, an Safety/Relief Valve (SRV) blowdown event, an Anticipated Transient Without Scram (ATWS) event, an Appendix R fire event, and a station blackout event has been evaluated and does not cause accident parameters to exceed acceptable values. In addition, the impact the proposed change has on the time to reach cold shutdown when using the alternate Residual Heat Removal (RHR) shutdown cooling function is negligible. The potential impact the proposed change to the suppression chamber pool water volume limits has on SRV line loads, SRV discharge line reflood height, wetwell pressurization, suppression chamber pool swell loads, vent thrust loads, and condensation oscillation and chugging loads was also reviewed. The change to the suppression chamber pool water volume limits has no significant adverse impact on any of these parameters. As delineated above, the capability of the suppression chamber water pool to perform its mitigative functions is not affected by the proposed change. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change revises the values of the minimum and maximum volume of the suppression chamber water pool. The proposed change will not alter any physical mechanism by which the suppression chamber water pool volume is maintained between the minimum and maximum values. The suppression chamber pool water level will continue to be maintained between -27 and -31 inches. The suppression chamber pool water level limits are retained in Technical Specification (TS) 3.6.2.1.a.1, since this is the information available to the operators regarding the suppression chamber pool water volume limits. These level limits are equivalent to the suppression chamber pool water volume limits; therefore, it is only the presentation of the equivalency that is being relocated to the Bases and the Updated Final Safety Analysis Report (UFSAR). As such, the relocated suppression chamber pool water volume limits are not required to be in the TS to provide adequate protection of the public health and safety. As a result of the proposed strainer changes, there are no physical changes to any other suppression chamber components or instrumentation. No new mode of operation is introduced as a result of the proposed change. Analyses have been performed which conclude that the proposed change will not affect the operability of the equipment designed to mitigate the consequences of an accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change revises the values of the minimum and maximum suppression chamber water pool volumes. The pressure suppression chamber water pool mitigates the consequences of several postulated accidents and transients by providing a heat sink for the primary coolant system. These accidents and events are the postulated design basis LOCA, an SRV blowdown event, an ATWS event, an Appendix R fire, and station blackout events. The consequences of the change in the suppression pool water volume limits have been evaluated for these events, and there is no significant reduction in the margin of safety.

The results of the analyses for the postulated accidents and events indicate the temperature of the suppression chamber pool water could increase slightly as a consequence of the decrease in the minimum suppression chamber pool water volume limit. However, the suppression chamber pool water and containment temperatures remain within acceptable values. The impact of the calculated increase in containment temperature on the available Net Positive Suction Head (NPSH) for the Residual Heat Removal (RHR) and Core Spray pumps has been evaluated for the postulated design basis LOCA and indicate[s] adequate NPSH is maintained throughout the event.

The potential impact of the proposed change to the suppression chamber pool water volume limits on the SRV line loads, SRV discharge line reflood height, wetwell pressurization, suppression chamber pool swell loads, vent thrust loads, and condensation oscillation and chugging loads was evaluated with the conclusion that there are no adverse impacts on these parameters.

In addition, a small suppression chamber pool water temperature increase could result due to the reduction in minimum suppression pool volume limit in the event reactor shutdown is conducted through a path utilizing the suppression chamber pool. Such a shutdown path is an alternative to the normal RHR shutdown cooling function, and the small potential increase in temperature results in a negligible increase in the time required to reach cold shutdown conditions. Cold shutdown conditions can still be reached well within the Technical Specification requirements.

The proposed increase in the suppression pool water volume limit does not adversely impact containment parameters as a result of postulated accidents and events. The potential increase in temperature of the pressure suppression chamber pool water does not significantly decrease the ability to maintain containment parameters within acceptable limits. The potential increase in time to reach cold shutdown conditions utilizing the suppression pool as an alternative to the normal RHR shutdown cooling function is negligible. Therefore, the proposed change to revise the minimum and maximum suppression water pool volumes does not involve a significant reduction in a margin of safety.

The suppression chamber pool water level limits are retained in TS 3.6.2.1.a.1, since this is the information available to the operators regarding the suppression chamber pool water volume limits. These level limits are equivalent to the suppression chamber pool water volume limits and the equivalency is being relocated to the Bases and the UFSAR. As such, the relocated suppression chamber pool water volume limits are not required to be in the TS to provide adequate protection of the public health and safety.

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

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

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

NRC Project Director: Gordon E. Edison, Acting

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

Date of amendment request: July 21, 1997

Description of amendment request: Technical Specification Change Request Concerning Emergency Feedwater Surveillance Testing. This request is to make several changes to the ANO-2 Technical Specifications including an extension of the emergency feedwater (EFW) pump surveillance testing frequency, a reduction in the minimum steam generator pressure required to perform the surveillance testing on the turbine-driven EFW pump, and a modification to the EFW pump testing 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:

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

The proposed changes included in this amendment request are being made to the emergency feedwater (EFW) system technical specification (TS) surveillances. These changes include surveillance interval modifications, allowances to perform the turbine driven EFW pump surveillance at a lower steam generator (S/G) pressure, removing the requirements to perform specific EFW surveillance requirements (SRs) during plant shutdowns, bases changes, and various administrative changes. These changes are consistent with the applicable SRs located in NUREG-1432 and have therefore, been previously approved by the NRC.

These changes do not alter the functional characteristics of any plant component and do not allow any new modes of operation of any component. The accident mitigation features of the plant are not affected by the proposed amendment request. No modifications have been made to the EFW system due to this amendment request. Although the minimum steam generator pressure has been reduced for the turbine driven EFW pump testing, calculations show that significant margin exists between the proposed value and that needed to adequately perform the test. The capability of the EFW pumps to perform their required safety function is not impacted by this change. The addition of the electric driven EFW flow path verification will help [to] assure proper alignment of both trains of EFW following extended outages.

The accident mitigation features of the plant are not affected by the proposed amendment. No modification has been made to the pump or turbine driver. The capability of the turbine driven EFW pump to perform its required function is not impacted by this change. The EFW pumps will be tested in accordance with the more restrictive of the data points required by the safety analysis or the inservice testing program. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

No new possibility for an accident is introduced by modifying the proposed specifications for the surveillance testing of the EFW pumps. The EFW surveillance requirements will continue to demonstrated the pump's ability to perform its safety function. The modifications to the proposed EFW surveillance requirements are consistent with the current revision of NRC approved NUREG -1432, ''Standard Technical Specifications Combustion Engineering Plants'' (ITS). Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The safety function of the EFW system is not altered as a result of this change. The capability of the EFW pumps to perform their required function is not impacted by this change. The capability of the EFW pumps is not impacted by this change. The EFW pumps will be tested and proven operable in accordance with the more restrictive of the data points required by the safety analysis of the inservice testing program. The addition of the electric driven EFW flow path verification will help assure [to] proper alignment of both trains of EFW following extended outages. Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve significant hazards consideration.

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

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

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

NRC Project Director: James Clifford, Acting

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

Date of amendment request: June 26, 1997

Description of amendment request: The proposed amendment would revise the Operating License No. DPR-72, License Condition 2.C.(5) and delete the requirement for installation and testing of flow indicators in the emergency core cooling system to provide indication of 40 gallons per minute flow for boron dilution from the license. Approval of this amendment will allow removal of the appropriate flow indicators, DH-45-Fl and DH-46-Fl, from the Crystal River 3 (CR3) Final Safety Analysis Report.

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

Criterion 1

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

This license amendment removes the requirement for flow indication on the DH drop line and auxiliary pressurizer spray line for boron precipitation mitigation during a LOCA [Loss of Coolant Accident]. The original need for these indicators was to provide flow indication to the operator to aid in decision making relative to an alternate active method for boron precipitation prevention. Alternate active methods have been replaced by the passive flow path through the gaps which exist between the reactor vessel and the reactor vessel internals. Since auxiliary pressurizer spray flow is no longer used, and no other active means is required to be employed by the operator in the event drop line flow is not indicated, the original usefulness of and need for this indication no longer exists. Removal of this requirement from the license condition does not involve a change in the Improved Technical Specifications. The operators do not use the flow indication for decision making in post-accident conditions. Since these instruments are no longer used for boron precipitation mitigation during a LOCA, abandonment or removal of flow indicator DH-45-Fl and DH-46-Fl does not increase the probability of an accident because no previously evaluated accidents at CR-3 are initiated by DH-45-Fl or DH-46-Fl. Those CR-3 accidents that are analyzed are contained in the Final Safety Analysis Report (FSAR) and include events such as Loss-of-Coolant Accidents, Main Steam Line Breaks, Station Blackout, Anticipated Transients Without Scram, etc. Since DH-45-Fl and DH-46-Fl are attached to the outside of the DH drop line and auxiliary pressurizer spray line, their removal will not change the design, material, or construction standards applicable to the DH System piping. The removal of the indicator will not affect overall system performance of the ECCS. All of these previously evaluated accidents described in the CR-3 FSAR have dose consequences which remain well within the requirements of 10 CFR Part 100 (25 rem whole body, 300 rem thyroid) and GDC [General Design Criterion] 19 (5 rem whole body, or its equivalent to any part of the body). Removal of DH-45-Fl and DH-46-Fl will not alter any assumptions made in evaluating the radiological consequences of any accident described in the FSAR nor will it affect any fission product barriers since the ECCS and containment systems will still perform to meet design requirements. Therefore, removal of DH-45-Fl and DH-46-Fl will not alter the consequences of an accident previously evaluated.

Criterion 2

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

The proposed license amendment removes the requirement for indicators which were originally installed to aid the operator in decision making relative to an alternate flow path for boron precipitation mitigation during a LOCA. These indicators no longer serve this purpose, since alternate active flow paths are no longer considered. Evaluations which consider boron precipitation no longer rely on three active methods of mitigation, but rather one active and one passive. Operator action is not required to effect the backup method in the event that the primary method fails due to a single active failure. The flow indicators are external to the DH System piping. They do not penetrate any piping so their removal cannot create the possibility of a new or different kind of accident. The accident mitigation strategies remain the same regardless of whether or not the flow indicators are present. Therefore, the flow indicators serve no purpose in the analyses. The proposed amendment does not affect any of the parameters or conditions that could contribute to the initiation of any accidents.

Criterion 3

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

Boron precipitation within the reactor vessel during post-LOCA conditions, if it were to occur, would challenge the margin of safety that is provided by assuring compliance with Criterion 5 of 10 CFR 50.46. The license amendment does not change the methodology of mitigating the consequences of boron precipitation following a LOCA as described in the current licensing basis. The primary method of flow through the DH drop line and the use of gap flow as the ''backup'' method for prevention of boron precipitation have been analyzed, shown to meet all the criteria of 10 CFR 50.46, and accepted by the NRC. The passive method requires no specific operator action for initiation, in the event that the primary method fails due to a single active failure. Therefore, the indication serves no safety function and does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: 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, Citrus County, Florida

Date of amendment request: July 18, 1997

Description of amendment request: The proposed amendment would revise the Crystal River 3 (CR-3) technical specifications (TS) to incorporate a new TS 3.4.11 for a Low Temperature Overpressure Protection (LTOP) System. The proposed changes would be consistent with the recommendations in the NRC Generic Letter 88-11, ''NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations.'' TS 3.5.3 and associated TS Bases would also be revised to reflect the proposed change.

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

This change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

There are currently no LTOP requirements in the CR-3 Improved Technical Specifications. CR-3 currently implements LTOP features through administrative controls and a lowered PORV [power-operated relief valve] setpoint. The proposed change will establish new LTOP technical specification requirements necessary to preclude an LTOP event from occurring. The proposed LTOP requirements are based on safety analyses that apply ASME [American Society of Mechanical Engineers] Code Case N-514. These requirements will decrease the probability of a low temperature overpressure event by providing protection for all pressure and temperature combinations for which a low temperature overpressure event may be postulated.

The consequences of a low temperature overpressure accident are not affected by this change. There is no change to the 10 CFR [Code of Federal Regulations] Part 100 dose calculation for a low temperature overpressure accident.

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

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

The new LTOP Technical Specification does not require modification to the plant nor does it create a new mode of plant operation. The LTOP system adds no new accident initiators.

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

The proposed change does not involve a significant reduction in the margin of safety and will provide added safety benefit gained through the requirements to preclude a low temperature overpressurization event to the RCS [reactor coolant system].

The margin of safety prior to having an LTOP system was limited due to the informal, administrative method of minimizing the impact of a low temperature overpressure accident. By formalizing these requirements into a technical specification, at the least, margin of safety is retained and perhaps improved due to the elevated significance of required actions.

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, Citrus County, Florida

Date of amendment request: July 29, 1997

Description of amendment request: The proposed amendment would revise the Crystal River Nuclear Generating Unit 3 (CR3) technical specifications (TS) to add subcooling margin and decay heat removal (low pressure injection) flow and correct certain nomenclature in the post-accident monitoring (PAM) instrumentation TS. In addition, the licensee proposes to add emergency diesel generator (EDG) kilowatt (kW) indication to the PAM instrumentation. Specifically, the following TS would be revised:

A. Table 3.3.17-1, Function 8: The descriptor is changed from ''Containment Pressure (Narrow Range)'' to ''Containment Pressure (Expected Post-Accident Range).''

B. Table 3.3.17-1, Function 18: The required channels for Core Exit Temperature (Backup) is changed from ''2 sets of 5'' to ''3 per core quadrant.''

C. Table 3.3.17-1: A new Function 20 is added and designated as ''Low Pressure Injection Flow'', with 2 required channels, and Condition E.

D. Table 3.3.17-1: A new Function 21 is added and designated as ''Degrees of Subcooling'', with 2 required channels, and Condition E.

E. Table 3.3.17-1: A new Function 22 is added and designated as ''Emergency Diesel Generator kW Indication'', with 2 required channels, and Condition E. A note clarifying the number of required channels is added: ''(c): one indicator per EDG''.

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 items A, B, C, D and E corresponds to the specific TS changes described above.

1. The proposed changes will not significantly increase the probability or consequences of an accident previously evaluated because:

A/B. The changes in containment pressure and core exit thermocouple nomenclature do not reflect any physical changes to the facility. This would have no impact on accident probability or consequences.

C/D/E. The addition of low pressure injection flow, degrees of subcooling, and EDG kW indication to the Post-Accident Monitoring Instrumentation LCO [Limiting Condition for Operation] is being done to comply with a commitment made during the technical specification improvement program to include in the technical specifications that instrumentation which monitors variables classified as Type A in accordance with Regulatory Guide 1.97. These three variables have been reclassified as Type A. The associated instruments are used in post-accident conditions to prompt the operators to take certain mitigative actions. Therefore, the probability of an accident occurring is unaffected. As part of the re-classification of these variables to Type A and inclusion in technical specifications, the associated monitoring instrumentation will be under more strict surveillance and control, which provides additional assurance that the prescribed manual operator actions will be implemented when necessary. This, in turn, assures the previously evaluated accident consequences remain valid.

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

A/B. The changes in containment pressure and core exit thermocouple nomenclature do not reflect any physical changes to the facility. The changes provide clarification for the instruments which are required to comply with the LCO. This would not create possibility of a new or different kind of accident.

C/D/E.The addition of low pressure injection flow, degrees of subcooling, and EDG kW indication to the Post-Accident Monitoring Instrumentation LCO is being done to comply with a commitment made during the technical specification improvement program to include in the technical specifications that instrumentation which monitors variables classified as Type A in accordance with Regulatory Guide 1.97. These three variables have recently been reclassified as Type A. The associated instruments are used after an accident occurs to prompt the operators to take certain mitigative actions. Since the instrumentation is used only post-accident, these changes do not create the possibility of a new or different kind of accident.

3. The proposed change will not involve a significant reduction to the margin of safety because:

A/B. The changes in containment pressure and core exit thermocouple nomenclature have no affect on the margin of safety. The changes provide clarification of the technical specifications. This reduces the potential for confusion regarding this instrumentation.

C/D/E. The addition of low pressure injection flow, degrees of subcooling, and EDG kW indication to the post-accident monitoring instrumentation table in technical specifications results in added controls on the OPERABILITY of this post-accident monitoring instrumentation and provides greater assurance that it will be available should an accident occur.

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

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

Date of amendment request: July 18, 1997

Description of amendment request: The proposed amendment adds a new Technical Specification and associated Bases to address the operability of the steam generator atmospheric relief bypass valves (SGARBVs).

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:

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed revision does not involve an SHC because the revision would not:

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

The operability of the SGARBVs provides a method to recover from a SGTR [steam generator tube rupture] event during which the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to limit the primary to secondary break flow into the ruptured steam generator. For other design events, the SGARBVs provide a safety grade method for cooling the unit to residual heat removal entry conditions should the preferred heat sink via the steam bypass system or the steam generator atmospheric relief valves be unavailable. This proposed revision to the Technical Specifications will add a new Technical Specification 3/4.7.1.6 and its associated Bases Section 3/4.7.1.6 which were developed bases on the information contained in the Westinghouse Improved Standard Technical Specifications, NUREG 1431, Rev. 1. The proposed specification and bases provide further assurance that the SGARBVs will be available to function as described in the accident analysis.

Therefore, the proposed revision 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.

This proposed revision to the Technical Specifications to add a new specification and bases for the SGARBVs does not cause a change in the operation of any system or component during normal or accident conditions.

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

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

The proposed new Technical Specification 3/4.7.1.6 and its associated Bases Section 3/4.7.1.6 were developed based on the information contained in the Westinghouse Improved Standard Technical Specifications, NUREG 1431, Rev. 1. The SGARBV's are not currently in the Technical Specifications of Millstone Unit No. 3 and are being added to ensure accident mitigation functional capability. The NUREG 1431, Rev. 1 surveillance frequency is 18 months. The NUREG 1431, Rev. 1 surveillance frequency bases reads ''operating experience has shown that these components usually pass the surveillance when performed at the 18 month frequency''. The proposed frequency acceptability has been evaluated by reviewing SGARBV AWO's [automated work order's] for the period from Jan. 1990 to April 1997 to confirm the absence of excessive work orders which indicate valve functional failures and none were identified. Additionally, each SGARBV line consists of one SGARBV and an associated block valve. These proposed changes are consistent with the design and operation of the SGARBVs. There is no negative affect on the dose consequences from any design basis event or core damage frequency.

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

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment requests: November 27, 1996

Description of amendment requests: The proposed amendment[s] would incorporate new steam generator tube sleeve designs and installation and examination techniques into the Prairie Island Technical Specifications.

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

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

The supporting technical evaluation and safety evaluation for the Combustion Engineering leak tight sleeves demonstrate that the sleeve configuration will provide steam generator tube structural and leakage integrity under normal operating and accident conditions. The sleeve configurations have been designed and analyzed in accordance with the requirements of the ASME [American Society of Mechanical Engineers] Code. Mechanical testing has shown that the sleeve and sleeve joints provide margin above acceptance limits. Ultrasonic examination is used to verify the leak tightness of the above the [sic] tubesheet sleeve welds. Testing has demonstrated the leak tightness of the hard roll joint as well as the structural integrity of the hard roll joint. Tube rupture can not occur at the hard roll joint due to the reinforcing effect of the tubesheet. Tests have demonstrated that tube collapse will not occur due to postulated LOCA [loss-of-coolant accident] loadings.

The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the event that significant leakage did occur from the sleeve joints or that a sleeve assembly ruptured. Any leakage through the sleeve assembly is fully bounded by the existing steam generator tube rupture analysis included in the Prairie Island Plant USAR [updated safety analysis report]. The proposed sleeving repair does not adversely impact any other previously evaluated design basis accident.

The sleeve minimum acceptable wall thickness used for developing the depth based plugging limit for the sleeve is determined using the guidance of draft Regulatory Guide 1.121 [≥Bases for Plugging Degraded PWR [Pressurized-Water Reactor] Steam Generator Tubes≥] and the pressure stress equation of Section III of the ASME Code. Evaluation of the minimum acceptable wall thickness for normal, upset, and postulated accident condition loading per the ASME Code finds that the limiting condition is established from normal operating conditions which then bounds the upset and accident condition values. Allowance for non-destructive examination and growth of existing sleeve wall degradation must be made when determining the sleeve plugging limit. The proposed plugging limit is 40% through wall degradation. The sleeve assembly will be examined by state of the art non-destructive examination techniques on a periodic basis to provide early indication of sleeve degradation. The corrosion resistance of the Alloy 690 sleeve has been verified by field experience at Prairie Island. The oldest Alloy 690 sleeves were installed May 1987. No indication of corrosion of the sleeve or the parent tube in the weld joint has been identified by state-of-the-art eddy current examination. These oldest sleeve welds did not receive post weld heat treatment. In addition, 5 sleeves were removed for destructive examination in February, 1996. No corrosion was found in any of these sleeves including those dating from October 1992. The pulled sleeves had received post weld heat treatment. Post weld heat treatment can be optionally applied to the free span sleeve weld joints to reduce the susceptibility of the weld joint and parent tube to stress corrosion cracking. Since the sleeve design meets the requirements of the ASME code and mechanical tests have demonstrated margins above acceptance criteria, the installation of the Combustion Engineering leak tight sleeves will not increase the probability or consequences of an accident previously evaluated.

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

Installation of sleeves does not introduce any significant changes to the plant design basis. The use of a sleeve to span a degraded region of steam generator tubing restores the structural and leakage integrity of the tubing to meet the original design bases. Stress and fatigue analysis of the sleeve assembly shows that the requirements for ASME Code are met. Mechanical testing has demonstrated that margin exists above the design criteria. Any hypothetical accident as a result of any degradation in the sleeved tube would be bounded by the existing tube rupture accident analysis.

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

The use of the sleeves to repair degraded steam generator tubing has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of the ASME Code and draft Regulatory Guide 1.121 and to maintain the primary to secondary pressure boundary under normal and postulated accident conditions. The safety factors used in the verification of the strength of the sleeve assembly are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The operational and faulted condition stresses and cumulative fatigue usage are bounded by the ASME Code requirements. The sleeve assembly has been verified by testing to prevent both tube pullout and significant leakage during normal and postulated accident conditions. A test program was conducted to ensure the rolled joint design for the lower joint in the tubesheet sleeve was leak tight and capable of withstanding the designs loads. The primary coolant pressure boundary of the sleeve assembly will be periodically inspected by non-destructive examination to identify sleeve degradation due to operation. Installation of sleeves will decrease the number of tubes which must be taken out of service. There is a small amount of primary coolant flow reduction due to sleeves for which an equivalent plugging sleeve to plug ratio is assigned and is used to assess the final equivalent plugging percentage used as an input to other safety analyses. Because the sleeve maintains the design basis requirements for the steam generator tubing, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR Part 50, Section 50.92.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment requests: May 15, 1997

Description of amendment requests: The proposed amendments would change the Technical Specifications (TS) to revise certain limitations on reactor coolant system leakage and steam generator tube surveillance. The proposed changes would implement a voltage-based repair criteria per the requirements of NRC Generic Letter 95-05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.'' In addition, a typographical error in TS Section 4.12.c. is being corrected.

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

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

The supporting technical evaluation and safety evaluation for the voltage based repair criteria demonstrate that steam generator tube structural and leakage integrity under normal operating and accident conditions will be maintained. Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate (TSP). Test data referenced in Generic Letter 95-05 indicates that tube burst cannot occur within the TSP, even for tubes which have 100% throughwall electric discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Since tube-to-TSP proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition, main steamline break (MSLB) pressure differential. The Regulatory Guide (RG) 1.121 [≥Bases for Plugging Degraded PWR [Pressurized-Water Reactor] Steam Generator Tubes≥] criterion requiring maintenance of a safety factor of 1.43 times the MSLB pressure differential on tube burst is satisfied by 7/8'' diameter tubing with bobbin coil indications with signal amplitudes less than the current 8.7 volts structural limit, regardless of the indicated depth measurement.

The upper voltage repair limit (V_{URL}) will be determined prior to each outage using the most recently NRC approved database to determine the tube structural limit (V_{SL}). The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{NDE}) and growth (V_{GR}) to establish V_{URL}. Using the Generic Letter (GL) 95-05 NDE and growth allowances for an example, the NDE uncertainty component of 20% and a voltage growth allowance of 30% per full power year can be utilized to establish a V_{URL} of 5.2 volts.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated MSLB outside of containment but upstream of the main steam isolation valve (MSIV) represents the most limiting radiological conditions to the plugging criteria. In support of [the] implementation of the revised plugging limit, analyses will be performed to determine whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary-to secondary leakage would result in postulated off site and control room doses exceeding the limits established for application of the voltage-based repair criteria at Prairie Island. A separate calculation has determined the maximum allowable MSLB leakage limit in a faulted loop. This limit was calculated using the technical specification reactor coolant system (RCS) Iodine-131 activity level of 1.0 microcuries per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800 [≥Standard Review Plan≥]. The projected MSLB leak rate calculation methodology prescribed in Section 2.b of Generic Letter 95-05 will be used to calculate the end-of-cycle (EOC) leakage. Projected EOC voltage distribution will be developed using the most recent EOC eddy current results and considering an appropriate voltage measurement uncertainty and indication growth allowance. The log-logistic probability of leakage correlation will be used to establish the MSLB leak rate used for comparison with the faulted loop allowable limit. Therefore, as implementation of the voltage-based repair criteria does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences, the proposed amendment[s] [do] not result in any increase in the probability or consequences of an accident previously evaluated in the Updated Safety Analysis Report (USAR).

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

Implementation of the proposed steam generator tube voltage-based repair criteria does not introduce any significant changes to the plant design basis. Use of the voltage-based repair criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations since tubes with outside diameter stress corrosion cracking (ODSCC) not occurring inside the thickness of the tube support plates will be plugged or repaired. Neither a single or multiple tube rupture event would be expected during all plant conditions in a steam generator in which the voltage based repair limit has been applied.

Northern States Power will implement a maximum primary-to-secondary leak rate limit of 150 gpd [gallons per day] per steam generator to help preclude the potential for excessive leakage during all plant conditions. The Regulatory Guide 1.121 criterion for establishing operational leak rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit provides for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length.

The operational leakage limit will be reduced to 150 gpd limit consistent with Generic Letter 95-05. This limit is expected to provide for plant shutdown prior to reaching critical lengths for MSLB conditions using the lower 95% leak rate data. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncover will provide benefit to the burst capacity of the intersection and only a small percentage of the TSPs are deflected greater than the TSP thickness during a postulated MSLB.

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

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

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

The use of the voltage-based repair criteria at Prairie Island maintains steam generator tube integrity commensurate with the criteria of the ASME [American Society of Mechanical Engineers] Code and Regulatory Guide 1.121. Regulatory Guide 1.121 describes a method acceptable to the Commission for meeting GDCs [General Design Criteria] 14, 15, 30, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be repaired or removed from service. Upon implementation of the proposed criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to the steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the tube support plate elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions in order to assure that radiological consequences meet the requirements of Generic Letter 95-05.

Previous evaluations have indicated a potential for tube deformation and collapse during a postulated loss-of-coolant accident (LOCA) plus safe-shutdown-earthquake (SSE) event. The tube collapse potential arises from TSP deformation at the support plate wedges. Evaluation of the Westinghouse umbrella seismic spectra provided in Westinghouse letter NSP-92-152 for Model 51 steam generators shows that Prairie Island is bounded by those spectra and that no tubes will undergo deformation due to the combined effects of LOCA plus SSE. Therefore, no tubes need to be excluded from application of the voltage based criteria due to deformation resulting from combined LOCA plus SSE loadings. Addressing Regulatory Guide 1.83 [≥Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes≥] considerations, implementation of the voltage-based repair criteria is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization, by an extensive bobbin coil inspection which will include 100% of the hot leg TSP intersections and cold leg intersections down to the lowest cold leg TSP with known ODSCC, by the determination of the TSPs having ODSCC using at least 20% random sampling of tubes inspected over their full length, and by rotating pancake coil inspection (or equivalent) requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

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

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or any Bases of the plant Technical Specifications.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR Part 50, Section 50.92.

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. In addition, the proposed correction to a typographical error has no effect on the three standards of 10

CFR 50.92(c). Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: June 4, 1997

Description of amendment request: The proposed Technical Specifications (TSs) amendment revises TS Surveillance Requirement 3.8.2.1 to no longer require that automatic emergency diesel generator (EDG) auto-start and trip bypass features must be functional when the emergency core cooling system (ECCS) is not required to be 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. The proposed change to the facility does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will eliminate an inconsistency between Technical Specifications 3.3.5.1, 3.5.2, and 3.8.2 by clarifying that the EDG auto-start and EDG trip bypass on ECCS initiation capability is not required during periods in which ECCS is not required to be OPERABLE. No physical changes to the facility will be made per this change. The systems, structures, and components affected by this change are considered to be accident mitigators and not accident initiators. The affected systems, structures, and components will continue to operate within the current design parameters. The ability of the EDGs to auto-start on a loss of offsite power or degraded voltage will remain unchanged. No new failure modes or conditions adverse to safety will be created as a result of this change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will eliminate an inconsistency between Technical Specifications 3.3.5.1, 3.5.2, and 3.8.2 by clarifying that the EDG auto-start and EDG trip bypass on ECCS initiation capability is not required during periods in which ECCS is not required to be OPERABLE. No physical changes to the facility will be made per this change. The systems, structures and components affected are considered to be accident mitigators not accident initiators. The affected systems, structures and components will continue to operate within the current design parameters. No new failure modes or conditions adverse to safety will be created as a result of this change. The plant conditions which do not require any ECCS to be OPERABLE, (i.e., the plant in MODE 5, the spent fuel storage pool gates are removed, water level is greater than or equal to 458 inches above reactor pressure vessel instrument zero, and there are no OPDRVs [operations with the potential of draining the reactor vessel] in progress) ensure sufficient coolant inventory to allow operator action to prevent uncovering the fuel. The ability of the EDGs to auto-start on a loss of offsite power or degraded voltage will remain unchanged. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will eliminate an inconsistency between Technical Specifications 3.3.5.1, 3.5.2, and 3.8.2 by clarifying that the EDG auto-start and EDG trip bypass on ECCS initiation capability is not required during periods in which ECCS is not required to be OPERABLE. The ECCS and EDGs capability to perform the required safety functions as described/required in the bases of the current plant Technical Specifications will be maintained. Therefore, the proposed change to the facility does not result in a significant reduction in any margin of safety.

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

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101

NRC Project Director: John F. Stolz

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-278, Peach Bottom Atomic Power Station, Unit No. 3, YorkCounty, Pennsylvania

Date of application for amendment: June 30, 1997

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) 2.1.1.2 safety limit minimum critical power ratios (SLMCPRs) to be consistent with the use of GE 13 fuel in the Unit 3 core for operating cycle 12.

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

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

The derivation of the cycle-specific SLMCPRs for incorporation into the TS, and its use to determine cycle-specific thermal limits, have been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE-24011-P-A-13, and U.S. Supplement, NEDE-24011-P-A-13-US, August, 1996, and the ''Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR.'' Amendment 25 was submitted by GENE to the U.S. Nuclear Regulatory Commission (USNRC) on December 13, 1996. This change in SLMCPRs cannot increase the probability or severity of an accident.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. The fuel licensing acceptance criteria for the SLMCPR calculation apply to PBAPS, Unit 3, Cycle 12 in the same manner as they have applied previously. The probability of fuel damage is not increased. Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR is a TS numerical value, designed to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core during the limiting postulated accident. It cannot create the possibility of any new type of accident. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE-24011-P-A-13, and U.S. Supplement, NEDE-24011-P-A-13-US, August, 1996, and the ''Proposed

Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR.'' Amendment 25 was submitted by GENE to the U.S. Nuclear Regulatory Commission (USNRC) on December 13, 1996.

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

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

The margin of safety as defined in the TS Bases will remain the same. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE-24011-P-A-13, and U.S. Supplement, NEDE-24011-P-A-13-US, August, 1996, and the ''Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR.'' Amendment 25 was submitted by GENE to the U.S. Nuclear Regulatory Commission (USNRC) on December 13, 1996. The fuel licensing acceptance criteria for the calculation of the SLMCPR apply to PBAPS [Peach Bottom Atomic Power Station], Unit 3 Cycle 12 in the same manner as they have applied previously. The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes do not involve a 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: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101

NRC Project Director: John F. Stolz

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

Date of amendment request: April 14, 1997

Description of amendment request: The proposed amendment revises Appendix A, Section 6 of the Technical Specifications. The changes will enable Safety Review Committee (SRC) to review plant staff performance by deleting the plant staff performance requirement from Section 6.5.2.9.b and incorporating a plant staff review requirement in Section 6.5.2.8. The amendment also replaces the position title of Vice President (VP) Regulatory Affairs and Special Projects (RASP) with Director of Regulatory Affairs and Special Projects.

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

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

Response:

This amendment application does not involve a significant increase in the probability or consequences of an accident previously analyzed. The proposed changes allow the SRC to perform a review, rather than an audit, of plant staff performance. This change does not diminish the SRC's effectiveness. A review of the 1995 QA [quality assurance] audit of plant staff performance shows that no findings related to plant staff performance were issued. This indicates that the other review mechanisms currently in place are sufficient to ensure that plant staff performance is monitored.

The position title change of VP-RASP to Director-RASP is an administrative change as all previously performed functions are being maintained. Therefore, the proposed changes do not affect the probability or consequences of any previously analyzed accident.

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

Response:

This amendment application does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes affect an SRC audit requirement and a management position title. These changes do not affect plant equipment or the way the plant operates. Therefore, they cannot create a new or different kind of accident.

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

Response:

This amendment application does not involve a significant reduction in a margin of safety. The requested Technical Specification revisions require the SRC to review rather than audit facility staff performance and will not diminish the effectiveness of the SRC. A review of the 1995 audit confirms that performance of the annual audit is redundant as no findings or recommendations concerning plant staff performance were made. The QA/ORG quarterly trend reports and SRC review of facility staff performance are adequate to ensure that plant staff performance is properly monitored.

The position title change (VP-RASP to Director-RASP) is an administrative change as all previously performed functions are being maintained. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019

NRC Project Director: Alexander W. Dromerick, Acting Project Director

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

Date of amendment request: March 31, 1997, as supplemented by letter dated July 16, 1997. The July 16, 1997, supplement supersedes the March 31, 1997 application.

Description of amendment request: The proposed amendment would provide changes to Technical Specification (TS) 2.1.2, ''THERMAL POWER, High Pressure and High Flow,'' ACTION a.1.c for TS 3.4.1.1, ''Recirculation Loops,'' and the Bases for TS 2.1, ''Safety Limits.'' These changes are being made to implement an appropriately conservative Safety Limit Minimum Critical Power Ratio, to include Cycle 8 specific analyses, for all Hope Creek core and fuel designs.

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

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

The derivation of the revised SLMCPRs for Hope Creek for incorporation into the Technical Specifications, and its use to determine cycle—specific thermal limits, have been performed using NRC approved methods. Additionally, interim implementing procedures which incorporate cycle—specific parameters have been used which result in a more restrictive value for SLMCPR. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

There are no significant increases in the consequences of an accident previously evaluated. The basis of the MCPR Safety Limit is to ensure that no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Therefore, the proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes contained in this submittal result from an analysis of the Cycle 7 and Cycle 8 core reloads using the same fuel types as previous cycles. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed Technical Specification changes do not create the possibility of a new or different kind of accident, from any accident previously evaluated.

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

The margin of safety as defined in the Technical Specification bases will remain the same. The new SLMCPRs are calculated using NRC approved methods which are in accordance with the current fuel design and licensing criteria. Additionally, interim implementing procedures, which incorporate cycle—specific parameters, have been used. The MCPR Safety Limit remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed Technical Specification changes do not involve a 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: J. J. Keenan, Esquire, Nuclear Business Unit - N21, P.O. Box 236, Hancocks Bridge, NJ 08038

NRC Project Director: John F. Stolz

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

Date of amendment request: April 1, 1997, as supplemented by letter dated May 30, 1997

Description of amendment request: The proposed amendment would provide changes to Technical Specifications (TSs) 4.6.1.1, ''Primary Containment Integrity,'' 3/4.6.1.2, ''Primary Containment Leakage,'' 3/4.6.1.3, ''Primary Containment Air Locks,'' 4.6.1.5.1, ''Primary Containment Structural Integrity,'' and 4.6.1.8.2, ''Drywell and Suppression Chamber Purge System.'' The amendment would also change the Bases for 3/4.6.1.2, ''Primary Containment Leakage,'' 3/4.6.1.3, ''Primary Containment Air Locks,'' 3.4.6.1.5, ''Primary Containment Structural Integrity,'' Section 6, ''Administrative Controls,'' and License Condition 2.D of Facility Operating License NPF-57. A new TS, 6.8.4.e, ''Primary Containment Leakage Rate Testing Program,'' would be added. These changes modify the TSs and the Facility Operating License to adopt the performance based containment leak rate testing requirements (Option B) of 10 CFR Part 50, Appendix J.

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

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

Containment leak rate testing is not an initiator of any accident. The proposed changes do not make any physical changes to the containment and do not affect reactor operations or the accident analyses. Therefore, the proposed changes do not involve a significant increase in the probability of any previously evaluated accident.

Since the allowable leakage rate is not being changed and since the analysis documented in NUREG-1493, ''Performance-Based Containment Leak-Test Program'' concludes that the impact on public health and safety due to extended intervals is negligible, the proposed changes will not involve a significant increase in the consequences of any previously evaluated accident.

Therefore, adoption of a performance-based leakage testing requirements will provide an equivalent level of safety and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No physical changes are being made to the plant, nor are there any changes being made to the operation of the plant as a result of the proposed changes. In addition, no new failure modes of plant equipment previously evaluated are being introduced.

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

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

The proposed changes are based on NRC-accepted provisions and maintain adequate levels of reliability of containment integrity. The performance-based approach to leakage rate testing recognizes that historically good results of containment testing provide appropriate assurance of future containment integrity. This supports the conclusion that the impact on the health and safety of the public as a result of extended test intervals is negligible. Since the analysis documented in NUREG-1493 confirms that the performance based schedule continues to maintain a minimal impact on public risk, it can be concluded that the margin of safety is not significantly affected by the proposed changes.

Therefore, the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: July 3, 1997

Description of amendment request: The proposed amendment would change Technical Specification Table 3.6.3-1, ''Primary Containment Isolation Valves'' to add valves to the list, therein.

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

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

The accidents previously evaluated in the UFSAR [Updated Final Safety Analysis Report] that could be possibly affected by this proposal are those involving loss of coolant scenarios such as a piping or instrument line break. The proposed relief valves, associated piping and the affected portions of containment penetration piping are not initiators of those accidents evaluated in the UFSAR. The proposed relief valves limit the post-accident maximum expected pressures of the affected piping segments within ASME [American Society of Mechanical Engineers] code allowables and system design pressures. The modification does not cause any system or component to be operated outside of their design rating allowed by applicable codes. The proposed relief valves will be safety-related and Seismic Category I components (except for the relief valve discharge piping, which will be non-safety related and seismically analyzed, and will meet the design, material and construction standards applicable to the affected piping segments[)].

The proposed modifications do not jeopardize the capability of the containment isolation valves in the affected penetrations to close on the receipt of a containment isolation signal or to mitigate the consequences of design basis accidents evaluated in the UFSAR. Although the modifications will result in system pressures to be above their currently established design values, the new peak operating pressures of the affected piping segments will be limited to within the requirements of the ASME code. The modification will not alter any assumptions previously made or change, degrade, or prevent actions described in or assumed in evaluating the radiological consequences of the postulated design basis accidents. Containment structure temperature and pressure limits will not be exceeded with this modification and the offsite dose consequences will not be affected.

Therefore these changes will not significantly increase the probability of an accident previously evaluated, nor involve a significant increase in the consequences of an accident previously evaluated.

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

Accidents or malfunctions of equipment important to safety previously evaluated in the UFSAR relating to the proposed modification involve the single active failure of a containment isolation valve to close upon receipt of a containment isolation signal or its failure to limit the containment bypass leakage following its closure. The proposed modification: 1) does not impact the automatic closure times of the containment isolation valves; 2) does not impact their capability to maintain leak tightness during a postulated design basis accident; and 3) does not adversely impact the manner in which any system is operated. The proposed modification does not compromise the UFSAR accident analysis assumptions and/or limits. The licensing basis safety analysis limits for all systems important to safety continue to be met. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed modifications and Technical Specification changes do not change the design limits, acceptance criteria or accident analysis assumptions pertaining to the containment isolation valves, their associated piping or any other safety-related systems, structures or components. The proposed modification does not impact the automatic closure times of the containment isolation valves, nor does it impact their capability to maintain leak tightness during a postulated design basis accident. For the systems affected by these penetration modifications, there is no change in system function or structural integrity introduced with these proposed changes. Therefore, the changes contained in this request do not result in a significant reduction in a margin of safety for the containment isolation capability of Hope Creek.

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: J. J. Keenan, Esquire, Nuclear Business Unit - N21, P.O. Box 236, Hancocks Bridge, NJ 08038

NRC Project Director: John F. Stolz

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

Date of amendment request: July 7, 1997

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.8.4.2, ''Motor Operated Valves - Thermal Overload Protection (BYPASSED),'' to relocate the list of applicable valves (TS Table 3.8.4.2-1) to the Hope Creek (HC) Generating Station Updated Final Safety Analysis Report (UFSAR).

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

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

The proposed TS revisions involve: 1) no hardware changes; 2) no changes to the operation of any systems or components in normal or accident operating conditions; and 3) no changes to existing structures, systems or components. The relocation of Technical Specification Table 3.8.4.2-1 to the UFSAR and existing surveillance procedures will continue to ensure that safety-related motor-operated valves (MOVs) are capable of performing their intended safety functions. Therefore these changes will not significantly increase the probability of an accident previously evaluated. To the extent practicable, these proposed changes were developed consistent with the changes approved by the NRC when developing NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4'', with the intent of having this relocated information controlled in other plant documents subject to 10CFR50.59 provisions. Since the plant systems associated with these proposed changes will still be capable of: 1) meeting all applicable design basis requirements; and 2) retain the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

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

Relocation of Technical Specification Table 3.8.4.2-1 to the UFSAR will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve: 1) no hardware changes; 2) no changes to the operation of any systems or components; and 3) no changes to existing structures, systems or components, there can be no impact on the occurrence of any accident. To the extent practicable, these proposed changes were developed consistent with the changes approved by the NRC when developing NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4'', with the intent of having this relocated information controlled in other plant documents subject to 10CFR50.59 provisions. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Relocation of Technical Specification Table 3.8.4.2-1 to the UFSAR is consistent, to the extent practicable, with the changes approved by the NRC when developing NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4''. The MOV thermal overload protection table will reside in the UFSAR and will ensure that the associated MOVs will be capable of performing their intended safety functions. Any changes to this UFSAR table will be subject to the provisions of 10CFR50.59 and a separate safety evaluation would be developed to support any proposed changes that would subsequently be made. Therefore, the changes contained in this request do not result in a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz

Tennessee Valley Authority, Docket Nos. 50-259, 50-260 and 50-296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: June 2, 1997 (TS 387)

Description of amendment request: The proposed amendment allows continued plant operation with a single reactor recirculation loop in service. The Nuclear Regulatory Commission has previously determined single loop operation is generically acceptable as set forth in Generic Letter 86-09, ''Technical Resolution of Generic Issue B-59-(N-1) Loop Operation in BWRs [boiling water reactors] and PWRs [pressurized-water reactors].'' Single loop operation is also recognized as a standard mode of operation in the BWR/4 Improved Standard TS.

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

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

An analysis of the limiting operational transients has been performed by GE [General Electric] for BFN as documented in NEDO-24236 to demonstrate adequate margin to the Safety Limit Minimum Critical Power Ratio (SLMCPR). In addition, SLO [single loop operation] has been specified as a operating option for the transient and accident evaluations performed as part of the cycle-specific core reload analyses for Units 2 and 3 which ensure that operating limit Minimum Critical Power Ratios (OLMCPRs) for the current fuel types are established that maintain required margin to the fuel cladding safety limit. A cycle-specific analysis with SLO will be performed for Unit 1 prior to restart and experience indicates similar results are expected as those for Units 2 and 3.

A review of the values used in the statistical analysis used in the basis of the fuel cladding safety limit determined that, due to increased uncertainties in total core flow readings and Traversing In-Core Probe (TIP) readings during SLO, an increase in the SLMCPR of .02 is bounding when in SLO. Therefore, while operating in single-loop mode, an additional .02 is added to the OLMCPR which maintains the same margin to the fuel cladding safety limit as that established for two-loop operation. This is a conservative approach because the two-loop transients have been shown to be more severe than the equivalent single-loop events and, therefore, the OLMCPRs established for two-loop operation would always be bounding. Thus, the margin of safety for fuel clad integrity is assured and the probability or consequences associated with reactor transients is not increased for SLO.

SLO results in backflow through the jet pumps in the inactive recirculation loop which perturbs the relationship between the core flow and recirculation drive flow on which the flow biased Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) setpoint equations are based. To compensate, the proposed TS [Technical Specification] changes modify the setpoint equations to correct for one-loop operation. With this adjustment, the setpoint equations preserve the original relationship between the setpoints and the effective recirculation drive flow such that the consequences of a RWE [rod withdrawal event] in SLO are bounded by the cycle-specific RWE analyses. Therefore, these changes do not increase the probability or consequences of the RWE transient previously evaluated.

Average Planar Linear Heat Generation Rate (APLHGR) limits are established to ensure the acceptance criteria for fuel and Emergency Core Cooling Systems established in 10 CFR 50.46 are met. A SLO Loss of Coolant Accident (LOCA) analysis was performed using the SAFER/GESTR computer code as documented in NEDC-32484P, Revision 1, ''Browns Ferry Nuclear Plant, Units 1, 2, and 3, SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis.''

The LOCA [loss of cooling accident] results for SLO using SAFER/GESTR showed that, with the application of an APLHGR multiplier as proposed in the TS change, the LOCA peak clad temperature for SLO will always be lower than that for limiting design basis pipe break for two-loop operation. An APLHGR multiplier of 0.9 is applicable for all current fuel types being used. This multiplier is documented in each cycle-specific reload analysis and included in the COLR [core operating limits report]. NEDC-32484P Revision 1 also concludes that the design basis accident (large breaks) are more affected than small break sequences and, therefore, the large break results are bounding for SLO.

The Recirculation Pump Seizure event in SLO was evaluated in NEDO-24236 and shown to be a non-limiting event. This conclusion is also supported by GE analyses on other BWRs.

In summary, based on the above discussion, the proposed changes for SLO do not increase the probability or consequences of an accident previously evaluated.

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

Although the proposed change allows extended operation in a configuration that was previously allowed for a limited period, analysis has shown (as described in item A above), that operation with one recirculation pump out-of-service is within existing analyses based on the proposed TS requirements. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to operate in single-loop recirculation mode has been analyzed in accordance with established transient and accident methodologies, and margins of safety for the design basis accidents and transients analyzed in Chapter 14 of the BFN UFSAR [updated final safety analysis report] have not been significantly reduced. The basis for this conclusion is outlined in item A above. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Athens Public Library, 405 E. South Street, Athens, Alabama 35611

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: June 9, 1997

Description of amendment request: The amendment proposes to update the Technical Specifications, Section 6.0, to add a reference to NRC-approved methodologies which will be used to validate or generate the operating limits in the Vermont Yankee Core Operating Limits Report.

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 will not involve any significant increase in the probability or consequences of an accident. The change updates the Technical Specifications to include [an] NRC approved method reference to allow calculation of thermal hydraulic stability limits. It does not affect plant operation and will not weaken or degrade the facility.

2. The proposed change will not create the possibility of a new or different kind of accident since the change is administrative. No physical alterations of the plant, setpoint changes, or operating conditions are proposed.

3. The proposed change will not involve a significant reduction in a margin of safety. The change involves an update to the Administrative Controls in Section 6.0 of the Technical Specifications by adding a reference to NRC approved methods. This administrative change does not alter plant safety margins.

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

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

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

NRC Project Director: Ronald B. Eaton, Acting

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.

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

Date of amendment request: June 9, 1997

Description of amendment request: The proposed amendments authorize a revision to the realistic dose values for the process gas system rupture in Section 15.0 of the Byron/Braidwood (B/B) Updated Final Safety Analysis Report (UFSAR). During preparation of a UFSAR change package, ComEd discovered that the Final Safety Analysis Report (FSAR) had not been updated to correct an error from the previous revision of the dose calculation. Since the correct dose value is greater than that previously reported, the consequences of the accident had increased, and an unreviewed safety question resulted.

Date of publication of individual notice in Federal Register: July 10, 1997 (62 FR 37079).

Expiration date of individual notice: August 11, 1997 (as corrected (62 FR 39282)).

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

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: June 27, 1997, as supplemented by letter dated July 2, 1997 The supplemental letter provided clarifying information and did not change the initial proposed no significant hazards consideration determination.

Brief description of amendment request: These amendments clarify, in the technical specifications (TSs) for each unit, the methodology used to satisfy surveillance requirements for the laboratory analysis of activated carbon (charcoal) samples from the standby gas treatment system (SGTS) and the control room emergency outside air supply system (CREOASS). The specific changes are made to Sections 4.6.5.3.b.2 and 4.6.5.3.c for the SGTS and to Sections 4.7.b.2 and 4.7.2.c for the CREOASS, to include a reference to American Society for Testing Materials (ASTM), ''Radioiodine Testing of Nuclear-Grade Gas Phase Adsorbents,'' ASTM D3803-79. Date of publication of individual notice in **Federal Register:** July 8, 1997 (62 FR 36580)

Expiration date of individual notice: August 7, 1997

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: April 14, 1997

Brief description of amendments: The amendments revise Technical Specification (TS) 3/4.3.8, ''Feedwater/Main Turbine Trip System Actuation Instrumentation'' by changing the minimum channels required from three to four. This change reflects a modification that is being installed to add an auxiliary contact to the trip system logic. In addition, the amendments revise the TS action statement for inoperable channels to be consistent with the Improved Standard Technical Specifications and to account for the additional channel.

Date of issuance: July 29, 1997

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

Amendment Nos.: 119 and 104

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33120). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 29, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 27, 1997, as supplemented July 7, 1997

Brief description of amendment: The amendment revises the Palisades Plant license and technical specifications to reflect the licensee's name change from ''Consumers Power Company'' to ''Consumers Energy Company.''

Date of issuance: July 21, 1997

Effective date: July 21, 1997

Amendment No.: 176

Facility Operating License No. DPR-20: Amendment revised the license and the technical specifications.

Date of initial notice in **Federal Register:** April 23, 1997 (62 FR 19828) The July 7, 1997, letter provided supplementary information within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 1997. No significant hazards consideration comments received: No.

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

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: May 27, 1997

Brief description of amendments: The amendments delete Section 4.7.13.3.a.2 of each unit's Technical Specifications, regarding the minimum volume and boron concentration of borated water available to the Standby Makeup Pump of the Standby Shutdown System.

Date of issuance: July 21, 1997

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

Amendment Nos.: 160 and 152

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33121) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

Date of application for amendment: February 17, 1997, as revised May 1, 1997.

Brief description of amendment: Changes to Technical Specification (TS) to implement 10 CFR 50, Appendix J Option B relating to containment leakage tests.

Date of issuance: July 24, 1997

Effective date: July 24, 1997

Amendment No.: 156

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

Date of initial notice in **Federal Register:** February 28, 1997 (62 FR 9214), as superseded June 4, 1997 (62 FR 30632) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629

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

Date of application for amendment: April 28, 1997

Brief description of amendment: Technical Specification (TS) 3.7.6 requires that flood protection be provided for the service water pump cubicles and components when the water level exceeds a specific value. The amendment (1) adds the closing of the service water pump cubicle sump drain valves to the TS, (2) revises the wording of the action statement to be consistent with the limiting condition for operation, and (3) revises the associated Bases section.

Date of issuance: July 28, 1997

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

Amendment No.: 144

Facility Operating License No. NPF-49: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30636) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 28, 1997. 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 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385

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

Date of application for amendment: January 23, 1997, as supplemented January 28, March 4, June 19, July 2, July 16 (2 letters), July 21, and July 25, 1997

Brief description of amendment: The amendment documents the staff's review and approval of the apparent unreviewed safety questions (USQs) associated with (1) the updated analysis of the design-basis accident (DBA) containment temperature and pressure response, and (2) the reliance on containment pressure to compensate for the potential deficiency in net positive suction head (NPSH) for the emergency core cooling system (ECCS) pumps during a DBA with the worst case scenario assumptions. The amendment also authorizes the licensee to change the Technical Specification bases and the Updated Safety Analysis Report, to reflect the reliance of containment pressure to compensate for the potential deficiency in NPSH for the ECCS pumps following a DBA.

Date of issuance: July 25, 1997

Effective date: July 25, 1997. Implementation shall be as specified in Appendix C to the license.

Amendment No.: 98

Facility Operating License No. DPR-22: Amendment revised the license and the licensee's updated safety analysis report.

Date of initial notice in **Federal Register:** February 12, 1997 (62 FR 6576) The June 19, 1997, submittal, expanded the scope of the initial submittal dated January 23, 1997, and therefore, another notice was issued in **Federal Register** on June 24, 1997 (62 FR 34086). The July 2, July 16 (2 letters), July 21, and July 25, 1997, submittals provided additional clarifying information within the scope of the application and did not change the NRC staff's proposed no significant hazards considerations determination that was based on the June 19, 1997, submittal. Therefore, renoticing was not warranted. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 1997. No significant hazards consideration comments received: No.

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

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: June 27, 1997, as supplemented by letter dated July 2, 1997 The supplemental letter provided clarifying information and did not change the initial proposed no significant hazards consideration determination.

Brief description of amendments: These amendments clarify, in the technical specifications (TSs) for each unit, the methodology used to satisfy surveillance requirements for the laboratory analysis of activated carbon (charcoal) samples from the standby gas treatment system (SGTS) and the control room emergency outside air supply system (CREOASS). The specific changes are made to Sections 4.6.5.3.b.2 and 4.6.5.3.c for the SGTS and to Sections 4.7.b.2 and 4.7.2.c for the CREOASS, to include a reference to American Society for Testing Materials (ASTM), ''Radioiodine Testing of Nuclear-Grade Gas Phase Adsorbents,'' ASTM D3803-79.

Date of issuance: July 30, 1997

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

Amendment Nos.: 167 and 141

Facility Operating License Nos. NPF-14 and NPF-22: The amendments revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 36580). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination by July 22, 1997. No comments have been received. The notice also provided an opportunity to request a hearing by August 7, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment. On July 9, 1997, the NRC staff issued a Notice of Enforcement Discretion in order to delay enforcement of the current, subject, TS requirements until the NRC could take formal action on the July 2, 1997, application. The Commission's related evaluation of the amendments, finding of exigent circumstances, consultation with the State of Pennsylvania, and final no significant hazards consideration determination are contained in a Safety Evaluation dated July 30, 1997.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701

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

Date of application for amendment: February 11, 1997.

Brief description of amendment: This amendment changes the Hope Creek Technical Specification (TS) Sections 3/4.8.1, ''A.C. Sources,'' 6.8, ''Procedures and Programs,'' and the Bases for Section 3/4.8, ''Electrical Power Systems,'' to include: 1) the relocation of existing surveillance requirements related to diesel fuel oil chemistry; 2) the introduction of a new program under TS 6.8.4.e, ''Diesel Fuel Oil Testing Program≥; 3) revisions to the TS Bases for Section 3/4.8 to incorporate information associated with the TS changes; and 4) editorial changes to implement required corrections.

Date of issuance: July 24, 1997

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

Amendment No.: 100

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 26, 1997 (62 FR 14469) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 3, 1997, as supplemented by letter dated May 5, 1997

Brief description of amendment: This amendment changes Hope Creek TSs as follows: (1) TS 3/4.3.1, ''Reactor Protection System Instrumentation,'' TS 3/4.3.2, ''Isolation Actuation Instrumentation,'' and TS 3/4.3.3, ''Emergency Core Cooling System Actuation Instrumentation,'' to include additional information concerning response time testing; (2) TS 4.0.5 to reference inservice inspection and test requirements; (3) TS 3/4.6.1, ''Primary Containment,'' and associated Bases to reflect a design modification; (4) TS 3/4.7.7, ''Main Turbine Bypass System,'' to specify a new operability requirement; and (5) the Bases for TS 3/4.8, ''Electrical Power Systems.''

Date of issuance: July 24, 1997

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

Amendment No.: 101

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33131) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 11, 1997, as supplemented on May 1, June 12, and July 23, 1997

Brief description of amendments: The amendments add a new Technical Specification, 3/4.7.10, ''Chilled Water System - Auxiliary Building Subsystem,'' and an associated Bases section to address the support function this system provides to other necessary safety systems.

Date of issuance: July 29, 1997

Effective date: Unit 1 to be implemented prior to entering Mode 6 from the current unit outage; Unit 2 as of its date of issuance, to be implemented within 10 days of issuance.

Amendment Nos.: 199 and 182

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

Date of initial notice in **Federal Register:** March 12, 1997 (62 FR 11497) The licensee's supplemental letters provided additional information that did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 29, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 23, 1996, as supplemented

December 11, 1996, January 31, February 10 and 24, March 11, April 4 and 11, May 28, June 26, and July 15, 1997.

Brief description of amendment: The amendment changes the Watts Bar Nuclear Plant, Unit 1, Technical Specifications (TS) to increase the spent fuel storage capacity from 484 fuel assemblies to 1610 fuel assemblies and to increase the initial enrichment of the fuel to be stored in the spent fuel storage racks from 3.5 weight percent (wt%) to 5.0 wt%. This modification also changes the center-to-center spacing of stored fuel assemblies and reflects the use of burnup credit rack modules to be installed peripherally along the pool walls.

The amendment, as proposed by the licensee, would also involve the installation of spent fuel racks in the spent fuel cask pit for 225 storage spaces thus increasing the total WBN spent fuel storage capacity to 1835 spent fuel assemblies. The licensee proposed to provide an impact shield that would be placed over the fuel in the cask pit when heavy loads are moved near or across the cask pit area. The staff is continuing its review of this aspect of the licensee's proposal. Accordingly, this amendment authorizes the reracking and usage of the main spent fuel pool, as proposed for a total of 1610 spent fuel spaces. However, it does not authorize the installation of storage racks or storage of spent fuel in the spent fuel cask pit. The staff's review of that aspect of the licensee's application will be addressed by further correspondence.

Date of issuance: July 28, 1997

Effective date: July 28, 1997

Amendment No.: 6

Facility Operating License No. NPF-90: Amendment revises the TS.

Date of initial notice in **Federal Register:** April 2, 1997 (62 FR 15733) The April 4, and 11, May 28, June 26 and July 15, 1997 letters provided clarifying informaion that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in an environmental assessment dated April 7, 1997, and a Safety Evaluation dated July 28, 1997. No significant hazards consideration comments received: None

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

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

Date of application for amendments: November 9, 1987, as supplemented March 31, 1988, June 8, 1992, and February 4, 1997

Brief description of amendments: These amendments reformat the operability and surveillance requirements for the intermediate range channels.

Date of issuance: July 30, 1997

Effective date: July 30, 1997

Amendment Nos.: 206 and 187

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33136) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 30, 1997. No significant hazards consideration comments received: No.

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

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

Date of amendment request: February 17, 1997

Brief description of amendment: The amendment revises the technical specifications to move Table 3.6-1, "Containment Isolation Valves" to Wolf Creek Generating Station procedures. In addition, the technical specifications have been modified to remove all references to Table 3.6-1. This change is in accordance with the guidance provided in Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," dated May 6, 1991.

Date of issuance: July 23, 1997

Effective date: July 23, 1997, to be implemented within 30 days from the date of issuance.

Amendment No.: 108

Facility Operating License No. NPF-42: The amendment revised the Technical Specifications and the Operating License.

Date of initial notice in **Federal Register:** April 23, 1997 (62 FR 19838) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 23, 1997. No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 6th day of August, 1997.

For the Nuclear Regulatory Commission

Jack W. Roe,

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

[Doc. 97–21244 Filed 8–12–97; 8:45 am]

BILLING CODE 7590–01–F

PENSION BENEFIT GUARANTY CORPORATION

Proposed Submission of Information Collection for OMB Review; Comment Request; Payment of Premiums

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of intention to request extension of OMB approval.

SUMMARY: The Pension Benefit Guaranty Corporation ("PBGC") intends to request that the Office of Management and Budget ("OMB") extend approval, under the Paperwork Reduction Act, of the collection of information under its regulation on Payment of Premiums (29 CFR part 4007), including Form 1–ES, Form 1, and Schedule A to Form 1, and related instructions (OMB control number 1212–0009; expires February 28, 1998). The collection of information also includes a certification (on Schedule A) of compliance with requirements to provide certain notices to participants under the PBGC's regulation on Disclosure to Participants (29 CFR part 4011), and surveys of plan administrators to assess compliance with those requirements. This notice informs the public of the PBGC's intent and solicits public comment on the collection of information.

DATES: Comments should be submitted by October 14, 1997.

ADDRESSES: Comments may be mailed to the Office of the General Counsel, suite 340, Pension Benefit Guaranty Corporation, 1200 K Street, NW., Washington, DC 20005–4026, or delivered to that address between 9 a.m. and 4 p.m. on business days. Written comments will be available for public inspection at the PBGC's Communications and Public Affairs Department, suite 240 at the same address, between 9 a.m. and 4 p.m. on business days.

Copies of the collection of information may be obtained without charge by writing to the PBGC's Communications and Public Affairs Department at the address given above