NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Pub. L. 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Pub. L. 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 June 22, 1998, through July 2, 1998. The last biweekly notice was published on July 1, 1998 (63 FR 35986).

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

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

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

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

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

By August 14, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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: May 29, 1998.

Description of amendment request: The proposed amendment would revise the technical specifications to credit the automatic function of the pressurizer power operated relief valves (PORVs) to provide mitigation for inadvertent safety injection at power accident. The limiting condition for operation and surveillance requirements for the PORVs would also be revised.

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

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

The changes to the Technical Specification (TS) Limiting Condition for Operation (LCO), Surveillance Requirements, and Bases do not involve an increase in the probability or consequences of the Inadvertent Operation of Emergency Core Cooling System (Spurious SI) at Power transient. Crediting the PORVs in the maximum pressurizer overfill case for this transient does not increase the probability of the occurrence of the transient since the automatic function of the PORVs for Reactor Coolant System (RCS) pressure control is not an initiator for the Spurious SI at Power transient. This change allows for the NRC Standard Review Plan (NUREG–0800) acceptance criteria to be met for the Spurious SI at Power transient, ensuring that the consequences of this transient remain within acceptable levels.

As documented in various Safety Evaluation Reports (SERs) from the NRC, the overpressure protection function of the PORVs was not originally considered to be a safety related function. In response to Generic Issue 70, the NRC performed a regulatory analysis related to PORV and block valve reliability in Pressurized Water Reactor (PWR) plants. This regulatory analysis is documented in NUREG–1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70, Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants," where the NRC staff concluded that it was not cost effective to backfit non-safety related PORVs to upgrade them to safety related status to perform safety related functions. The safety related functions were those detailed in Section 2.1 of NUREG–1316 and any other safety related function identified in the future. As an example, the PORVs are credited for the cold overpressure protection function of the reactor pressure vessel during low temperature operations. The analysis documented in this License Amendment request demonstrates that the PORVs provide an acceptable level of quality and performance to allow them to be credited to mitigate the consequences of the Spurious SI at Power transient documented in Byron and Braidwood Updated Final Safety Analysis Report (UFSAR) Section 15.5.1. The PORVs are equipped with safety related actuators and safety related accumulator tanks which maintain valve function during a loss of instrument air. The position indication and control switches in the Main Control Room (MCR) are safety related. All pressurizer PORV open/close functions and circuitry are supplied with uninterruptible Class 1E power supplies. The automatic portion of the PORV circuitry which processes the high pressurizer and high RCS pressure at low temperature is designated non-safety related and is isolated from the safety related portions of the circuitry by safety related interposing relays which actuate on a faulted condition. However, both Byron and Braidwood Stations have implemented modifications for both Units 1 and 2, which ensure that automatic control of both PORVs is available during loss of offsite power conditions. In addition, the PORV function is monitored within the scope of the Maintenance Rule Program and the postulated failure of the PORV automatic function does not result in unacceptable risk.

The probability of a Spurious SI at Power transient is not affected by this proposed change and the above analysis demonstrates that the PORVs will adequately function in automatic mode to mitigate the consequences of the transient. As such, there are no changes in the type or amount of any effluent released offsite as a result of this change. Therefore, based on this evaluation, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change does not create the possibility of a new or different accident from any accident previously evaluated. This change would specifically allow for the PORV automatic function to be credited in Modes 1, 2, and 3 for the Spurious SI at Power transient only. This change allows for added assurance that the acceptance criteria as documented in the NRC Standard Review Plan (NUREG–0800) for ANS Condition II transients will be met. The acceptance criteria of concern is that a Condition II transient must not lead to an event (Condition III or IV) of more significant consequences without additional failures occurring. The PORV automatic function is to be credited with mitigating the maximum pressurizer overfill case for the Spurious SI at Power transient. This case has the acceptance criteria that the pressurizer must not go water solid prior to RCS pressure reaching the setpoint of the pressurizer safety relief valves (PSRVs). This conservative acceptance criteria is based on the fact that the PSRVs are not qualified to pass subcooled water and reseat, thereby creating a concern for an uncontrolled release path from the RCS. This proposed change helps ensure that the acceptance criteria for this accident are met. There is a small probability that the PORV function, either automatic or manual, would not successfully mitigate this transient due to the failure of one or both PORVs. However, the low likelihood of a total failure of the PORV function during the Spurious SI at Power transient does not create a new accident because a similar scenario is already addressed by UFSAR Section 15.6.1, ''Inadvertent Opening of a Pressurizer Safety or Relief Valve.'' The UFSAR analysis for the Section 15.6.1 ANS Condition II transient indicates that the radiological consequences of this transient are significantly less than that of a LOCA and are therefore, acceptable. The same arguments for radiological consequences apply to the Spurious SI at Power transient in the event the PORV automatic function fails and water relief occurs through the PSRVs.

The proposed change to the LCO requirements in TS Section 3/4.4.4 would allow for the PORV block valve to be closed but remain energized in the event a PORV was considered inoperable due to the automatic actuation circuitry. Currently, the PORV block valve is closed but remains energized only if a PORV is considered inoperable due to excessive seat leakage. The proposed change would extend the allowance to include the circumstance where the PORV was inoperable due to the automatic actuation circuitry. This allows a PORV to remain functional in the manual mode for other safety related functions consistent with the discussion contained in NRC NUREG–1316. However, this revised LCO requirement would not represent a new failure mode or accident over what has been previously evaluated.

In summary, the proposed changes documented in this TS amendment to credit the automatic PORV function and to revise the TS LCO requirements for PORV inoperability do not create the potential for any new or different accidents from what was previously evaluated.

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

The current TS bases do not credit the function of the pressurizer PORVs for any Mode 1, 2, or 3 transients. This change would allow for the PORV automatic function to be credited for the Spurious SI at Power transient only. This does not represent a significant reduction in the margin of safety. This change would allow for the conservative acceptance criteria for the current UFSAR design analysis to be met. The PORVs are reliable and are maintained in a manner consistent with their proposed safety related function to mitigate the Spurious SI at Power transient. This proposed change would not result in a significant increase in risk or consequences, and therefore, does not involve a significant reduction in the margin of safety.

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

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.

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: May 28,1998.

Description of amendment request: The proposed amendment proposes changes to the Final Safety Analysis Report (FSAR) to include a description of the use of Generic Letter (GL) 87–11, ''Relaxation in Arbitrary Intermediate Pipe Rupture Requirements,'' and NUREG/CR–2913, ''Two-Phase Jet Loads,'' as a part of the approved licensing basis and design basis for Crystal River Unit 3. GL 87–11 will be used to determine where high energy line breaks (HELB) are postulated to occur for high energy lines located inside the Reactor Building (RB) and analyzed in accordance with the guidelines described in USAS B31.1.0–1967, ''USA Standard Code for Pressure Piping, Power Piping.'' NUREG/CR–2913 will be used to determine the effects of the resultant jet impingement from postulated Reactor Coolant System (RCS) piping ruptures on safety-related systems, structures, and components (SSCs).

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

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

The use of new design methodologies for determining postulated break locations of RCS piping and other high energy lines located inside containment, and the dynamic effects of postulated ruptures of RCS piping on SSCs required for safe shutdown or accident mitigation, does not impact the design of these high energy lines such that previously analyzed ruptures would now be more likely to occur. The approval of the license amendment will not result in an actual modification to RCS piping or other high energy lines which would reduce their design capabilities to maintain pressure boundary integrity during normal operating and accident conditions. By using these new design methodologies, protection of SSCs required for accident mitigation is assured. Protection of SSCs required for accident mitigation will continue to be assured by use of these well-defined design methodologies if modifications to those SSCs are implemented in the future. Therefore, there will be no reduction in the capability of those SSCs in limiting the consequences of previously evaluated accidents, and the proposed amendment does not significantly increase the probability or consequence of an accident previously evaluated.

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

The use of new design methodologies for determining postulated break locations of RCS piping and other high energy lines located inside containment, and the dynamic effects of postulated ruptures of RCS piping on SSCs required for safe shutdown or accident mitigation, does not impact the design of these high energy lines such that previously unanalyzed ruptures would now occur. The approval of the license amendment will not result in an actual modification to RCS piping or other high energy lines which would reduce their design capabilities to maintain pressure boundary integrity during normal operating and accident conditions. By using these new design methodologies, the current design of RCS piping and other high energy lines located inside containment can be shown to include sufficient design margin to prevent unanalyzed ruptures from occurring. Therefore, use of these design methodologies instead of the previous licensing basis requirements cannot create the possibility of a new or different kind of accident.

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

The use of new design methodologies for determining postulated break locations of RCS piping and other high energy lines located inside containment, and the dynamic effects of postulated ruptures of RCS piping on SSCs required for safe shutdown or accident mitigation, does not impact the design of these high energy lines such that unanalyzed ruptures would now occur, and cannot create a reduction in the margin of safety for those ruptures of high energy lines previously analyzed. The approval of the license amendment will not result in an actual modification to RCS piping or other high energy lines which would reduce their design capabilities to maintain pressure boundary integrity during normal operating and accident conditions. By using these new design methodologies, protection of SSCs required for accident mitigation is assured. Protection of SSCs required for accident mitigation will continue to be assured by use of these well-defined design methodologies if modifications to those SSCs are implemented in the future. Therefore, the capability of those SSCs to limit the consequences of previously evaluated accidents at levels below the approved acceptance limits will continue to be assured. As a result, use of these design methodologies instead of the previous licensing basis and design basis requirements cannot significantly reduce the existing 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.

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

Date of amendment request: May 28, 1998.

Description of amendment request: Revision of Technical Specification (TS) 4.5.A.1 such that the first Type A test required by the primary containment leakage rate testing program be performed during refueling outage 18 rather than refueling outage 17.

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

The proposed changes does not alter the design, function or manner of operation of any structures, systems or components. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents.

NUREG–1493 found that the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of the containment. The major contributor to the total identified leakage from Primary Containment comes from Type B and C tested components. Only a small portion of the total leakage is detectable soley through Type A testing. The leaks that have been found by Type A tests have been only marginally above existing requirements. In addition, Oyster Creek has two means (monitoring nitrogen use and performing torus to drywell vacuum breaker leak tests) of detecting gross containment leakage. The proposed change does not alter the requirements to perform Type B and C testing in accordance with the Primary Containment Leakage Rate Testing Program and does not affect the ability of the facility to mitigate the consequences of an accident.

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

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

Deferring the Type A test for an operating cycle does not alter the design, function or manner of operation of any structures, systems or components. The proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents nor does it introduce any new mechanisms which could contribute to the creation of a new or different kind of accident than previously evaluated.

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

The proposed change does not alter the design, function or manner of operation of any structures, systems or components. The proposed change does not impact the primary containment system's ability to provide a barrier against the uncontrolled release of fission products in the event of a break in the reactor coolant system nor does the proposed change impact the primary containment accident leak rate. In addition, NUREG–1493's Summary of Technical Findings states ''Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.'' Therefore, the proposed TS 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: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Attorney for licensee: Ernest L. Blake, Jr., Esquire. Shaw, Pitman, Poets & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cecil O. Thomas.

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: June 10, 1998.

Description of amendment request: The proposed revision to the Millstone Unit 3 licensing basis would address post-accident mitigation activities, vital area access travel routes, and time. NNECO determined that the Final Safety Analysis Report (FSAR) description of post-accident vital area routing was out of date because the radiological control area boundary fence created an access problem on the designated routes to the hydrogen recombiner and fuel building. The FSAR change would revise the routes to accommodate the fence location and allow for the time to unlock gates.

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 10 CFR 50.92 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 10 CFR 50.92(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.

Final Safety Analysis Report (FSAR) Section 12.3.1.3.2, Post-accident access to vital areas, and its associated Figures and Tables are being updated. The current FSAR descriptions are out of date and as such do not include all required post-accident actions. Therefore, this FSAR change adds actions to those listed in the FSAR as well as incorporating the recalculation of the doses associated with the required post-accident actions. The dose calculations utilize the appropriate post-accident source terms, area access requirements and stay times, including the appropriate routes to the areas. The calculations show that for all design basis required post-accident actions the calculated dose to the Operators/Emergency workers performing those actions remains below the 5 rem criterion of General Design Criteria (GDC) 19. The revision to the FSAR provides the required post-accident required operator actions. Changing the FSAR to include the current post-accident vital access requirements and associated information for the supporting dose calculations [cannot] cause an accident. In addition, the calculated dose to the Operators/Emergency workers for all design basis required actions is below the GDC 19 limit of 5 rem.

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

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

The change is to the calculated post-accident vital access dose analyses and the FSAR description of that analyses. No new procedural Operator/Emergency worker actions are associated with the change.

However, since the information in the FSAR was outdated, there are Operator/Emergency actions being added to the FSAR. Dose calculations associated with those actions have been performed utilizing the appropriate assumptions with respect to source terms, vital area access travel routes and stay times, and times when the post-accident actions would be performed. The analyses confirmed that the calculated doses associated with all required post-accident actions are less than the 5 rem limit of GDC 19. There are no changes to the Emergency Operating Procedures associated with this change.

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 dose calculations confirm that the calculated dose associated with all design basis post-accident Operator/Emergency worker actions is below the limit of 5 rem of GDC 19. There is one action, initiation of hydrogen purge, for which the calculated dose to the Operator/Emergency worker exceeds 5 rem. This action is a backup means of limiting the hydrogen concentration inside containment post-accident. This action would only be performed for multiple failures which would disable both trains of the safety-grade hydrogen recombiner system. As such this action is not a required design basis action and does not need to meet the 5 rem limit. The calculated dose for this action is below the 25 rem limit that is specified in the Station Emergency Plan for severe accident mitigation actions.

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, Connecticut.

NRC Deputy Director: Phillip F. McKee.

PECO Energy Company, Public Service Electric and Gas Company, Demarva 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: February 4, 1998.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) Surveillance Requirement (SR) concerning Secondary Containment doors at Peach Bottom Atomic Power Station, Units 2 and 3.

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

TS SR 3.6.4.1.2 will be revised to require either all inner or outer secondary containment access doors to be closed in each air lock. This revision will not adversely affect the ability of the Secondary Containment to mitigate the radiological consequences of a Loss-of-Coolant Accident or fuel handling accident, and does not involve a significant increase in the probability or consequences of an accident previously evaluated. During those times that one or more inner (or outer) doors are open, the closed outer (or inner) doors will serve as the Secondary Containment boundary.

Allowing certain inner or outer Secondary Containment access doors in an air lock to be open does not compromise the design of the Secondary Containment. No commitment is made in the UFSAR to consider the single failure of passive structural components such as Secondary Containment doors. As discussed in Section 1.5 of the UFSAR, ''* * * Essential safety actions shall be carried out by equipment of sufficient redundance and independence that no single failure of *active* components can prevent the required actions''. The same UFSAR section goes on to state that, ''For systems or components to which IEEE–279 (1968) is applicable, single failures of *passive electrical components* are considered, as well as single failures of active components, in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.'' Therefore, based on this UFSAR discussion, it is concluded that failure of outer (inner) secondary containment doors need not be postulated with the inner (outer) door being open.

The performance of the Secondary Containment and the Standby Gas Treatment System is unaffected by this activity. Surveillance testing will prove the capability to maintain Secondary Containment with only inner or only outer doors closed. This change will not result in greater or more frequent loading of Secondary Containment doors, and does not result in changes that impact the reliability of the Secondary Containment and the Standby Gas Treatment System.

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

The Secondary Containment, in conjunction with the Standby Gas Treatment System, provides the means for mitigating the radiological consequences of an accident. The configuration of the Secondary Containment has no effect on accident initiators which lead to a new or different kind of accident. This change will not involve any changes to plant systems, structures, or components which could act as new accident initiators. The design, function, and reliability of Secondary Containment and the Standby Gas Treatment System are also not impacted by this change. Therefore, this change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

No margins of safety are reduced as a result of this change to the TS. No safety limits will be changed as a result of this TS change. The Secondary Containment will continue to perform its intended safety function of limiting the ground level release of airborne radioactive materials and to provide a means for controlled elevated release of the building atmosphere so that off-site doses from the postulated design basis accidents are below the limits of 10 CFR 100. The design and reliability of the Secondary Containment are also not impacted as a result of this change.

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: Robert A. Capra.

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

Date of amendment request: November 13, 1997.

Description of amendment request: The proposed amendment will reduce the maximum test interval from 1 year to 6 months for the test frequency of the main turbine stop and control valves (TS & CVs) in Table 4.1–3 and add a footnote.

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

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

Response

The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change increases the frequency of testing of the TS & CVs by reducing the maximum allowable test interval. The maximum test interval is reduced from one year to six months. Thus, the proposed change will make the maximum test interval more conservative. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve the addition of any new or different type of equipment, nor does it involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the Final Safety Analysis Report.

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

Response

The proposed license amendment does not involve a significant reduction in a margin of safety. The proposed change does not adversely affect performance of any safety related system or component, instrument operation, or safety system setpoints and does not result in increased severity of any accidents considered in the safety analysis. The proposed change does not reduce the frequency of testing of these valves but updates the methodology for determination of the test frequency and reduces the maximum test interval from one year to six months. It establishes a more conservative acceptance criteria of 5.0×10^{-6} per year than the NRC acceptance criteria of 1.0×10^{-5} for a turbine missile event. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: S. Singh Bajwa, Director.

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: June 16, 1998.

Description of amendment request: The proposed amendment would relocate the Safety Review Committee review, audit and related record keeping requirements from the Technical Specifications (TSs) to Chapter 17 of the Final Safety Analysis Report (FSAR) (i.e., Quality Assurance Program).

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

(1) 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 relocation of the SRC [Safety Review Committee] review, audit, and related record keeping requirements from the TS to the FSAR does not alter the performance or frequency of these activities. Future changes to the QA [Qualify Assurance] program, located in Chapter 17 of the FSAR, which constitute a reduction in commitments, are governed by 10 CFR 50.54(a). Therefore, sufficient controls for these requirements exist and these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

(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 involve the relocation of SRC requirements from the TS to the FSAR. Relocation of these requirements does not affect plant equipment or the way the plant operates. The reviews, audits, and record keeping will continue to be performed in the identical manner as they are currently being performed. Therefore, the proposed revisions 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 relocate SRC review, audit and related record keeping requirements from the TS to the FSAR. These requirements are not being altered by this relocation. The reviews, audits, and record keeping will continue to be performed in the identical manner as they are currently being performed. Any changes to these requirements which constitute a reduction in commitments will be processed in accordance with 10 CFR 50.54(a). Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Project Director: S. Singh Bajwa.

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

Date of application for amendments: April 30, 1998 (TS 98–01).

Brief description of amendments: The amendments would change the Sequoyah (SQN) Technical Specifications (TSs) to allow surveillance testing of the reactor coolant system (RCS) pressurizer power-operated relief valves (PORVs) in Modes 3, 4, and 5.

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

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the TSs, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

The possibility of occurrence or the consequences for an accident or malfunction of equipment is not increased as the test conditions for the PORVs in Mode 5 are representative conditions based on a steam bubble being present, and testing in this mode is more conservative, if RCS pressure is less, since there is less fluid force to aid the solenoid force in opening the valve. Testing in Modes 3 and 4 was the initial request of GL [Generic Letter] 90–06. No changes are proposed to operation of the PORV block valves. Offsite dose consequences are unchanged by this request.

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

A possibility for an accident or malfunction of a different type than any evaluated previously in SQN's Final Safety Analysis Report is not created; nor is the possibility for an accident or malfunction of a different type. A new test method is not required. No new failure modes are introduced.

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

The margin of safety has not been reduced for testing in Mode 5 since the proposed test conditions are equal to or more conservative, if RCS pressure is less, than those currently in use with existing SRs [surveillance requirements]. Testing in Modes 3 and 4 was the initial request of GL 90–06. The results of the accident analysis remain unchanged by this request.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of application for amendments: June 26, 1998 (TS 98–02).

Brief description of amendments: The amendments would change the Sequoyah Nuclear Plant (SQN) Technical Specifications (TS) and their Bases to lower the specific activity of the primary coolant from 1.0 microcurie/gram dose equivalent iodine-131 to 0.35 microcurie/gram, as provided for in NRC Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.'' This change allows a proportional increase in main steam line break induced primary-to-secondary leakage when implementing the alternate steam generator tube repair criteria, which the NRC has already approved for Units 1 and 2.

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

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the TS [or operating license(s)], does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

The proposed TS change lowers the [maximum allowable] reactor coolant specific activity, which allows an increase in the leakage quantity that would be postulated to occur during a MSLB accident. This in turn allows a larger quantity of tubes with axial ODSCC to remain in service. The methodology for identifying and defining the ODSCC and for developing the leakage quantity remains unchanged. Therefore, the proposed change does not result in a significant increase in the probability of an accident.

An increase in the consequences of an accident would not occur because the proportional decrease in reactor coolant specific activity, while proportionally increasing the primary-to-secondary leakage during a postulated MSLB accident, has been evaluated to confirm the amount of activity released to the environment remains unchanged. The evaluation uses the same methodology used to establish the original primary-to-secondary leak limits in [Westinghouse Topical Report] WCAP–13990.

The control room dose, the low population zone dose, and the dose at the exclusion area boundary remains bounded by the acceptance criteria of NUREG–0800 and continue to satisfy an appropriate fraction of the 10 CFR 100 dose limits and GDC [General Design Criterion] 19. Therefore, the proposed TS change does not result in a significant increase in the consequences of an accident previously analyzed.

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

The proposed TS change does not alter the configuration of the plant. The changes do not directly affect plant operation. The change will not result in the installation of any new equipment or systems or the modification of any existing equipment or systems. No new operating procedures, conditions or modes will be created by this proposed change. SG [steam generator] tube structural integrity, as defined in draft Regulatory Guide 1.121, remains unchanged.

Therefore the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

Lowering the reactor coolant specific activity, while allowing the proportional increase in the primary-to-secondary leakage during a postulated MSLB accident, keeps the amount—of activity released to the environment unchanged. Design basis and offsite dose calculation assumptions remain satisfied. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: February 27, 1998 (TXX–98033), June 10, 1998 (TXX 98145).

Brief description of amendments: The proposed amendment would increase the RWST Low-Low level setpoint from ''greater than or equal to 40%'' to ''greater than or equal to 45%'' of span for CPSES, Units 1 and 2. The change raises the RWST Low-Low level setpoint in order to increase the volume available to complete containment spray switchover without turning off the containment spray pumps.

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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

The changes in the License Amendment Request proposes more restrictive setpoint Allowable Values for the RWST Low-Low setpoint. This more restrictive value assures that all applicable safety analysis limits are being met. Changing an RWST Low-Low setpoint from greater than or equal to 40% to greater than or equal to 45% in the Technical Specifications has no impact on the probability of occurrence of any accident previously evaluated. None of the accident analyses were affected, therefore, the consequences of all previously evaluated accidents remain unchanged.

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

The proposed changes involve the use of a more conservative value for the RWST Low-Low setpoint. As such, none of the changes effect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

There were no changes made to any of the accident analyses or safety analysis limits as a result of this proposed change. Further, the proposed change does not affect the acceptance criteria for any analyzed event. ECCS, Containment spray, and the RWST will remain capable of performing their safety function, and the new requirement will continue to provide adequate assurance of that capability. Raising the RWST Low-Low setpoint from 40% to 45% has no impact on the assumptions used in the safety analysis as discussed in Chapter 15 of the FSAR. The margin of safety established by the Limiting Conditions for Operation also remains unchanged. Thus there is no effect on the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92 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 Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, N.W., Washington, DC 20036.

NRC Project Director: John N. Hannon.

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

Date of amendment request: June 19, 1998. This amendment request supersedes the November 5, 1997, submittal in its entirety (63 FR 19981).

Description of amendment request: The proposed Operating License change and changes to the technical specifications (TS) would permit the use of a temporary alternate supply line (jumper) to provide service water (SW) to the component cooling heat exchangers. The temporary jumper will permit maintenance to be performed on the existing supply line.

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:

Virginia Electric and Power Company has reviewed the proposed changes against the criteria of 10 CFR 50.92 and has concluded that the changes do not pose a significant safety hazards consideration as defined therein. The proposed Operating License and Technical Specifications and Bases changes are necessary to allow the use of a temporary, seismic, non-missile protected jumper to provide service water (SW) to the Component Cooling Heat Exchangers (CCHXs) while maintenance work is performed on the existing SW supply line to the CCHXs. Since there is only one SW supply line to the CCHXs, an alternate SW supply must be provided whenever the line is removed from service. The temporary jumper provides this function. The jumper will only be used for a 35-day period during each of two Unit 1 refueling outages.

The use of the temporary jumper has been thoroughly evaluated, and appropriate constraints and compensatory measures (including a Contingency Action Plan) have been developed to ensure that the temporary jumper is reliable, safe, and suitable for its intended purpose. A complete and immediate loss of SW supply to the operating CCHXs is not considered credible, given the project constraints and the unlikely probability of a generated missile or heavy load drop. Existing station abnormal procedures already address a loss of component cooling, and the use of alternate cooling for a loss of decay heat removal, in the unlikely event that they are required. Furthermore, appropriate mitigative measures have been identified to address potential flooding concerns. The minor administrative changes merely correct a table format inconsistency and update Basis section references.

Consequently, the operation of Surry Power Station with the proposed amendment and license condition will not:

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

The SW and CC Systems will function as designed under the Unit operating constraints specified by this project (i.e., Unit 2 in operation and Unit 1 in a refueling outage), and the potential for a loss of component cooling is already addressed by Station Abnormal Procedures. Therefore, there is no increase in the probability of an accident previously evaluated. The possibility of flooding due to failure of the temporary SW supply jumper in the Turbine Building basement has been evaluated and dispositioned by the implementation of appropriate precautions and compensatory measures to preclude damage to the temporary jumper and to respond to a postulated flooding event. A flood watch will be present around-the-clock with authority and procedural guidance to isolate the jumper, if required. Furthermore, the CCHXs serve no design basis accident mitigating function. Therefore, the consequences of an accident previously evaluated are not increased.

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

The SW and CC Systems' design functions and basic configurations are not being altered as a result of using a temporary SW supply jumper. The temporary jumper is designed to be safety-related and seismic with all of the design attributes of the normal SW supply line, except for the automatic isolation function and complete missile and heavy load drop protection. The design functions of the SW and CC systems are unchanged as a result of the proposed changes due to (1) required plant conditions, (2) compensatory measures, (3) a Contingency Action Plan for restoration of the normal SW supply if required, and (4) strict administrative control of the temporary SW isolation valve to preclude flooding or to isolate non-essential SW within the design basis assumed time limits. Unit 1 will be in a plant condition which will provide adequate time to restore the normal SW supply, if required. Therefore, since the SW and CC systems will basically function as designed and will be operated in their basic configuration, the possibility of a new or different type of accident than previously evaluated in the UFSAR [Updated Final Safety Analysis Report] is not created.

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

The margin of safety as defined in the Technical Specifications is not reduced since an operable SW flowpath to the required number of CCHXs is provided, and Unit operating constraints, compensatory measures and contingencies will be implemented as required to ensure the integrity and the capability of the SW flowpath. The use of the temporary jumper will be limited to the time period when missile producing weather is not expected, and Unit 1 meets specified unit conditions. Therefore, the temporary SW jumper, under the imposed project constraints and compensatory measures, provides the same reliability as the normal SW supply line. Furthermore, the Probabilistic Safety Assessment for Surry Power Station has been reviewed relative to the use of the temporary SW jumper. It has been determined that due to the SW restoration project's compensatory and contingency measures, as well as the configuration restrictions that will be imposed by the Maintenance Rule online risk matrix, the impact on core damage frequency is negligible.

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: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: P. T. Kuo, Acting.

Westinghouse Electric Corporation (Licensee), Westinghouse Test Reactor, Waltz Mill Site, Westmoreland, Pennsylvania, Docket No. 50–22, License No. TR–2

Date of amendment request: December 22, 1997, supplemented on June 15, 1998.

Description of amendment request: In 1959, the Westinghouse Electric Corporation was granted a license for the Westinghouse Test Reactor (WTR) at Waltz Mill. On December 22, 1997, the licensee informed the Nuclear Regulatory Commission it had changed its name to CBS Corporation, and requested the license to be amended to reflect the name change.

On June 15, 1998, the CBS Corporation agreed that the name of the WTR licensee, as reflected on the license, can be revised to ''CBS Corporation acting through its Westinghouse Electric Company Division.'' Therefore, the purpose of this amendment is to change the name of the licensee as indicated on the WTR license from Westinghouse Electric Corporation to CBS Corporation acting through its Westinghouse Electric Company Division.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if 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 staff agrees with the licensee's no significant hazards consideration determination submitted on June 15, 1998 for the following reason.

This corporate name change does not involve any change in the management, organization, location, facilities equipment, or procedures related to or personnel responsible for the licensed activities of the WTR license. All existing commitments, obligations and representations remain in effect.

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

Attorney for license: Lisa A. Campagna, Assistant General Counsel, Law Department, CBS Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

NRC Project Director: Seymour H. Weiss.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: February 26, 1998 (TSCR 204).

Description of amendment request: The proposed changes would modify Technical Specifications (TS) and bases to reflect a lower containment leakage limit, a revised program for control of primary coolant sources outside containment, a revised control room emergency filtration design, and the addition of the primary auxiliary building exhaust filtration system.

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

1. Operation of this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

The probabilities of accidents previously evaluated are based on the probability of initiating events for these accidents. Initiating events for accidents previously evaluated for Point Beach include: Control rod withdrawal and drop, CVCS [chemical volume control system] malfunction (Boron Dilution), startup of an inactive reactor coolant loop, reduction in feedwater enthalpy, excessive load increase, losses of reactor coolant flow, loss of external electrical load, loss of normal feedwater, loss of all AC [alternating current] power to the auxiliaries, turbine overspeed, fuel handling accidents, accidental releases of waste liquid or gas, steam generator tube rupture, steam pipe rupture, control rod ejection, and primary coolant system ruptures.

This license amendment request proposes to change the limiting conditions for operation, action statements, allowable outage times, and surveillance requirements for the Point Beach Nuclear Plant [PBNP] Technical Specifications associated with the maximum permissible containment leak rate, control room emergency filtration, primary auxiliary building exhaust filtration, and primary coolant sources outside containment. These proposed changes do not cause an increase in the probabilities of any accidents previously evaluated because these changes will not cause an increase in the probability of any initiating events for accidents previously evaluated. In particular, these changes affect accident mitigation systems and equipment which do not cause accidents.

The consequences of the accidents previously evaluated in the PBNP FSAR [Final Safety Analysis Report] are determined by the results of analyses that are based on initial conditions of the plant, the type of accident, transient response of the plant, and the operation and failure of equipment and systems. The changes proposed in this license amendment request provide appropriate limiting conditions for operation, action statements, allowable outage times, and surveillance requirements for maximum permissible containment leak rate, control room emergency filtration, primary auxiliary building exhaust filtration, and primary coolant sources outside containment.

The proposed changes affect components that are required to ensure the proper operation of accident mitigation systems and equipment. The proposed changes do not increase the probability of failure of this equipment or its ability to operate as required for the accidents previously evaluated in the PBNP FSAR.

Therefore, this proposed license amendment does not affect the consequences of any accident previously evaluated in the Point Beach Nuclear Plant FSAR, because the factors that are used to determine the consequences of accidents are not being changed.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of

a new or different kind of accident from any accident previously evaluated.

New or different kinds of accidents can only be created by new or different accident initiators or sequences. New and different types of accidents (different from those that were originally analyzed for Point Beach) have been evaluated and incorporated into the licensing basis for Point Beach Nuclear Plant. Examples of different accidents that have been incorporated into the Point Beach Licensing basis include anticipated transients without scram and station blackout. The changes proposed by this license amendment request do not create any new or different accident initiators or sequences because these changes to limiting conditions for operation, action statements, allowable outage times, and surveillance requirements for maximum permissible containment leak rate, control room emergency filtration, primary auxiliary building exhaust filtration, and primary coolant sources outside containment will not cause failures of equipment or accident sequences different than the accidents previously evaluated. Therefore, these proposed Technical Specifications changes do not create the possibility of an accident of a different type than any previously evaluated in the Point Beach FSAR.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection.

The changes proposed by this license amendment request provide the appropriate limiting conditions for operation, action statements, allowable outage times, and surveillance requirements for maximum permissible containment leak rate, control room emergency filtration, primary auxiliary building exhaust filtration, and primary coolant sources outside containment. This ensures that the safety systems that protect the reactor and containment will operate as required. The design and operation of the reactor and containment are not affected by these proposed changes. Therefore, the margins of safety for Point Beach are not being reduced because the design and operation of the reactor and containment are not being changed and the safety systems and limiting conditions of operation for these safety systems that provide their protection that are being changed will continue to meet the requirements for accident mitigation for Point Beach Nuclear Plant.

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: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cynthia A. Carpenter.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: May 28, 1998 (TSCR 203).

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) to provide a specific numerical setting for reactor trip, reactor coolant pump trip, and auxiliary feedwater initiation on a loss of power to the 4 kilovolt (kV) buses. Changes to the bases for the affected TS sections are also being made.

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

1. Operation of the Point Beach Nuclear Plant [PBNP] in accordance with the proposed amendments will not create a significant increase in the probability or consequences of an accident previously evaluated.

The probabilities of accidents previously evaluated are based on the probability of initiating events for these accidents. Initiating events for accidents potentially affected by the proposed amendments previously evaluated for Point Beach include losses of reactor coolant flow, loss of external electrical load, loss of normal feedwater, and loss of all AC [alternating current] power to the auxiliaries.

This license amendment request proposes to clarify the setting limit for the undervoltage reactor trip, auxiliary feedwater initiation and reactor coolant pump trip by providing an actual numerical value in place of the word ''Normal'' thereby eliminating any confusion as to the actual value used in the setting limit for this protection function.

This proposed change does not cause an increase in the probabilities of any accidents previously evaluated because the change will not cause an increase in the probability of any initiating events for accidents previously evaluated. In particular, the proposed change more clearly defines the actual setting limit for the 4 KV undervoltage protection function taking into account the effects of voltage decay and response times. This is a protection function for mitigation of these events. Appropriate delay times are implemented in this function to ensure momentary voltage transients do not initiate these events while ensuring appropriate protection for these loss of power events. Therefore, there is no significant increase in the probability or consequences of any event previously analyzed.

The consequences of the accidents previously evaluated in the PBNP FSAR [Final Safety Analysis Report] are determined by the results of analyses that are based on initial conditions of the plant, the type of accident, transient response of the plant, and the operation and failure of equipment and systems. The changes proposed in this license amendment request provide appropriate limiting conditions for the setting limits for the Point Beach Nuclear Plant Technical Specifications for the 4 KV undervoltage protection function. Thus the analyses of the events remain valid and demonstrate that there are no radiological consequences from these events.

Therefore, this proposed license amendment does not affect the consequences of any accident previously evaluated in the Point Beach Nuclear Plant FSAR, because the factors that are used to determine the consequences of accidents are not being changed.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

New or different kinds of accidents can only be created by new or different accident initiators or sequences. New and different types of accidents (different from those that were originally analyzed for Point Beach) have been evaluated and incorporated into the licensing basis for Point Beach Nuclear Plant. Examples of different accidents that have been incorporated into the Point Beach Licensing basis include anticipated transients without scram and station blackout.

The change proposed by the amendments to provide specific undervoltage setting limits does not create any new or different accident initiators or sequences because the change to the 4 KV undervoltage protection function will not cause failures of equipment or accident sequences different than the accidents previously evaluated. Therefore, the proposed Technical Specification change does not create the possibility of an accident of a different type than any previously evaluated in the Point Beach FSAR.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments [will] not create a significant reduction in a margin of safety.

The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection.

The change proposed by this license amendment request provides the appropriate setting limit for the 4 KV undervoltage protection function. This ensures that the safety systems that protect the reactor and containment will operate as required. The design and operation of the reactor and containment are not affected by these proposed changes. Therefore, the margins of safety for Point Beach are not being reduced because the design and operation of the reactor and containment are not being changed.

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: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cynthia A. Carpenter.

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.

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

Date of amendment request: June 17, 1998, as supplemented June 23, 1998.

Brief description of amendment request: The proposed amendment would revise Section 3.1.1c of the Technical Specifications (TS), Appendix A of the Operating License for the Palisades Nuclear Plant, to change the minimum required primary coolant system flow. The currently specified value is 140.7×10^{6} lb/hr [pounds per hour] or greater, when corrected to 532 °F. The licensee proposed to revise the TS to specify a value of greater than or equal to 352,000 gpm [gallons per minute], which is equivalent to approximately 135×10^{6} lb/hr, when corrected to 532 °F.

Date of publication of individual notice in **Federal Register**: July 2, 1998 (63 FR 36271)

Expiration date of individual notice: August 3, 1998.

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

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

Date of amendment request: June 26, 1998 (NRC–98–0040).

Brief description of amendment request: The proposed amendment would provide a one-time extension of the interval for a number of technical specification (TS) surveillance requirements that will be performed in the sixth refueling outage. TS 4.0.2 and Index page xxii would be revised and TS tables 4.0.2–1 and 4.0.2–2 would be replaced to reflect the extensions.

Date of publication of individual notice in **Federal Register**: July 2, 1998 (63 FR 36273).

Expiration date of individual notice: August 3, 1998.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

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

Date of amendment request: June 19, 1998.

Brief description of amendment request: This amendment revises the Beaver Valley Power Station, Units 1 and 2 (BVPS–1 and BVPS–2), Technical Specifications (TS) definition of a channel calibration to add two sentences stating that (1) the calibration of instrument channels with resistance temperature detector or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel and (2) whenever a sensing element is replaced, the next required channel calibration shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. This proposed change would make the BVPS–1 and BVPS–2 TS definition of channel calibration consistent with the definition of a channel calibration contained in the NRC's improved Standard Technical Specifications for Westinghouse Plants (NUREG–1431, Revision 1).

Date of publication of individual notice in **Federal Register**: June 26, 1998.

Expiration date of individual notice: July 27, 1998 (63 FR 34939).

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

Pennsylvania Power and Light Company, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: June 17, 1998.

Brief description of amendment request: This amendment revises the applicability requirement in TS Sections 3.4.2, ''Safety/Relief Valves'' (Action c); 4.4.2; 3.3.7.5, ''Accident Monitoring Instrumentation'' (TS Table 3.3.7.5–1, Action 80 and 4.3.7.5, ''Surveillance Requirements,'' Table 4.3.7.5–1 ''Accident Monitoring Instrumentation Surveillance Requirements''). The change to the referenced TSs adds the following applicability footnote:

Compliance with these requirements for the ''J'' SRV acoustic monitor is not required for the period beginning June 15, 1998, until the next unit shutdown of sufficient duration to allow for containment entry, not to exceed the 9th refueling and inspection outage.

Date of publication of individual notice in **Federal Register**: June 23, 1998 (63 FR 34200).

Expiration date of individual notice: July 23, 1998.

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: May 27, 1997, as supplemented on August 1, 1997, and March 24, 1998.

Brief description of amendments: The amendments revise Technical Specification Section 6, ''Administrative Controls,'' to incorporate revised organizational titles and delete Unit 1 Facility Operating License Condition 2.C.(30)(a). In addition, the amendments change the submittal frequency of the Radiological Effluent Release Report from semiannually to annually and make several administrative and editorial changes.

Date of issuance: June 26, 1998.

Effective date: Immediately, to be implemented within 90 days.

Amendment Nos.: 128 and 113.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Unit 1 Facility Operating License and the Technical Specifications.

Date of initial notice in **Federal Register**: July 30 ,1997. The August 1, 1997, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The March 24, 1998, submittal changed the scope of the initial **Federal Register** notice. The proposed amendments were renoticed on May 20, 1998 (63 FR 27759).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 26, 1998.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket No. 50–373, LaSalle County Station, Unit 1, LaSalle County, Illinois

Date of application for amendment: November 24, 1997, as supplemented April 16, 1998.

Brief description of amendment: The amendment revises Technical Specification 3/4 3.2, ''Isolation Actuation Instrumentation'' to add/revise various isolation setpoints for leak detection instrumentation. These changes are necessary due to modifications to the Reactor Water Cleanup (RWCU) system to restore ''hot'' suction to the RWCU pumps and due to a re-evaluation of the high energy line break analysis. In addition, the amendment eliminates isolation actuation trip functions for the Residual Heat Removal system steam condensing mode and shutdown cooling mode.

Date of issuance: July 6, 1998.

Effective date: Immediately, to be implemented prior to restart from L1F35

Amendment No.: 129.

Facility Operating License No. NPF–11: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 14, 1998 (63 FR 2278). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application of amendments: May 30, 1997, as supplemented May 7, 1998 and June 18, 1998.

Brief description of amendments: The amendments revise the Facility Operating License and Technical Specifications to reflect the permanently shut down and defueled status of the reactor.

Date of issuance: June 30, 1998.

Effective date: As of the date of issuance (June 30, 1998) and shall be implemented within 90 days.

Amendment No.: 193.

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

Date of initial notice in **Federal Register**: July 16, 1997 (62 FR 38132 and 62 FR 38133). The May 7, 1998, supplement relocated the provisions of Technical Specification 3/4.9.15. The June 18, 1998, supplement consisted of supporting technical information. The supplements did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the original notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 30, 1998.

No significant hazards consideration received: No.

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

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1, Monroe County, Michigan

Date of amendment request: January 27, 1998 (Reference NRC–98–0023).

Brief description of amendment: This amendment revises the Fermi 1 License to allow Detroit Edison to receive, acquire, possess, use, and transfer byproduct material without restriction to chemical form for sample analysis, instrument calibration, or associated with radioactive apparatus, hardware, tools, and equipment, provided the cumulative radioactive material quantity of the byproduct material does not exceed the criteria contained in Section 30.72, Schedule C, Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release.

Date of issuance: June 22, 1998.

Effective date: Within 60 calendar days from the date of issuance of this amendment.

Amendment No.: 12.

Facility Operating License No. DPR–9: Amendment revised License by adding a subpart 3 to Part 2.B.

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17223).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 22, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

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

Date of application of amendments: June 4, 1998.

Brief description of amendments: The amendments revise Technical Specification 4.17.2 to allow continued operation with certain steam generator tubes that exceed their repair limit as a result of tube end anomalies. This action temporarily exempts these tubes from the requirement for sleeving, rerolling, or removal from service until they are repaired during or before the next scheduled refueling outages for the respective unit. This action supersedes the Notice of Enforcement Discretion that was issued by the staff on June 4, 1998.

Date of Issuance: July 1, 1998.

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

Amendment Nos.: Unit 1—230; Unit 2—227.

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

Public comments requested as to proposed no significant hazards consideration: Yes. (63 FR 33097 dated June 17, 1998). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by July 16, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated July 1, 1998.

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

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

Date of amendment request: June 3, 1997, as supplemented by letter dated May 1, 1998.

Brief description of amendment: The amendment changed the Appendix A Technical Specifications (TSs) by changing the action requirements for TS 3/4.3.2 for the Safety Injection System Sump Recirculation Actuation Signal (RAS). It revised the allowed outage time for a channel of RAS to be in the tripped condition from ''prior to entry into the applicable MODE(S) following the next COLD SHUTDOWN'' to the more restrictive time limit of 48 hours, and added a shutdown requirement. Additionally, the TS 3.0.4 exemption was removed from the action for the tripped condition. A change to TS Bases Section 3/4.3.2 was also included.

Date of issuance: July 2, 1998.

Effective date: July 2, 1998, to be implemented within 60 days.

Amendment No.: 143.

Facility Operating License No. NPF–38: Amendment revised the Technical Specifications. *Date of initial notice in* **Federal Register**: June 18, 1997 (62 FR 33124).

The additional information contained in the supplemental letter dated May 1, 1998, was clarifying in nature and thus, it was within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 2, 1998, as supplemented by letter dated April 21, 1998.

Description of amendment request: The amendment revised Technical Specification 4.5.2.b.1 for the emergency core cooling system subsystems to delete the requirement to vent the operating chemical volume and control system centrifugal pump casing.

Date of issuance: June 24, 1998.

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

Amendment No.: 58.

Facility Operating License No. NPF–86. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17225).

The supplemental letter provided clarifying information that did not change the staff's proposed no significant hazards determination.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of application for amendment: April 13, 1998.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by adding a new TS 3.5.5, ''Emergency Core Cooling Systems—Trisodium Phosphate (TSP).'' The TSP surveillance requirements in TSs 4.5.2.c.3 and 4.5.2.c.4 are relocated to new TS 3.5.5 as TS 4.5.5.1 and TS 4.5.5.2, respectively. Also, the amount of TSP is increased, the surveillance requirements are modified, a new limiting condition of operation is included, and the applicable TS Index pages and Bases sections are updated to reflect the changes.

Date of issuance: June 22, 1998.

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

Amendment No.: 217.

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

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25114).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 22, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 13, 1997, as supplemented on December 29, 1997, and April 8, 1998.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by modifying TS 3.1.2.1, ''Flow Paths—Shutdown;'' TS 3.1.2.2, ''Flow Paths—Operating;'' TS 3.1.2.3, ''Charging Pump—Shutdown;'' TS 3.1.2.4, ''Charging Pumps—Operating;'' TS 3.1.2.5, ''Boric Acid Pumps—Shutdown;'' TS 3.1.2.6, ''Boric Acid Pumps—Operating;'' TS 3.1.2.8, ''Borated Water Sources—Operating;'' TS 3.4.1.3, ''Coolant Loops and Coolant Circulation—Shutdown;'' TS 3.4.3, ''Relief Valves;'' TS 3.4.9.1, ''Reactor Coolant System;'' TS 3.4.9.2, ''Pressurizer;'' TS 3.4.9.3, ''Overpressure Protection Systems;'' TS 3.5.3, ''ECCS Subsystems—T_{avg} < 300 °F;'' and TS 3.10.3, ''Pressure/Temperature Limitation—Reactor Criticality,'' and their associated Bases in the areas that are affected by the modified Low Temperature Overpressure Protection system, the updated reactor coolant system pressure and temperature curves and heatup and cooldown limits. Additionally, minor changes are made to correct various items, such as, updating of redundant or outdated TSs.

Date of issuance: July 1, 1998.

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

Amendment No.: 218.

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

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4315).

The December 29, 1997, and April 8, 1998, letters provided clarifying information that did not change the scope of the November 13, 1997, application and the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 27, 1997, as supplemented by letter dated December 4, 1997.

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, (DCPP) Unit Nos. 1 and 2 to change Technical Specification (TS) 3/4.8.1.1, ''A.C. Sources—Operating,'' to clarify that emergency diesel generator (EDG) testing is initiated from standby conditions rather than ''ambient'' conditions. The associated TS Bases were revised to discuss the temperature range that satisfies EDG standby conditions. TS 3/4.3.2, ''Instrumentation—Engineering Safety Features Actuation System Instrumentation'' was also changed. This revision clarified that when one or both of the first level load shed relays, or one or both of the second level undervoltage relays are inoperable, the associated EDG for that bus shall be declared inoperable.

Date of issuance: June 5, 1998.

Effective date: June 5, 1998, to be implemented within 90 days from the date of issuance.

Amendment Nos.: Unit 1—127; Unit 2—125

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17240).

The December 4, 1997, supplemental letter provided additional clarifying information and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 5, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 10, 1998, as supplemented by letter dated May 1, 1998.

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Units 1 and 2 to revise TS 6.2.2.g and TS 6.3 to change the name of the Operations Manager to Operations Director, to add the position of Operations Middle Manager, and to change the requirement for the Operations Director to hold a senior reactor operator (SRO) license.

Date of issuance: June 11, 1998.

Effective date: June 11, 1998.

Amendment Nos.: Unit 1–128; Unit 2–126.

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

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25116).

The May 1, 1998, supplemental letter provided additional clarifying information and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 11, 1998.

No significant hazards consideration comments received: Yes.

The Commission received one letter with comments which did not change its finding and conclusion as discussed in the safety evaluation.

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

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: May 12, 1998

Brief description of amendments: These amendments relocate certain requirements related to fire protection from the TSs to the Updated Final Safety Analysis Report. The TS sections to be relocated are: 3/4.3.7.9, Fire Detection Instrumentation; 3/4.7.6, Fire Suppression Systems; 3/4.7.7, Fire

Rated Assemblies; and 6.2.2e, Fire Brigade Staffing. The amendments also replace License Condition 2.C.(6) for Unit 1 and License Condition 2.C.(3) for Unit 2. These amendments are consistent with the guidance of NRC Generic Letter (GL) 86–10, ''Implementation of Fire Protection Requirements,'' and GL 88–12, ''Removal of Fire Protection Requirements from Technical Specifications.''

Date of issuance: June 24, 1998.

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

Amendment Nos.: 177 and 150.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications and Licenses.

Date of initial notice in **Federal Register**: May 21, 1998 (63 FR 28010).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 26, 1997, as supplemented April 17, 1998.

Brief description of amendment: The amendment relocates snubber operability, surveillance, and records requirements from the Technical Specifications to plant controlled documents. A condition is added to the license to require that the relocated requirements be described in the Final Safety Analysis Report such that 10 CFR 50.59 will apply to future changes to those requirements.

Date of issuance: June 30, 1998.

Effective date: As of the date of issuance to be implemented within 90 days.

Amendment No.: 243.

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

Date of initial notice in **Federal Register**: January 29, 1997 (62 FR 4352).

The April 17, 1998, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 6, 1995, as supplemented by letters dated January 9, 1998, and February 3, 1998, for the safety injection tanks (SITs), and November 8, 1995, as supplemented by letters dated January 9, 1998, and February 3, 1998, for the low pressure safety injection (LPSI).

Brief description of amendments: The amendments modify the technical specifications (TSs) to extend the allowed outage times (AOTs) for a single inoperable SIT from one hour to 24 hours, and for a single inoperable SIT specifically due to malfunctioning SIT water level or nitrogen cover pressure instrumentation inoperability from one hour to 72 hours. In addition, the amendments extend the AOT for a single inoperable LPSI train from 72 hours to 7 days. The amendments also add a Configuration Risk Management Program to the TSs that puts a proceduralized probabilistic risk assessment-informed process in place that ensures the licensee assesses the overall impact of plant maintenance on plant risk.

Date of issuance: June 19, 1998.

Effective date: June 19, 1998, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2—139; Unit 3—131.

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

Date of initial notice in **Federal Register**: April 10, 1996 (61 FR 15995) and February 11, 1998 (63 FR 6991).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 19, 1998.

No significant hazards consideration comments received: No.

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

Southern Nuclear Power Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: September 4, 1997, as supplemented by letters dated November 20, 1997, May 19 and June 12, 1998.

Brief description of amendments: The changes to the common Technical Specifications allow an increase in the Unit 1 spent fuel storage capacity from 288 to 1476 fuel assemblies.

Date of issuance: June 29, 1998.

Effective date: As of the date of issuance to be implemented on a schedule consistent with the receipt and storage of new fuel in the fall of 1998 for the spring 1999 refueling outage of Unit 1.

Amendment Nos.: Unit 1—102; Unit 2—80.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications, Operating Licenses, and Appendix D.

Date of initial notice in **Federal Register**: December 31, 1997 (62 FR 68317); and renoticed on May 11, 1998 (63 FR 25883).

The supplements dated May 19 and June 12, 1998, provided clarifying information that did not change the scope of the September 4, 1997, application and the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia.

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

Date of application for amendments: June 7, 1996, as supplemented on September 26, 1997, January 21, 1998, May 28, 1998, and June 29, 1998 (TS 95–19).

Brief description of amendments: The amendments change the Technical Specifications (TS) by relocating portions of Section 6, ''Administrative Controls,'' to the Sequoyah Nuclear Quality Assurance Plan. This Change is consistent with NUREG–1431, ''Standard Technical Specifications—Westinghouse Plants.''

Date of issuance: July 1, 1998.

Effective date: July 1, 1998.

Amendment Nos.: 233 and 223.

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

Date of initial notice in **Federal Register**: July 17, 1996 (61 FR 37302).

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

No significant hazards consideration comments received: None.

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

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

Date of application for amendment: December 23, 1997, as supplemented by letter dated June 11, 1998.

Brief description of amendment: This amendment revises Technical Specification (TS) Section 1.0, ''Definitions,'' to clarify the meaning of core alteration; relocates TS Section 3/4.9.5, ''Refueling Operations—Communications,'' and the associated bases to the Technical Requirements Manual; and adds TS Section 3.0.6 and associated bases to address the return to service of inoperable equipment.

Date of issuance: June 30, 1998.

Effective date: June 30, 1998.

Amendment No.: 224.

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

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

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

.

No significant hazards consideration comments received: No.

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

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: March 25, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Sections 6.1.1; 6.2.1.b; 6.5.1.1; 6.5.1.6.a,d,h, and m; 6.5.1.7.c; 6.5.1.8; 6.14.1.2; 6.15.b; 6.2.3.5; 6.5.1.2; 6.5.1.7.a for Unit 1 and 6.1.1; 6.2.1.b; 6.5.1.1; 6.5.1.6.a,d,h, and m; 6.5.1.7.c; 6.5.1.8; 6.13.b; 6.14.b; 6.2.3.5; 6.5.1.2; and 6.5.1.7.a for Unit 2, changing the title of Station Manager to Site Vice President, and the titles of the Assistant Station Managers to Manager-Station Operation and Maintenance and Manager-Station Safety and Licensing.

Date of issuance: June 23, 1998.

Effective date: June 23, 1998.

Amendment Nos.: 212 and 193.

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

Date of initial notice in **Federal Register**: April 22, 1998 (63 FR 19980).

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

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.

Dated at Rockville, Maryland, this 8th day of July 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–18684 Filed 7–14–98; 8:45 am]

BILLING CODE 7590–01–P

PENSION BENEFIT GUARANTY CORPORATION

Interest Assumption for Determining Variable-Rate Premium; Interest on Late Premium Payments; Interest on Underpayments and Overpayments of Single-Employer Plan Termination Liability and Multiemployer Withdrawal Liability; Interest Assumptions for Multiemployer Plan Valuations Following Mass Withdrawal

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of interest rates and assumptions.

SUMMARY: This notice informs the public of the interest rates and assumptions to be used under certain Pension Benefit Guaranty Corporation regulations. These rates and assumptions are published elsewhere (or are derivable from rates published elsewhere), but are collected and published in this notice for the convenience of the public. Interest rates are also published on the PBGC's web site (http://www.pbgc.gov).

DATES: The interest rate for determining the variable-rate premium under part 4006 applies to premium payment years beginning in July 1998. The interest assumptions for performing multiemployer plan valuations following mass withdrawal under part 4281 apply to valuation dates occurring in August 1998. The interest rates for late premium payments under part 4007 and for underpayments and overpayments of single-employer plan termination liability under part 4062 and multiemployer withdrawal liability under part 4219 apply to interest accruing during the third quarter (July through September) of 1998.

FOR FURTHER INFORMATION CONTACT: Harold J. Ashner, Assistant General Counsel, Office of the General Counsel, Pension Benefit Guaranty Corporation, 1200 K Street, NW., Washington, DC 20005, 202–326–4024. (For TTY/TDD users, call the Federal relay service toll-free at 1–800–877–8339 and ask to be connected to 202–326–4024.)

SUPPLEMENTARY INFORMATION:

Variable-Rate Premiums

Section 4006(a)(3)(E)(iii)(II) of the Employee Retirement Income Security Act of 1974 (ERISA) and § 4006.4(b)(1) of the PBGC's regulation on Premium Rates (29 CFR part 4006) prescribe use of an assumed interest rate in determining a single-employer plan's variable-rate premium. The rate is the ''applicable percentage'' (described in the statute and the regulation) of the annual yield on 30-year Treasury securities for the month preceding the beginning of the plan year for which premiums are being paid (the ''premium payment year''). The yield figure is reported in Federal Reserve Statistical Releases G.13 and H.15.

For plan years beginning before July 1, 1997, the applicable percentage of the 30-year Treasury yield was 80 percent. The Retirement Protection Act of 1994 (RPA) amended ERISA section 4006(a)(3)(E)(iii)(II) to change the applicable percentage to 85 percent, effective for plan years beginning on or after July 1, 1997. (The amendment also provides for a further increase in the applicable percentage '' to 100 percent '' when the Internal Revenue Service adopts new mortality tables for determining current liability.)

The assumed interest rate to be used in determining variable-rate premiums for premium payment years beginning in July 1998 is 4.85 percent (*i.e.*, 85 percent of the 5.70 percent yield figure for June 1998).

(Under section 774(c) of the RPA, the amendment to the applicable percentage was deferred for certain regulated public utility (RPU) plans for as long as six months. The applicable percentage for RPU plans has therefore remained 80 percent for plan years beginning before January 1, 1998. For ''partial'' RPU plans, the assumed interest rates to be used in determining variable-rate premiums can be computed by applying the rules in § 4006.5(g) of the premium rates regulation. The PBGC's 1997 premium payment instruction booklet also describes these rules and provides a worksheet for computing the assumed rate.)

The following table lists the assumed interest rates to be used in determining variable-rate premiums for premium