The Subcommittee will discuss proposed ACRS activities and related matters, including: ACRS priorities for CY 1999; emerging technical issues; and ACRS report to the Commission on the NRC Safety Research Program. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff person named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. A detail agenda for this meeting is available for downloading or viewing on the internet at *http://www.nrc.gov/ACRSACNW.*

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting the cognizant ACRS staff person, Dr. John T. Larkins (telephone: 301/415–7360) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any changes to the schedule, etc., that may have occurred.

Dated: December 23, 1998.

Michael T. Markley,

Acting Chief, Nuclear Reactors Branch.

[FR Doc. 98–34571 Filed 12–29–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATES: Weeks of December 28, 1998, January 4, 11, and 18, 1999.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of December 28

There are no meetings scheduled for the week of December 28, 1998.

Week of January 4—Tentative

Wednesday, January 6

11:30 a.m.
Affirmation Session (Public Meeting) (if needed).

Week of January 11—Tentative

Monday, January 11

2:00 p.m.
Briefing on Risk-Informed Initiatives (Public Meeting).

Tuesday, January 12

9:00 a.m.
Briefing on Decommissioning Criteria for West Valley (Public Meeting).

Wednesday, January 13

10:00 a.m.
Briefing on Reactor Licensing Initiatives (Public Meeting).
11:30 a.m.
Affirmation Session (Public Meeting) (If Needed).

Friday, January 15

9:00 a.m.
Briefing on Investigative Matters (Closed—Ex. 5 & 7).
10:00 a.m.
Briefing by Executive Branch (Closed—Ex. 1).

Week of January 18—Tentative

Tuesday, January 19

2:00 p.m.
Briefing on Status of Third Party Oversight of Millstone Station's Employee Concerns Program and Safety Conscious Work Environment (Public Meeting).

Wednesday, January 20

9:30 a.m.
Briefing on Reactor Inspection, Enforcement And Assessment (Public Meeting).
11:00 a.m.
Affirmation Session (Public Meeting) (If Needed).

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings, call (Recording)—(301) 415–1292.

CONTACT PERSON FOR MORE INFORMATION: Bill Hill (301) 415–1661.

* * * * *

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

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–145–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: December 24, 1998.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 98–34679 Filed 12–28–98; 12:01 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 7, 1998, through December 17, 1998. The last biweekly notice was published on December 16, 1998 (63 FR 69332).

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 January 29, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: October 27, 1998.

Description of amendment request: The Carolina Power & Light Company, licensee for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, proposed amendments to the Operating Licenses for the BSEP units. The amendments are administrative in nature and would delete various completed license conditions, make editorial changes, and provide clarifying information.

The licensee has concluded that the proposed license amendments do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three standards set forth in 10 CFR 50.92 is provided below.

Basis for a 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 license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes revise the BSEP, Unit Nos. 1 and 2, Facility Operating Licenses to delete various license conditions that have been completed, make editorial changes, and provide clarifying information. The changes are administrative and only provide updated and clarifying information. No physical or operational changes to the facility will result from the proposed changes. Therefore, the proposed license amendments do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes revise the BSEP, Unit Nos. 1 and 2, Facility Operating Licenses to delete various license conditions that have been completed, make editorial changes, and provide clarifying information. The changes are administrative and only provide updated and clarifying information. The proposed license amendments do not alter any plant operation and will not result in a physical change to the facility. Therefore, the proposed license amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes revise the BSEP, Unit Nos. 1 and 2, Facility Operating Licenses to delete various license conditions that have been completed, make editorial changes, and provide clarifying information. The changes are administrative and only provide updated and clarifying information. No physical or operational changes to the facility will result from the proposed changes. Therefore, the proposed license amendments do not involve a reduction in a margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: November 30, 1998.

Description of amendment request: This amendment request proposes to relocate, to a licensee controlled document, the requirement for removal of the Reactor Protection System (RPS) shorting links. Removal of the shorting links enables a non-coincident scram on high neutron flux as detected by the Source Range Monitors (SRMs).

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

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

The RPS shorting links are not precursors to any previously evaluated accident. The Source Range Monitors (SRMs), and the ability of the SRMs to provide a RPS trip, are also not precursors to any previously evaluated accident. Therefore, relocating the RPS shorting link requirement to administrative controls [the Updated Final Safety Analysis Report (UFSAR)] will not increase the probability of an accident previously evaluated.

The RPS shorting links are not assumed to be removed in any accident analysis, and the SRMs are not assumed to provide a RPS trip in any accident analysis. The refueling interlocks and SHUTDOWN MARGIN calculations will continue to provide assurance of reactivity control. Therefore, relocating the RPS shorting link requirements to administrative controls [the UFSAR] will not increase the consequences of an accident previously evaluated.

The RPS shorting link requirements will be relocated to administrative controls that are administered pursuant to the requirements of 10 CFR 50.59, thereby reducing the level of regulatory control. The level of regulatory control has no impact on the probability or consequences of an accident previously evaluated.

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

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

Relocating the RPS shorting link requirements to administrative controls [the UFSAR] does not create any new failure mechanisms. No new equipment will be installed or utilized, and no new operating conditions will be initiated as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The refuel interlocks and SHUTDOWN MARGIN calculations will continue to ensure that the reactor stays subcritical in the Refuel Mode. The margin to safety as represented by the SHUTDOWN MARGIN designed into the core and verified in the SHUTDOWN MARGIN calculations will be unaffected by relocation of the RPS shorting link requirements to administrative controls [the UFSAR]. The margin to safety as represented by the fuel bundle drop assumptions protected by the refuel interlocks will be unaffected. In addition, no accident analysis assumes that the RPS shorting links are removed. In addition, the RPS shorting link requirements will be relocated to administrative controls [the UFSAR] for which future change will be evaluated pursuant to the requirements of 10 CFR 50.59. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite, and, thus, these changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: October 30, 1998 (LAR–236).

Description of amendment request: The proposed amendment would change the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS) Section 5.6.2.19, Section 3.4.11, Bases 3.4.11 and Bases 3.4.3. The changes reflect the use of fluence methodology described in Topical Report BAW–2241P, ''Fluence and Uncertainty Methodologies,'' and the use of American Society of Mechanical Engineers (ASME) Code Case N–514, ''Low Temperature Overpressure Protection,'' for developing Low Temperature Overpressure Protection (LTOP) limits. Reference to Topical Report BAW–1543A, ''Integrated Reactor Vessel Surveillance Program,'' was also added to ITS Section 5.6.2.19. ITS Section 3.4.11 (Low Temperature Overpressure Protection System), was revised to reflect the new LTOP limits based on revised fluence projections through 32 Effective Full Power Years (EFPY). The Pressure/Temperature (P/T) Limits Report is being revised to reflect the new P/T limits for heatup, cooldown, hydrostatic and leak test, and to incorporate the CR–3 LTOP curve.

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.

LAR [License Amendment Request] #236 proposes several changes to the ITS operational limits. These changes are being proposed to maintain the necessary margins of safety through 32 EFPY using analyses based on methodologies that have been previously approved for use at CR–3, ASME Code Case N–514 and LTOP SER [Safety Evaluation Report], and are currently being reviewed by the NRC staff:

—NRC to FPC letter, 3N1293–30, dated December 20, 1993, ''Crystal River Unit 3—Issuance of Amendment RE: Improved Technical Specifications (TAC No. M74563)''
—NRC to FPC letter, 3N1297–16, dated December 22, 1997, ''Crystal River Unit 3—Staff Evaluation and Issuance of Amendment RE: Low-Temperature Overpressure Protection (TAC No. M99277)''
—NRC to FPC letter, 3N079705, dated July 3, 1997, ''Crystal River 3—Exemption from Requirements of 10 CFR 50.60, Acceptance Criteria for Fracture Prevention for Lightwater Nuclear Power Reactors for Normal Operation (TAC No. M98380)''
—BAW–2241P, ''Fluence and Uncertainty Methodologies''

The limiting transient for LTOP remains a failed-open makeup valve. Existing LTOP controls (maximum of one makeup pump capable of injecting into the RCS [reactor coolant system], high pressure injection (HPI) deactivated, the CFTs [core flood tanks] isolated, pressure relief capability and maintaining a gas volume in the RCS) remain unchanged from the current ITS 3.4.11 as approved by Reference 3, except the setpoints proposed herein. The setpoints are being updated to reflect the new 32 EFPY fluence analysis and P/T limits. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated since they do not introduce new systems, failure modes or plant perturbations. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes will not involve a significant reduction in the margin of safety since the proposed P/T limitations have been developed consistent with the requirements of 10 CFR 50.60. The operational limits have been developed to maintain the necessary margins of safety as defined by ASME through 32 EFPY using methodologies previously reviewed and approved by the NRC. The objective of these limits is to prevent non-ductile failure during any normal operating condition, including anticipated operational occurrences and system hydrostatic tests.

The LTOP safety factors are based on reanalyzed conditions for 32 EFPY of operation utilizing methodology contained in ASME Code Case N–514 which has been approved for use at CR–3. The Code Case provides an acceptable margin of safety against flaw initiation and reactor vessel failure. The application of Code Case N–514 for CR–3 ensures an acceptable level of safety. Therefore, this 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: 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 14024, St. Petersburg, Florida 33733–4042.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: October 30, 1998.

Description of amendment request: The proposed amendment requests approval of a change to the Crystal River Unit 3 (CR–3) Final Safety Analysis Report (FSAR) regarding the methodology for performing the Spent Fuel Pool (SFP) B criticality analysis. Recent Boraflex samples from the SFP B demonstrate a weight loss in excess of the available margin within the current licensing basis calculation. The criticality analysis calculations proposed in this amendment request demonstrate that the burnup/enrichment curves in the current Improved Technical Specifications (ITS) have sufficient margin to accommodate up to a 20% loss in Boraflex neutron absorption, and still maintain SFP B at less than or equal to 0.95 k-effective when fully loaded and flooded with unborated water. Florida Power Corporation has concluded that the change in the criticality analysis methodology represents an unreviewed safety question, and thus requires prior NRC approval.

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.

No. The two possible accidents are: (1) criticality during normal storage and (2) criticality due to a misloaded fuel assembly during handling fuel. Each are discussed below:

(1) Criticality during normal storage.

For criticality during normal storage to occur, there must be a loss of negative reactivity since an addition of positive reactivity is not possible without fuel movement. A loss in negative reactivity could result only from reduction in Boraflex inventory below that needed to meet the design basis. The proposed criticality analysis for Spent Fuel Pool B demonstrates that Spent Fuel Pool B is capable of maintaining the design basis requirement of k-effective less than or equal to 0.95 when flooded with unborated water and with a loss of up to 20% of the Boraflex absorber material. Therefore, allowing up to 20% Boraflex loss with the new analysis does not significantly increase the probability of an accident previously evaluated.

(2) Criticality during fuel handling.

Criticality during fuel handling could occur due to loss of negative reactivity, or the addition of positive reactivity. Loss of negative reactivity could result from loss of Boraflex as discussed above.

Addition of positive reactivity would result from the misloading of fuel in a fashion not in accordance with ITS LCO 3.7.15, such as the misloading of a fresh 5.05% enriched fuel assembly into Region 2 or side-by-side with another fresh fuel assembly in Region 1. The minimum required boron concentration of ITS LCO 3.7.14 and CR–3 FSAR 9.3.2.1.2 are intended to compensate for just such an accident. Consistent with the double-contingency principle, a boron dilution is not required to be considered concurrent with a misloaded new fuel assembly (bases of ITS LCO 3.7.14). The use of a new calculational method will not increase the probability of fuel assembly misloading. A boron dilution event without an accompanying misloaded fuel assembly is not impacted by the new criticality analysis, since the design basis allows for unborated water for normal storage conditions.

Therefore, since the proposed criticality analysis does not increase the probability of a misloaded fuel assembly, the probability of an occurrence of an accident previously evaluated is not significantly increased.

Boraflex is credited with preventing inadvertent criticality. It is not credited with mitigating the effects, or dose consequences, to the public or to plant personnel from an inadvertent criticality. The criticality analysis does not affect or mitigate the dose consequences to the public or plant personnel from an inadvertent criticality.

There are no other SAR accidents that could be affected. Therefore, the use of the proposed criticality analysis, does not significantly increase the consequences of an accident previously evaluated.

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

No. The only purpose, or function, of Boraflex is reactivity control. Therefore, the use of the proposed criticality analysis can only result in reactivity related accidents, such as an inadvertent criticality. Though a spent fuel pool criticality accident is not discussed in detail, a calculation to ensure such an accident could not occur is referenced by both FSAR 9.3 and 9.6. Therefore, this is an accident already discussed by the SAR and dependence on a new criticality analysis does not create the possibility of an accident of a new or different kind than any previously evaluated.

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

No. The proposed analysis demonstrates that the safety function and design basis are met even for a Boraflex loss of up to 20%. Though the proposed criticality analysis methodology is more realistic, and has been licensed at other sites, it is less conservative than the existing, NRC approved analysis that is currently part of the CR–3 licensing basis. Additionally, it permits operation with a greater loss of Boraflex than the existing analysis.

The current licensing basis, BAW–2209, ‘‘Crystal River Unit 3 Spent Fuel Storage Pool Criticality Analysis’’, provides the analytical basis of both ITS LCO 3.7.14 and LCO 3.7.15. This analysis uses very conservative assumptions and methodologies, and results in very little margin remaining for identified Boraflex loss. The margin of safety, although less than previously evaluated, is not significantly reduced with reliance on the current criticality analysis. The margin of safety is restored with use of the proposed criticality analysis. Therefore, the margin of safety is not significantly reduced with use of the proposed criticality analysis.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied.

Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: November 23, 1998.

Description of amendment request: The proposed amendment would change the CR–3 Improved Technical Specifications (ITS) to raise the Engineered Safeguards Actuation System (ESAS) setpoint for reactor coolant system (RCS) low pressure from

1500 psig to 1625 psig. This change is intended to provide for earlier actuation of high pressure injection (HPI) following certain small break loss of coolant accidents and result in a lower peak center line temperature (PCT) during these transients. The applicability requirement for ESAS operability would be changed from greater than 1700 psig to greater than 1800 psig to maintain the previous margin above the ESAS setpoint. Similarly, the reactor protection system (RPS) setpoint for RCS low pressure and the RPS setpoint for Shutdown Bypass (RCS High Pressure) would each be raised by 100 psig to maintain the previous pressure margins. In addition, Surveillance Requirement 3.5.2.5 would be revised such that valves in the HPI flowpath that are throttled to balance flow between the four HPI lines would be verified in the correct position. The need for these changes resulted from planned modifications to the HPI system to improve performance and reliability of this system. Changes to ITS Bases necessitated by the system modifications and setpoint changes are included in the submittal.

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

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

The setpoint changes for reactor trip and High Pressure Injection (HPI) actuation will result in a very small (approximately one-percent) increase in the probability for reactor trips. Review of industry data shows that this increase is not significant. The revised accident analysis has determined that transients which reduce Reactor Coolant System (RCS) pressure below the new setpoints, warrant the associated action. Engineered Safeguards Actuation System (ESAS) and Reactor Protection System (RPS) actuations are used to mitigate accidents and are not the initiator of analyzed accidents. Therefore, the probability of previously evaluated accidents is not affected.

RPS and ESAS functions are assumed to actuate to mitigate transients. The revised setpoints will ensure earlier actuation of the RPS and ESAS on a low RCS pressure condition. Raising the ESAS Low RCS Pressure Setpoint will ensure earlier automatic HPI actuation for a portion of the spectrum of pressure decreasing events. For rapid depressurization events, such as main steam line break and large break Loss of Coolant Accident (LOCA), this will have little impact. For slower events, or those that do not reach the current setpoint during the initial subcooled blowdown phase, HPI will be automatically initiated substantially earlier in the event. This will increase the integrated HPI flow to the RCS during the time the core is likely to be uncovered, thereby reducing the consequential PCT. This additional flow results in a significant peak clad temperature (PCT) decrease for small break LOCA scenarios less than 0.07 square feet. Based on the above, the consequences of previously evaluated accidents will not be increased.

The HPI system characteristics will not be affected such that the probability of any accident is increased. The system flow restriction for protection from low temperature overpressure (LTOP) events will be maintained. The HPI system is used for accident mitigation and is not the initiator of evaluated accidents other than LTOP. The proposed surveillance changes will ensure that all valves throttled in the HPI flowpath are verified and secured in the correct position. The throttle valves and stop check valves will be positioned to ensure HPI flow is within analyzed limits. Therefore, the consequences of accidents that rely on HPI flow will not be increased.

Based on the above evaluation, the probability or consequences of evaluated accidents are not significantly increased by these changes.

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

The change to RPS and ESAS setpoints will not change the functions of plant equipment, no new system interactions will be created, and no new failure modes will be introduced. The setpoint changes will permit earlier actuation for the associated actions. However, no new plant conditions will be introduced by the setpoint changes.

The HPI modifications include the installation of throttle valves that will change the flow characteristics of the system. The new throttle valves are manual valves that will be secured in position. The revised surveillance requirements will ensure these valves are positioned such that HPI flow is within analyzed limits. Therefore, no conditions are created that could cause a new type of accident.

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

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

The safety function of the affected portions of the RPS and ESAS systems is to actuate their respective functions if RCS pressure drops below the setpoint. The raised RPS and ESAS setpoints will provide earlier actuation for these protective features. These changes will increase the margin of safety provided by the associated Technical Specifications.

The safety function of the HPI system is to provide cooling to limit fuel peak clad temperature. The revised surveillance requirements will ensure valves are positioned such that HPI flow is within analyzed limits. Therefore, the margin of safety provided by the HPI surveillance requirements is maintained.

Based on the above evaluation, there is no reduction in the margin of safety associated with the equipment and systems affected by this change.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied.

Therefore, the NRC staff proposes to determine that the 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. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: December 3, 1998.

Description of amendment request: The proposed change revises the TMI–1 Core Protection Safety Limits and Core Protection Safety Bases, as specified in Technical Specification Figures 2.1–1 and 2.1–3, to provide more restrictive limits which reflect the decrease in reactor coolant system flow resulting from the analysis of increased once-through steam generator (OTSG) tube plugging limits (total allowable number of tubes plugged). The licensee is currently restricted to a total of 2,000 tubes plugged in both OTSGs which corresponds to 6.4 percent of the total number of tubes. The licensee's more restrictive Core Protection Safety Limits reflect the reduction in reactor coolant flow that would exist if an average of 20 percent of the OTSG tubes were plugged.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. An increase in the average steam generator tube plugging (SGTP) level to 20% results in a small reduction of reactor coolant system (RCS) flow rates and primary to secondary heat transfer. These changes result in small changes to the primary and secondary side operating parameters, and do not result in any additional challenges to plant equipment. The proposed Technical Specification Changes resulting from the increase in allowable tube plugging limits are more restrictive but remain bounded by the existing reactor protection system (RPS) trip setpoints. The assessment of the NSSS [nuclear steam supply system] primary components, including the reactor pressure vessel, reactor core, reactor coolant pump, steam generator, pressurizer, control rod drive mechanisms, and RCS piping concluded that the integrity of these components will be unaffected by the increase in average SGTP level.

A re-analysis of the bounding Updated Final Safety Analysis Report (UFSAR) Chapter 14 accidents, specifically the startup accident, loss of coolant flow, loss of feedwater, and large and small break LOCA demonstrated compliance with the acceptance criteria. The RCS pressure boundary is not challenged, and the DNBR [departure from nucleate boiling ratio] and peak clad temperature values remain within the specified limits of the licensing basis. An analysis of the loss of electric power accident demonstrated the ability of the plant to transition smoothly to natural circulation with an average of 20% SGTP or with asymmetric plugging. It was also determined that the current mass and energy release data used for the containment integrity and equipment qualification remain bounding. Since the design requirements and safety limits continue to be met, system functions are not adversely impacted, and the integrity of the RCS pressure boundary is not challenged, the radiological consequences remain unchanged. Therefore, this activity does not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed Technical Specification changes are more restrictive core protection safety limits but remain bounded by the existing RPS trip setpoints. This proposed change assures safe operation commensurate with the effects of steam generator tube plugging. This increase in the average level of SGTP to 20% will not introduce any new accident initiator mechanisms. No new failure modes or limiting single failures have been identified. Since the safety and design requirements continue to be met and the integrity of the RCS pressure boundary is not challenged, no new accident scenarios have been created. This change does not add any new equipment, modify any interfaces with existing equipment, or change the equipment function or the method of operating the equipment. Reactor core, RCS, and steam generator parameters remain within appropriate design limits during normal operation. Therefore, this activity does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The existing RPS trip setpoints bound the proposed Technical Specification changes resulting from 20% SGTP. This change assures safe operation commensurate with the effects of steam generator tube plugging. The TMI–1 DNB design basis, RCS pressure limits, peak clad temperature limits and dose criteria are maintained for all UFSAR transients. Therefore, this activity does not reduce the margin of safety.

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

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

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

NRC Project Director: Cecil O. Thomas.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2 (NMP2), Oswego County, New York

Date of amendment request: November 16, 1998.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) related to the implementation of systems for the detection and suppression of coupled neutronic/thermal-hydraulic instabilities in the reactor. Average Power Range Monitor (APRM) flow control trip reference cards will initiate a reactor scram to limit the oscillation magnitude at reactor trip so as to limit the associated Critical Power Ratio change and, in conjunction with Minimum Critical Power Ratio (MCPR) operating limits, assure compliance with the MCPR safety limit. In addition, the changes would increase the APRM flow biased neutron flux scram and control rod block settings to allow plant operation in the Extended Load Line Limit Analysis region. Thus, the proposed changes are in regard to setpoints and calculations for fuel cladding integrity and the associated TS Bases. In the Bases for TS 2.1.1, the proposed change would reference new equations in TS 2.1.2a. In TS 2.1.2a, the proposed change would be to the equation for determining the flow biased APRM scram and rod block trip setpoints. In the Bases for TS 2.1.2a, the proposed change would reflect the new setpoints. In the Bases for TS 2.2.2, the proposed change would be to the description of the setpoint methodology which is based upon General Electric Report NEDC–31336, ''GE Instrumentation Setpoint Methodology.'' In Note (m) of TS Table 3.6.2/4.6.2, the proposed change would be to the calibration range for the APRM channel setpoint. In the Bases for TS 3.6.2/4.6.2, the proposed change would be to the equations and methodology for determining APRM scram and rod block setpoints. In TS 6.9.1.f, which identifies documents approved by NRC for analytical methods used to determine core operating limits, the proposed change would add ''NEDO–32465–A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, August 1996.''

Basis for proposed no significant hazards consideration determination:

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

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The APRM neutron monitoring system is not an initiator or a precursor to an accident. The neutron monitoring system monitors the power level of the reactor core and provides automatic core protection signals in the event of a power transient. A Restricted Region will be maintained such that the probability of a stability event is not increased. Therefore, the proposed TS changes cannot affect the probability of a previously evaluated accident.

The proposed TS changes will revise the APRM flow-biased neutron flux scram TS setting to provide automatic protection to assure that anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits. The proposed changes will result in a more restrictive APRM flow-biased scram trip setting in the low flow regions of the power/flow operating map (i.e., operational conditions where reactor instabilities are most probable). In other words, the new settings will provide a scram sooner (at a lower power level) than the existing settings. The associated control rod block setting will also be revised. A margin between the control rod block and flux scram has been determined by calculation.

The proposed changes will also revise the APRM flow-biased neutron flux scram and control rod block TS settings to provide an increase above the current values in operating conditions not susceptible to reactor instabilities. Specifically, the proposed changes will implement a 2% increase in the analytical limit of the APRM flow-biased flux scram and a 7% increase in the analytical limit of the APRM flow-biased control rod block. Evaluation demonstrates that these proposed analytical limit increases have negligible impact on the transient events results for NMP1 [Nine Mile Point Unit 1] as documented in Chapter XV of the NMP1 UFSAR, [Updated Final Safety Analysis Report], including the limiting transient events which are reanalyzed each reload. Of the twenty-five (25) transient events analyzed in Section XV of the NMP1 UFSAR, only the Inadvertent Startup of Cold Recirculation Loop event and the Recirculation Flow Controller Malfunction—Increase Flow event have potentially impacted results. The Chapter XV Control Rod Drop Accident as well as the Turbine Trip with No Bypass at Partial Power event were also evaluated.

For the Inadvertent Startup of Cold Recirculation Loop event, the proposed 2% increase in the high neutron flux scram would result in an increase in the fuel average surface heat flux response. However, there is significant margin between the surface heat flux value for this event and the current limiting MCPR [Minimum Critical Power Ratio] event (the Feedwater Controller Failure Maximum Demand event). As such, any small change to the fuel surface heat flux response due to the high neutron flux scram analytical limit increase would not result in the fuel thermal margin requirements for the Inadvertent Startup of Cold Recirculation Loop event to exceed the MCPR limits set by the limiting reload analysis event.

The reactor neutron flux for the Recirculation Flow Controller Malfunction—Increase Flow event also showed an increasing trend from its initial value. However, the peak response for this parameter (104% of rated) is significantly below the high neutron flux scram analytical limit. Accordingly, the proposed increase to the high neutron flux scram analytical limit does not affect the response to this transient event.

The Control Rod Drop Accident is included in Chapter XV of the NMP1 UFSAR. As noted in NEDE–24011–P–A, ''GESTAR II: General Electric Standard Application for Reactor Fuel,'' the initial power burst from this event is terminated by the Doppler reactivity feedback while the scram provides the final event termination several seconds later. The 120% APRM scram limit was conservatively chosen. The time delay introduced by the small change in analytical limit will be inconsequential due to the extremely rapid power rise for this event (i.e., the time of scram for a 120% analytical limit vs. a 122% analytical limit is essentially the same).

The proposed Bases changes to TS 3.6.2/4.6.2 and TS 2.2.2 simply provide details of the setpoint methodology currently used as well as specific allowable values.

Therefore, the proposed TS changes to implement a more restrictive flow-biased scram setting to protect against reactor instabilities and the proposed change to increase the high neutron flux scram and rod block analytical limits do not result in a significant increase in the consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will revise the APRM flow-biased neutron flux scram TS settings to assure anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits in the low flow regions of the power/flow operating map as well as revise the associated control rod block settings. These changes also propose a 2% increase in the analytical limit of the APRM flow-biased neutron flux scram and a 7% increase in the analytical limit of the APRM flow-biased control rod block. These changes do not introduce any new accident precursors and do not involve any alterations to plant configurations which could initiate a new or different kind of accident. The proposed changes do not affect the intended function of the APRM system nor do they affect the operation of the system in a way which would create a new or different kind of accident.

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

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

More conservative APRM flow-biased neutron flux scram and control rod block settings will be implemented in the low flow regions of the power/flow operating map. The scram setting change will assure that anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits. The proposed changes will also implement a 2% increase in the APRM flow-biased neutron flux scram and a 7% increase in the APRM flow-biased control rod block in those operating regions not susceptible to reactor instabilities. Evaluation demonstrates that these proposed increases have negligible impact on the transient events or accident results for NMP1. The impacted transient events are either not the limiting MCPR event, the peak response to the event is significantly below the high neutron flux scram analytical limit or in the case of the Control Rod Drop Accident, the time delay introduced by the change will be inconsequential due to the extremely rapid power rise. No other events are adversely affected. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: S. Singh Bajwa.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2 (NMP2), Oswego County, New York

Date of amendment request: November 19, 1998.

Description of amendment request: The proposed amendment would change the surveillance frequencies in Technical Specifications (TSs) 4.8.4.4a, ''Surveillance Requirements—Reactor Protection System Electric Power Monitoring (RPS Logic),'' and 4.8.4.5a, ''Surveillance Requirements—Reactor Protection System Electric Power Monitoring (Scram Solenoids),'' to require channel functional testing of the RPS Motor Generator Set (M/G) and RPS Uninterruptible Power Supplies (UPS) Electrical Protection Assemblies (EPAs) at least once every 6 months. These TSs currently require that channel functional testing be performed each time the plant is in cold shutdown for a period of more than 24 hours, unless performed within the previous 6 months.

Basis for proposed no significant hazards consideration determination: During the last refueling outage, the licensee modified the Nine Mile Point Unit No. 2 (NMP2) design for the RPS M/G and RPS UPS EPAs to provide relay actuated protection systems. The relays of the new design may be individually isolated from an essential power circuit for testing and may be actuated without tripping the associated breaker. The relay actuated system will allow the EPA system monitoring an essential power supply to be functionally tested with the plant on-line. The EPA relay actuation setpoints are not affected by the modification or the proposed TS changes. The licensee states that the design, installation, and testing of the new units meet the criteria of the same standards that were applied to the previous units.

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

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes affect surveillance testing frequency only. The new relay actuated protection system design functions in the same fail safe manner as the old units. Also, the new design in conjunction with the testing capability has increased EPA reliability, while introducing little risk to testing the EPAs with the plant in operation. Therefore, the proposed changes to the NMP2 TS do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes affect surveillance testing frequency of relay actuated protection circuits only. The proposed changes do not introduce any new or different accident initiators from any that were previously evaluated. EPA relay actuation setpoints are not affected. The actual fail safe system conditions required for EPA actuation will remain the same. Therefore, the operation of NMP2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The function of the EPA systems is to isolate the loads from supply power. That function was not altered by the proposed change. Reliability of the EPA systems is improved. Therefore, the operation of NMP2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: S. Singh Bajwa.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit 1 (NMP1), Oswego County, New York

Date of amendment request: November 30, 1998.

Description of amendment request: The proposed amendment would correct Technical Specification (TS) 3.1.2, ''Liquid Poison System,'' and the associated TS Bases. Specifically, in the Bases for TS 3.1.2, the boron-10 concentration of 120 ppm (which is incorrectly calculated using atomic percent instead of weight percent) would be changed to 109.8 ppm. In TS 3.1.2, the minimum volume of the sodium pentaborate solution contained in the Liquid Poison System storage tank would be increased from 1185 gallons to 1325 gallons.

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

1. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Liquid Poison System is designed to provide the capability to bring the reactor from a full design rating to a shutdown condition assuming none of the control rods can be inserted. The system is manually initiated in response to a failure of the Control Rod Drive System to shutdown the reactor. The proposed changes revise the required liquid poison solution volume and concentration. The proposed changes to the Technical Specifications and the Bases require no changes to the physical facility which could adversely affect any accident precursors. Therefore, the proposed changes cannot significantly increase the probability of an accident.

The proposed changes will assure that the Liquid Poison System continues to provide the capability to shutdown the reactor during an ATWS [Anticipated Transient Without Scram] event. In addition, the system will continue to be capable of bringing the reactor to cold shutdown, 3 percent delta k subcritical (0.97 k_{eff}), from a full design rating of

1850 megawatts thermal assuming none of the control rods can be inserted, and considering the combined effects of coolant voids, temperature change, fuel doppler, and xenon and samarium. Therefore, the change to the Technical Specifications does not significantly increase the consequences of a previously evaluated accident.

2. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Injection of the sodium pentaborate solution into the reactor vessel has been considered in the plant design. The proposed changes revise the required liquid poison solution volume and concentration. The proposed changes make no physical modification to the plant which could create the possibility of a new or different kind of accident. The proposed changes will maintain the capability of the Liquid Poison System to shutdown the reactor from its full design rating assuming none of the control rods are inserted, and considering the combined effects of coolant voids, temperature change, fuel doppler, and xenon and samarium. Consequently, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed changes revise the required liquid poison solution volume and concentration. The proposed changes make no physical modification to the plant which could reduce the margin of safety. These changes will assure compliance with the requirements of 10CFR50.62, ''Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.'' In addition, these changes will maintain the capability of the Liquid Poison System to bring the reactor from a full design rating of 1850 megawatts thermal to greater than 3 percent delta k subcritical (0.97 k_{eff}) assuming none of the control rods can be inserted, and considering the combined effects of coolant voids, temperature change, fuel doppler, xenon and samarium.

The required volume of boron-10 solution in the Liquid Poison System storage tank includes an additional 25 percent margin beyond the amount needed to shutdown the reactor to allow for any unexpected non-uniform mixing. Also, the total storage tank volume of sodium pentaborate solution incorporates 197 gallons of solution which is unavailable for injection into the reactor vessel and a 25 gallon margin for conservatism. Additionally, using one 30 gpm Liquid Poison System pump, the injection time is greater than 17 minutes thereby assuring adequate mixing. The proposed changes to the liquid poison concentration and volume ensure the NMP1 [Nine Mile Point Unit 1] Liquid Poison System is able to meet its safety function requirements. Therefore, this change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: S. Singh Bajwa.

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: December 4, 1998.

Description of amendment request: The proposed amendment would eliminate the need to cycle the plant and its components through a shutdown-startup cycle by allowing the next snubber surveillance interval to be deferred until the end of refueling outage 6 or September 10, 1999, whichever date is earlier.

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 10CFR50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

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

The proposed change is for a one time extension to the surveillance interval of snubber inspections required by Technical Specification 4.7.10.e. The change involves revising the calendar time for snubber interval inspections to 36 months to coincide with the time frame of the current cycle 6 operation.

Snubber testing experience at Millstone Unit No. 3 has shown that historical failure rates of snubbers are low. During the third refueling outage, after an operating cycle of approximately 22 months, the functional testing program identified multiple Type A failures attributed primarily to original plant construction, and resulted in a full inspection of all Type A snubbers. The snubber inspection interval was extended to approximately 30 months by a one-time extension to the Technical Specifications for the fourth refueling outage and only one Type A snubber failure was identified. Subsequent outages with operating durations of 18 and 17 months also identified only a single Type B failure in each outage. The results of piping stress analysis which have been performed to assess the impact of snubbers which have failed to meet functional test acceptance criteria have shown that neither piping system functionality or structural integrity have ever been compromised.

During the recent cycle 6 operation Millstone 3 has experienced an extended midcycle shutdown, where temperature, vibration effects and normal wear on snubbers have been minimized as compared to a normal operating cycle. The last snubber surveillance interval inspections were completed during this midcycle shutdown. Although the calendar surveillance interval is impacted by this change the primary conditions that present challenges to snubbers have not been prevalent during the extended shutdown. Given the low failure rates of snubbers over the last 3 surveillance intervals, and the fact the operating time of the remainder of cycle 6 will be approximately 1 year, snubber failures are expected to be similar to previous intervals.

Accordingly the possibility of a snubber failure leading to a Decrease in Reactor Coolant Inventory or a Decrease in Heat Removal by the Secondary System is not increased and there is no affect on the probability of previously evaluated accidents.

This change does not include any physical changes to the plant and does not affect acceptance criteria or the required actions for functional failures of snubbers. Accordingly there is no increase in the consequences of previously evaluated accidents resulting in a Decrease in Reactor Coolant Inventory or a Decrease in Heat Removal by the Secondary System.

Thus it is concluded that 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.

This proposed revision to the surveillance interval does not change the operation of any plant system or component during normal or accident conditions. The proposed change extends the surveillance interval of snubber inspections required by Technical Specification 4.7.10.e. The change involves revising the calendar time for snubber interval inspections to coincide with the time frame of current cycle 6 operation. This change does not include any physical changes to the plant and does not affect acceptance criteria or the required actions for functional failures of snubbers.

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

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

The proposed change extends the surveillance interval of snubber inspections required by Technical Specification 4.7.10.e. The change involves revising the calendar time for snubber interval inspections to coincide with the time frame of current cycle 6 operation. This change does not include any physical changes to the plant and does not affect acceptance criteria or the required actions for functional failures of snubbers. The service life of the snubbers or parts as required by Technical Specification 4.7.10.i will not be impacted by this change since the required replacements have already occurred and no additional service life dates will expire prior to September 10, 1999.

Thus, it is concluded that 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 Project Director: William M. Dean.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: November 24, 1998.

Description of amendment request: The proposed amendment would revise the Ginna Station Improved Technical Specifications description of the fuel cladding material (TS 4.2.1) and to update the list of references provided in Specification 5.6.5 for the Core Operating Limits Report.

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

Evaluation of Administrative Changes

The administrative changes [related to the update of references provided in Specification 5.6.5 for the Core Operating Limits report] do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes revise Administrative Controls Section 5.6.5.b to update the references to NRC approved documents which support the analysis for the Heat Flux Hot Channel Factor in the Core Operating Limits Report and to provide clarification to the currently applicable methodology. It revises the Design Features Section 4.2.1 to provide clarification of the types of zirconium alloy filler rod material that have received previous NRC approval and to clarify that the application shall be NRC approved. Section 4.2.1 is revised to clarify that the analyses performed to verify compliance with the fuel safety design bases shall be cycle specific. As such, these changes are administrative in nature and do not impact initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed administrative changes do not affect the manner by which the plant is operated and no new equipment will be installed. The proposed administrative changes will not impose any new or different requirements. All original design and performance criteria continue to be met, and no new failure modes have been created for any system, component, or piece of equipment. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because the methodology has been shown to meet all applicable design criteria and ensure that all pertinent licensing basis acceptance criteria are met. As such, no question of safety is involved, and the changes do not involve a significant reduction in a margin of safety.

Evaluation of Less Restrictive Changes

The less restrictive change [related to the fuel cladding material (TS 4.2.1)] does not involve a significant hazards consideration as discussed below:

(1) Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The Westinghouse 14×14 VANTAGE + fuel assemblies containing fuel rods fabricated with ZIRLO alloy meet the same fuel assembly and fuel rod design bases as Westinghouse 14×14 OFA [Optimized Fuel Assembly] fuel assemblies in the other fuel regions. In addition, the 10 CFR 50.46 criteria will be applied to the fuel rods fabricated with ZIRLO alloy. The use of these fuel assemblies will not result in a change to the proposed Ginna Westinghouse 14×14 OFA reload design and safety analysis limits. The ZIRLO alloy is similar in chemical composition and has similar physical and mechanical properties as that of Zircaloy-4. Thus the cladding integrity is maintained and the structural integrity of the fuel

assembly is not affected. The ZIRLO clad fuel rods improve corrosion resistance and dimensional stability. The use of ZIRLO does not impact the radiological consequences of accidents previously evaluated in the Safety Analysis. The RCS [reactor coolant system] isotopic inventory is negligibly impacted; therefore, changes in postulated releases from the RCS or the secondary systems are negligible. Assumptions of fuel melting in the radiological analyses are not based on the type of fuel cladding. For those accidents where fuel melting is postulated to occur (control rod ejection, locked [seized] RCP rotor), the amount of fuel undergoing melting and clad damage using ZIRLO clad is bounded by the current values used in the Safety Analysis. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

(2) Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The Westinghouse 14×14 VANTAGE + fuel assemblies containing fuel rods fabricated with ZIRLO alloy will satisfy the same design bases as that used for Westinghouse 14×14 OFA fuel assemblies in the other fuel regions. Since the original design criteria is being met, the fuel rods fabricated with ZIRLO alloy will not be an initiator for any new accident. All design and performance criteria will continue to be met and no new single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alterations to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

(3) Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The Westinghouse 14×14 VANTAGE + fuel assemblies containing fuel rods fabricated with ZIRLO alloy do not change the proposed Ginna Westinghouse 14×14 OFA reload design and safety analysis limits. The use of these fuel assemblies containing fuel rods fabricated with ZIRLO alloy will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core, these fuel assemblies will be specifically evaluated using approved reload design methods and approved fuel rod design models and methods as specified in Technical Specifications. This will include consideration of the core physics analysis peaking factors and core average linear heat rate effects. In addition, the 10 CFR 50.46 criteria will be applied each cycle to the fuel rods fabricated with ZIRLO alloy. Analyses or evaluations will be performed each cycle to confirm that 10 CFR 50.46 will be met. Therefore, the margin of safety as defined in the Bases to the Ginna Technical Specifications is not significantly reduced.

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: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Project Director: S. Singh Bajwa.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362,

San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: November 23, 1998.

Description of amendment requests: The proposed change would revise the Technical Specifications (TS) to (1) reinstate the log power reactor trip at or above 4E–5% RATED THERMAL POWER (RTP); (2) reinstate reactor trips for Reactor Coolant Flow—Low (RCS flow), the Local Power Density—High (LPD), and the Departure from Nucleate Boiling Ratio—Low (DNBR); (3) remove the word ''automatically'' from notes (a) and (d) of Table 3.3.1–1 to clarify that the manual enable of the trip is permissible; and, (4) clarify that the setpoints on Table 3.3.1–1 are set relative to logarithmic power, not thermal power.

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

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

The proposed change to TS 3.3.1 does not adversely impact structure, system, or component design or operation in a manner which would result in a change in the frequency of occurrence of accident initiation. SCE has re-analyzed the relevant accidents and established that accident consequences are not significantly increased by the proposed changes to the bypass-permissive and enable setpoints. The reactor trip bypass and automatic enable functions are not accident initiators. Consequently, the proposed TS change will not significantly increase the probability of accidents previously evaluated. Therefore, this amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No new or different accidents result from changing the reactor trip bypass-permissive and automatic enable setpoints. Introducing an uncertainty band for the enable setpoints delays the mitigation action of the reactor trip for the design basis analysis for the events that credit this trip. The enable setpoint itself does not cause any accident. Therefore, the amendment request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

SCE [Southern California Edison Company] has re-analyzed the accidents and determined that the consequences of the accidents are within their acceptance criteria under the proposed amendment so that the margin of safety that bounds the setpoint in both directions remains intact. The analyses are relatively insensitive to the reactor trip automatic enable setpoints, and no significant reduction in the margins of safety ensues from the relatively minor proposed changes to the bypass-permissive and enable setpoints, nor from establishing allowable values for these points.

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 requests involve no significant hazards consideration.

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: November 23, 1998.

Description of amendment request: The proposed amendment relocates descriptive design information from Technical Specification 3/4.7.1.1 (Table 3.7–2), regarding orifice sizes for main steam line Code safety valves, to the Bases section.

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

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

The proposed change relocates the orifice size design information for the main steam line Code safety valves, found in Table 3.7–2, that does not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The affected descriptive design information is not related to any assumed initiators of analyzed events and is not assumed to mitigate accident or transient events. The limiting condition for operation for the main steam line Code safety valves is not altered by the proposed change. The orifice size design information will be relocated from Table 3.7–2 of Specification 3/4.7.1.1 to the Bases section for that same Technical Specification and will be maintained pursuant to 10 CFR 50.59. In addition, surveillance testing details for this Technical Specification are addressed in existing surveillance procedures, which are also controlled by 10 CFR 50.59, and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates the orifice size design information for the main steam line Code safety valves, found in Table 3.7–2, that does not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates the orifice size design information for the main steam line Code safety valves, found in Table 3.7–2, that does not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. In addition, the relocated orifice size design information remains the same as the existing Technical Specifications. Since any future changes to this orifice size information (that will be located in the Bases section) will be evaluated per the requirements of 10 CFR 50.59, there is no reduction in a margin of safety.

The proposed change is also consistent with the Westinghouse Plants (Improved) Standard Technical Specification, NUREG–1431, approved by the NRC Staff. Revising the Technical Specification to reflect the approved content of NUREG–1431 ensures no significant reduction in the margin of safety. Therefore, the 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

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: November 11, 1998.

Brief description of amendments: The proposed amendments revise core safety limit curves and Overtemperature N–16 reactor trip setpoints based on analyses of the core configuration and expected operation for Comanche Peak Steam Electric Station (CPSES) Unit 2, Cycle 5. The changes apply equally to CPSES Units 1 and 2 licenses since the Technical Specifications are combined.

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?

A. Revision to the Unit 2 Core Safety Limits

Analyses of reactor core safety limits are required as part of reload calculations for each cycle. TU Electric has performed the analyses of the Unit 2, Cycle 5 core configuration to determine the reactor core safety limits. The methodologies and safety analysis values result in new operating curves which, in general, permit plant operation over a similar range of acceptable conditions. This change means that if a transient were to occur with the plant operating at the limits of the new curve, a different temperature and power level might be attained than if the plant were operating within the bounds of the old curves. However, since the new curves were developed using NRC approved methodologies which are wholly consistent with and do not represent a change in the Technical Specification BASES for safety limits, all applicable postulated transients will continue to be properly mitigated. As a result, there will be no significant increase in the consequences, as determined by accident analyses, of any accident previously evaluated.

B. Revision to Unit 2 Overtemperature N–16 Reactor Trip Setpoints

As a result of changes discussed, the Overtemperature reactor trip setpoint has been recalculated. These trip setpoints help ensure that the core safety limits are protected and that all applicable limits of the safety analysis are met.

Based on the calculations performed, no significant changes to the safety analysis values for Overtemperature reactor trip setpoint were required. The f(delta I) trip reset function was revised due to less top-skewed axial power distributions predicted for this cycle. The analyses performed show that, using the TU Electric methodologies, all applicable limits of the safety analysis are met. This setpoint provides a trip function which allows the mitigation of postulated accidents and has no impact on accident initiation. Therefore, the changes in safety analysis values do not involve an increase in the probability of an accident and, based on satisfying all applicable safety analysis limits, there is no significant increase in the consequences of any accident previously evaluated.

In addition, sufficient operating margin has been maintained in the overtemperature setpoint such that the risk of turbine runbacks or unnecessary reactor trips due to upper plenum flow anomalies or other operational transients will be minimized, thereby, reducing potential challenges to the plant safety systems.

C. Administrative changes to reflect plant nomenclature

Changes to the N–16 trip setpoint equation are for clarification only to more accurately reflect CPSES plant nomenclature. This change is administrative in nature and does not increase in the probability or consequences of an accident previously evaluated.

Summary

The changes in the amendment request apply NRC approved methodologies to changes in safety analysis values, new core safety limits and new N–16 setpoint and parameter values to assure that all applicable safety analysis limits have been met. The potential for an operational transient to occur has not been affected and there has been no significant impact on the consequences of any accident previously evaluated.

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 calculation of new reactor core safety limits and overtemperature reactor trip setpoint resets. As such, the changes play an important role in the analysis of postulated accidents but none of the changes effect plant hardware or the operation of plant systems in a way that could initiate an accident. Changes to the N–16 trip setpoint equation are for clarification only to more accurately reflect CPSES plant nomenclature. 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?

In reviewing and approving the methods used for safety analyses and calculations, the NRC has approved the safety analysis limits which establish the margin of safety to be maintained. While the actual impact on safety is discussed in response to question 1, the impact on margin of safety is discussed below:

A. Revision to the Unit 2 Reactor Core Safety Limits

The NRC-approved TU Electric reload analysis methods have been used to determine new reactor core safety limits. All applicable safety analysis limits have been met. The methods used are wholly consistent with Technical Specification BASES 2.1 which is the bases for the safety limits. In particular, the curves assure that for Unit 2, Cycle 5, the calculated DNBR is no less than the safety analysis limit and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The acceptance criteria remains valid and continues to be satisfied; therefore, no change in a margin of safety occurs.

B. Revision to Unit 2 Overtemperature N–16 Reactor Trip Setpoints

Because the reactor core safety limits for CPSES Unit 2, Cycle 5 are recalculated, the Reactor Trip System instrumentation setpoint values for the Overtemperature N–16 reactor trip setpoint which protect the reactor core safety limits must also be recalculated. The Overtemperature N–16 reactor trip setpoint helps prevent the core and Reactor Coolant System from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. The most relevant design basis analysis in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) which is affected by the Overtemperature reactor trip setpoint is the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2). This event has been analyzed with the new safety analysis value for the Overtemperature reactor trip setpoint to demonstrate compliance with event specific acceptance criteria. Because all event acceptance criteria are satisfied, there is no degradation in a margin of safety.

The nominal Reactor Trip System instrumentation setpoint values for the Overtemperature N–16 reactor trip setpoint (Technical Specification Table 2.2–1) are determined based on a statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. The total uncertainty plus additional margin is applied in a conservative direction to the safety analysis trip setpoint value to arrive at the nominal and allowable values presented in Technical Specification Table 2.2–1. Meeting the requirements of Technical Specification Table 2.2–1 assures that the Overtemperature reactor trip setpoint assumed in the safety analyses remains valid. The CPSES Unit 2, Cycle 5 Overtemperature reactor trip setpoint is not significantly different from the previous cycle, and thus provides operational flexibility to withstand mild transients without initiating automatic protective actions. Although the value of the f(delta I) trip reset function setpoint is different, the Reactor Trip System instrumentation setpoint values for the Overtemperature N–16 reactor trip setpoint are consistent with the safety analysis assumptions which have been analytically demonstrated to be adequate to meet the applicable event acceptance criteria. Thus, there is no reduction in a margin of safety.

Using the NRC approved TU Electric methods, the reactor core safety limits are determined such that all applicable limits of the safety analyses are met. Because the applicable event acceptance criteria continue to be met, there is no significant reduction in the margin of safety.

C. Administrative changes to reflect plant nomenclature

Changes to the N–16 trip setpoint equation are for clarification only to more accurately reflect CPSES plant nomenclature. This change is administrative in nature and has no impact 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(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of 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.

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

Date of amendment request: December 10, 1998.

Description of amendment request: The licensee proposed to correct an error in the technical specifications by changing to the use of ''hydrogen, balance air'' rather than the incorrect ''hydrogen balance nitrogen'' for calibration of the Augmented Offgass System hydrogen monitors.

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:

Based on the criteria for defining a significant hazards consideration in 10CFR50.92, operation of VYNPS in accordance with this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated, because:

The proposed change is purely administrative in nature—correcting instrument calibration requirements to conform the Technical Specification with the instrument manufacturer's recommendations. The change has no effect on plant hardware, plant design, safety limit setting, or plant system operation and therefore does not modify or add any initiating parameters that would significantly increase the probability or consequences of an accident previously evaluated. This change to the Technical Specifications is a correction of an error which occurred when the particular Technical Specification was issued. The function of this surveillance requirement remains unchanged.

No new modes of operation are introduced by the proposed change such that adverse consequences would result. Accordingly, the consequences of previously analyzed accidents are not affected by this proposed change.

The Augmented Off-Gas (AOG) System hydrogen monitors do not serve a reactor safety function. In this context, the determination of no significant hazards consideration defined in 10CFR50.92 is made based on the ''accident previously evaluated'' being a postulated hydrogen detonation within the off-gas system downstream of the hydrogen recombiners. The hydrogen monitors do not mitigate the consequences of an accident, but rather function to preclude a hydrogen explosion within the off-gas system. The function of the Augmented Off-Gas System hydrogen monitors to prevent a hydrogen detonation is not affected by this change.

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

Since this change merely corrects Technical Specification wording to reflect the actual manufacturer's recommended gas mixture to be used for calibrating these instruments, no new or different types of accidents are created. Since the calibration gas mixture has a very low (approximately 2%) hydrogen concentration, its use does not introduce the possibility of fires, explosions, or other hazards which might adversely affect safety-related equipment. Therefore, use of the proper calibration gas does not create the possibility of a new or different kind of accident.

This change does not affect the operation of any systems or components, nor does it involve any potential initiating events that would create any new or different kind of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated for the Vermont Yankee Nuclear Power Station.

(3) Involve a significant reduction in a margin of safety, because:

This proposed change involving the specification of the correct calibration gas mixture ensures that the off-gas system hydrogen monitors are properly calibrated and therefore preserve the margin of safety in precluding a hydrogen explosion in the off-gas system. Administratively changing this specification only establishes the appropriate calibration gas for the actual, installed hydrogen monitors. Changing the specification to reflect correct practice will not reduce the margin of safety.

The proposed change does not affect any equipment involved in potential initiating events or safety limits. Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Project Director: Cecil O. Thomas.

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: July 30, 1998 (TSCR 206).

Description of amendment request: The purpose of the proposed amendments is to incorporate changes to the Technical Specifications to more clearly define the requirements for Service Water (SW) System operability.

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 in accordance with the proposed amendment[s] does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The Service Water System is primarily a support system for systems required to be operable for accident mitigation. Portions of the SW system supplying the containment fan coolers also function as part of the containment pressure boundary under post accident conditions. Failures within the SW system are not an initiating condition for any analyzed accident.

Analyses performed demonstrate that under the Technical Specifications allowable configurations, the SW system will continue to perform all required functions. The SW system is capable of supplying the required cooling water flow to systems required for accident mitigation. That is, the SW system removes the required heat from the containment fan coolers and residual heat removal heat exchangers ensuring containment pressure and temperature profiles following an accident are as evaluated in the FSAR [final safety analysis report]. This in turn ensures that environmental qualification of equipment inside containment is maintained and thus function as required post-accident.

SW system response post accident is within all design limits for the system. Transient and steady state forces within the system remain within all design and operability limits thereby maintaining the integrity of the system inside containment and the integrity of the containment pressure boundary. Assumptions dependent on containment pressure profile for containment leakage assumed in the radiological consequence analyses remain valid.

In addition, removing required heat from containment ensures that cooling of the reactor core is accomplished for long-term accident mitigation.

Therefore, operation of the SW system as proposed will not result in a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes do not alter the way in which the SW system performs its design functions nor the design limits of the system. The proposed changes do not introduce any new or different normal operation or accident mitigation functions for the system. Therefore, no new accident initiators are introduced by the proposed changes. Operation of SW system as proposed cannot result in a new or different kind of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

Analyses performed in support of the proposed amendments demonstrate that the SW system continues to perform its function as assumed and credited in the accident analyses and radiological consequence analyses performed for the Point Beach Nuclear Plant. Therefore, the analyses and results are not changed. All analysis limits remain met. The SW system continues to be operated and responds within all design limits for the system. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments cannot result in a reduction in a margin of safety.

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

Local Public Document Room location: 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: September 23, 1998 (TSCR 209).

Description of amendment request: The purpose of the proposed amendments is to remove the test requirements for snubbers from the Technical Specifications (TS). These requirements are already included in the Point Beach Nuclear Plant In-Service Inspection 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. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of an accident previously evaluated.

These changes do not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to delete TS 15.4.3. The proposed TS change does not introduce any new accident initiators.

Initiating conditions and assumptions are unchanged and remain as previously analyzed for accidents in the PBNP Final Safety Analysis Report. The proposed TS change does not involve any physical changes to systems or components, nor does it alter the typical manner in which the systems or components are operated. Therefore, these changes do not increase the probability of previously evaluated accidents.

As noted above, the snubber testing requirements included in the ASME/ANSI OM–4 Code are more comprehensive and in general more conservative than the snubber testing requirements currently contained in TS 15.4.13.

These changes do not involve a significant increase in the consequences of an accident or event previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. The snubber program ensures that snubbers function as required, therefore related systems continue to function as designed and analyzed. Existing system and component redundancy and operation is not being changed by these proposed changes. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated. Therefore, these changes do not affect the consequences of previously evaluated accidents.

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.

These changes do not introduce nor increase the number of failure mechanisms of a new or different type than those previously evaluated since there are no physical changes being made to the facility. As noted above, the snubber testing requirements included in the ASME code in general are more comprehensive than the snubber testing requirements currently contained in TS 15.4.13 and provide the requisite level of assurance of snubber operability. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not introduced.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant reduction in the margin of safety because existing component redundancy is not being changed by these proposed changes. There are no changes to the initial conditions contributing to accident severity or consequences, and safety margins established through the design and facility license including the Technical Specifications remain unchanged. Therefore, there are no significant reductions in a margin of safety introduced by [these] proposed amendment[s].

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: October 7, 1998 (TSCR 207).

Description of amendment request: The purpose of the proposed amendments is to incorporate changes to the Technical Specifications (TS) to ensure the 4 kV bus undervoltage input to reactor trip is controlled in accordance with the design and licensing basis for the facility. One additional administrative change is requested which removes the footnote related to the definition of Rated Power in TS 15.1.j.

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

1. 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 changes proposed ensure the Point Beach Nuclear Plant continues to be operated in accordance with the design and licensing basis for the facility.

The first change removes a footnote qualifying the definition of Rated Power as applied to PBNP Unit 2. This restriction was eliminated with the replacement of Unit 2 steam generators as approved by Amendments 173 and 177, dated July 1, 1997. The analyses for those amendments were performed based on the minimum flow requirements specified in Technical Specification 15.3.1.G.3. The note should have been deleted from the Technical Specifications at that time. Elimination of this note does not result in a change in the operation of PBNP from that analyzed and approved in Amendments 173 and 177. Therefore, this change is administrative and cannot result in an increase in probability or consequences of an accident previously evaluated.

The second change modifies the Limiting Condition For Operation [LCO] for the undervoltage reactor trip protection function. This trip function is the primary protective function credited in the complete loss of flow event analysis in the Final Safety Analysis Report (FSAR) Section 14.1.8. As a primary protective function, this trip is required to be single failure proof as stipulated in proposed IEEE 279–1968 documented in FSAR Section 7.2. This change ensures that this protective feature is maintained in a condition where single failure considerations are satisfied. When single failure criteria cannot be met, appropriate action is stipulated to shutdown the unit placing it in a condition where the protective function is no longer required. Therefore, this change ensures PBNP is operated in accordance with its design and licensing basis and cannot result in an increase in the probability or consequences of an accident previously evaluated.

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.

The changes proposed by this request remove a footnote qualifying the definition of rated power as it applies to PBNP Unit 2 operation, and modify the LCO related to the undervoltage reactor trip protective function to ensure this function is maintained as required by the PBNP design and licensing basis. These changes are in agreement with approved analyses. These changes do not introduce any new accident initiators or alter the response of the PBNP Units to previously analyzed accidents. Therefore, operation of PBNP in accordance with the proposed changes cannot result in a new or different kind of accident from any accident previously evaluated.

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

Operation of the PBNP in accordance with the proposed amendments is within the bounds of approved design and licensing basis of the facility. The design and licensing basis establish appropriate margins of safety. Since operation of the PBNP remains within the approved design and licensing basis of the facility, a reduction in a margin of safety cannot result.

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 Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: November 18, 1998

Description of amendment request: The proposed amendment would revise the pressure/temperature (P/T) limits and the low-temperature overpressure protection (LTOP) requirements in the facility technical specifications.

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

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

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

Failure of a reactor vessel is not an accident that has been previously evaluated; design provisions ensure that this is not a credible event. Since the potential consequences of a reactor vessel failure are so severe, industry and governmental agencies have worked together to ensure that failure will not occur. Compliance with 10 CFR 50 Appendix G and H ensures that failure of a reactor vessel will not occur. The proposed changes do not impact the capability of the reactor coolant pressure boundary piping (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore do not increase the potential for the occurrence of a LOCA [loss-off-coolant accident].

The LTOP setpoint, revised enabling temperature, and revised P/T limits reflected in proposed Figures TS 3.1–1 and TS 3.1–2 ensure that the Appendix G pressure/temperature limits are not exceeded, and therefore, ensure that RCS integrity is maintained. The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, construction standards, or setpoints. The reactor coolant system full power operating pressure (2235 psig) is not being changed by this proposed amendment. The LTOP valve setpoint remains at less than or equal to 500 psig. The LTOP enabling temperature based on Figure

TS 3.1–2 is 200°F and is consistent with ASME Code Case N–514 guidance of RT_{NDT} + 50°F. The revised enabling temperature is lower than the 355°F value in the current TS. However, the allowable combination of Appendix G pressures and temperatures (refer to the 0°F isothermal cooldown limit) is greater for the revised limit curves. The combination of greater allowable Appendix G pressure and temperature limits and lower enabling temperature produces a larger operating window. A larger operating window reduces the likelihood of inadvertently lifting the LTOP relief valve while maneuvering the plant through the knee of the P–T curve during startup and shutdown. The probability of an LTOP event occurring is independent of the pressure-temperature limits for the RCS [reactor coolant system] pressure boundary and enabling temperature. Therefore, the probability of a[n] LTOP event is not increased.

The revised heatup and cooldown limit curves and LTOP enabling temperature were developed using test results from unirradiated and/or irradiated specimens that represent the KNPP [Kewaunee Nuclear Power Plant] reactor vessel beltline circumferential weld, closure head flange, and intermediate forging. The circumferential beltline weld and intermediate forging are the most limiting materials in the reactor coolant pressure boundary due to the effects of neutron irradiation which cause the flow properties to increase and the toughness to decrease. 10 CFR 50, Appendix G states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. Drop weight and Charpy V-notch testing of IP3571 weld metal and the intermediate forging material has been performed and used for derivation of the revised PTS [pressurized thermal shock] assessment, the proposed Appendix G heatup and cooldown limit curves, and the corresponding LTOP system enabling temperature. The revised limit curves and corresponding LTOP enabling temperature have been developed using accepted engineering practices, methods derived from the ASME Boiler and Pressure Vessel Code, criteria set forth in NRC Regulatory Standard Review Plan 5.3.2, and 10 CFR 50.61. Utilization of the revised heatup and cooldown limit curves and corresponding LTOP enabling temperature ensures adequate fracture toughness for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary. These limit curves provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, and low temperature overpressure protection (corresponding to isothermal events during low temperature operations (i.e., less than or equal to 200°F)) thus ensuring the integrity of the reactor coolant pressure boundary.

The changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Radiological off-site exposures from normal operation and operational transients, and faults of moderate frequency do not exceed the guidelines of 10 CFR 100. In addition, the changes do not affect any fission product barrier. The changes do not degrade or prevent the response of the LTOP relief valve or other safety-related systems to previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. Therefore, the consequences of an accident previously evaluated will not be increased.

Thus, operation of KNPP in accordance with the PA does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Since the potential consequences of a reactor vessel failure are so severe, industry and governmental agencies have worked together to ensure that failure will not occur. Compliance with 10 CFR 50 Appendix G and H ensures that failure of a reactor vessel will not occur. The proposed heatup and cooldown limit curves have been constructed by combining the most conservative pressure-temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves. With NRC approval to use Code Case N–588, the intermediate forging and closure head flange become the controlling materials for development of the heatup limit curve and the cooldown limit curves at low temperatures. At high temperatures, the circumferential weld continues to be limiting for development of the cooldown limit curves. Use of conservative pressure-temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves, does not modify the reactor coolant system pressure boundary, nor make any physical changes to the LTOP setpoint or design. Proposed Figures TS 3.1–1 and TS 3.1–2 were prepared in accordance with regulatory and code requirements and were derived using more conservative material property basis and more limiting requirements of neutron exposure projections thru 33 EFPY [effective full-power years] instead of 20 EFPY.

The revised LTOP system enabling temperature and the proposed Appendix G pressure temperature limitations were prepared using methods derived from the ASME Boiler and Pressure Vessel Code and the criteria set forth in NRC Regulatory Standard Review Plan 5.3.2. The changes do not cause the initiation of any accident nor create any new credible limiting failure for safety-related systems and components. The changes do not result in any event previously deemed incredible being made credible. As such, it does not create the possibility of an accident different than previously evaluated.

The changes do not have any adverse effect on the ability of the safety-related systems to perform their intended safety functions. The combination of higher allowable Appendix G pressure and temperature limits and lower enabling temperature produces a larger operating window. The ASME Section XI, Working Group on Operating Plant Criteria (WGOPC) has prepared a technical bases document for Code Case N–514. The technical bases document is contained in Attachment 3 of Reference 1. This technical bases document provides justification for enabling the LTOP system at temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater.

WGOPC, which has responsibility for Appendix G of Section XI, has considered the burden and safety impact imposed by the LTOP criteria, and has developed Code guidelines for determining the LTOP set-point pressure and the required enabling temperature. These guidelines will relieve some operational restrictions, yet provide adequate margins against failure for the reactor vessel. Further, by relieving the operational restrictions, these guidelines result in a reduced potential for activation of pressure relieving devices, thereby improving plant safety. Thus, a slightly larger operating window at KNPP is viewed to reduce the likelihood of inadvertently lifting the LTOP relief valve while maneuvering the plant through the knee of the P–T curve during startup and shutdown. The new LTOP operating window (i.e., less than or equal to 200°F) is within the existing operating band for the residual heat removal system; operating procedures allow the LTOP system to be placed into service at <400°F. At KNPP, as long as the LTOP relief valve is operable, the LTOP system is enabled anytime the residual heat removal system is in communication with the reactor coolant system.

The proposed changes do not make physical changes to the plant or create new failure modes. Thus, the PA does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Involve a significant reduction in the margin of safety.

The proposed Appendix G pressure temperature limitations and LTOP enabling temperature were prepared using methods derived from the ASME Boiler and Pressure Vessel Code, including Code Cases N–514 and N–588, and the criteria set forth in NRC Regulatory Standard Review Plan 5.3.2. Reference 1 to this letter provides information to support NRC approval to use Code Case N–514 and Code Case N–588 for the KNPP PTS evaluation, development of the heatup and cooldown limit curves, and establishment of the LTOP system enabling temperature. These documents and practices along with the calculational limitations specified in 10 CFR 50.61 are an acceptable method for implementing the requirements of 10 CFR 50 Appendices G and H.

Use of the methodology set forth in the ASME Boiler and Pressure Vessel Code, NRC Regulatory Standard Review Plan 5.3.2., 10 CFR 50.61, and 10 CFR 50 Appendices G and H ensures that proper limits and safety factors are maintained. Thus, the PA does not involve a significant reduction in the margin of safety.

The revised heatup and cooldown limit curves and LTOP system enabling temperature were prepared using drop weight and Charpy V-notch data for the beltline weld, closure head flange, and intermediated forging material along with practices described herein and methods derived from the ASME Boiler and Pressure Vessel Code and 10 CFR 50.61. The safety factors and margins used in the development of the limit curves and LTOP system enabling temperature meet the criteria set forth by these documents. Application of low leakage core designs decreases the rate of shift in transition temperature from ductile to nonductile behavior. The revised limit curves and LTOP enabling temperature provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, and low temperature overpressure protection (corresponding to isothermal events during low temperature operations (i.e., less than or equal to 200°F)). With the preparation of the revised limit curves in accordance with the latest criteria and guidance, this PA ensures that proper limits and safety factors are maintained.

Thus, the PA 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: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001.

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

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.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: May 15, 1998, as supplemented September 25 and October 13, 1998.

Brief description of amendment: The amendment would revise Technical Specification 5.5, ''Storage of Unirradiated and Spent Fuel'' to reflect a planned modification to increase the number of fuel assemblies that can be stored in the spent fuel pool from 2776 to 4086.

Date of publication of individual notice in Federal Register: November 24, 1998 (63 FR 64973).

Expiration date of individual notice: December 24, 1998.

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

Notice of 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.

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

Date of application for amendments: October 16, 1998.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.3.1 ''Reactor Protective System (RPS) Instrumentation-Operating'' and TS 3.3.2, ''Reactor Protective System (RPS) Instrumentation-Shutdown,'' to clarify an inconsistency between the TS wording and the design bases as described in the TS Bases and the Updated Final Safety Analysis Report. Specifically, the change replaces the operating bypass input process variable, Thermal Power, in Footnotes (a), (b), and (d) of Table 3.3.1 and in the Note to Limiting Condition for Operation 3.3.2 with Nuclear Instrument Power.

Date of issuance: December 8, 1998.

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

Amendment Nos.: 229 & 204.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 27, 1998 (63 FR 57320).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 25, 1996, as supplemented on September 5, 1996, August 8, 1997, March 26, July 31, and August 24, 1998.

Brief description of amendment: This amendment revises Technical Specifications (TSs) 3/4.5.F.1, ''Core and Containment Cooling systems'' to extend the allowed outage time (AOT) for the emergency diesels, TSs 3.9.B.1 and 3.9.B.4, ''Auxiliary Electrical System'' to reduce the AOT from 7 days to 3 days and reduce the AOT for the combination of an EDG and startup transformer or shutdown transformer from 72 hours to 48 hours, and add Configuration Risk Management Program in TS 5.5, ''Programs and Manuals'' of Section 5.0 ''Administrative Controls''. Various TS pages were re-numbered in Section 5.0. In addition, TSs 3.9, ''Auxiliary Electrical System,'' and 3.9.A, ''Auxiliary Electrical Equipment,'' have been reformatted to be consistent with TS 3.9.B approved in a previous amendment. The associated Bases sections have also been changed to reflect the new TSs.

Date of issuance: December 11, 1998.

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

Amendment No.: 179.

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

Date of initial notice in Federal Register: September 23, 1998 (63 FR 50934).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 11, 1998.

No significant hazards consideration comments received: No.

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

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: July 15, 1997, as supplemented March 3, April 13, June 16, October 26, and November 5, 1998.

Brief description of amendments: The amendments revised the Technical Specifications to add new requirements for the main steamline break instrumentation and resolved issues related to Inspection and Enforcement Bulletin 80–04.

Date of Issuance: December 7, 1998.

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

Amendment Nos.: 234—Unit 1; 234—Unit 2; 233—Unit 3.

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

Date of initial notice in Federal Register: September 24, 1997 (62 FR 50001).

The March 3, April 13, June 16, October 26, and November 5, 1998, letters provided clarifying information that did not change the scope of the July 15, 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 December 7, 1998.

No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of application for amendment: September 24, 1998, as supplemented November 3, 1998.

Brief description of amendment: This amendment revised technical specification 3.1.2.8 in two places to change the term ''contained volume'' to usable volume.'' This change eliminates the potential for a non-conservative interpretation of the specification values for the Refueling Water Storage Tank and Boric Acid Storage Tank and thereby eliminates the need for temporary administrative controls, which have been used correctly to properly interpret the specification values as usable volumes.

Date of issuance: December 14, 1998.

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

Amendment No: 95.

Facility Operating License No. NPF–73. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 4, 1998 (63 FR 59591).

The November 3, 1998, letter did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 14, 1998.

No significant hazards consideration comments received: No.

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

Illinois Power Company, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: August 17, 1998.

Brief description of amendment: The amendment reduces the load at which diesel generators are tested.

Date of issuance: December 14, 1998.

Effective date: December 14, 1998.

Amendment No.: 118.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 7, 1998 (63 FR 53949).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 14, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

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

Date of application for amendments: August 1, 1997.

Brief description of amendments: The amendments delete a portion of a technical specifications surveillance test requirement that specifies that the steam driven auxiliary feedwater pumps be tested ''when the secondary steam supply pressure is greater than 310 psig.'' This removes any misunderstanding that the secondary steam pressure must be just above 310 psig for this test.

Date of issuance: December 10, 1998.

Effective date: December 10, 1998, with full implementation within 45 days.

Amendment Nos.: 225 and 209.

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

Date of initial notice in Federal Register: December 31, 1997 (62 FR 68308).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 10, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 5, 1998.

Brief description of amendment: This amendment changes the Technical Specifications to update the terminology and references to 10 CFR 50.55a(f) and (g) consistent with the 1989 edition of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, and consistent with the second 10-year interval of the Inservice Inspections and Inservice Testing Program Plans.

Date of issuance: December 3, 1998.

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

Amendment No.: 84

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

Date of initial notice in Federal Register: March 11, 1998 (63 FR 11920).

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

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

Date of application for amendments: August 8, 1996, as supplemented June 30, 1997 and August 26, 1998.

Brief description of amendments: The amendments eliminate the response time testing requirements for selected sensors and specified instrument loops for (1) the reactor protection system, (2) the isolation system, and (3) the emergency core cooling system.

Date of issuance: December 14, 1998.

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

Amendment Nos.: 132 and 93.

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

Date of initial notice in Federal Register: November 6, 1996 (61 FR 57489).

The June 30, 1997 and August 26, 1998, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 14, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

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

Date of application for amendment: July 10, 1998, as supplemented October 16, 1998.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.6/4.6 and associated bases to relocate portions of the reactor coolant chemistry to the Updated Final Safety Analysis Report and to applicable plant procedures. Changes to the relocated requirements will be controlled by the provisions of 10 CFR 50.59.

Date of issuance: December 1, 1998.

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

Amendment No.: 247.

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

Date of initial notice in Federal Register: July 29, 1998 (63 FR 40560).

The October 16, 1998, submittal fell with the scope of, and did not change, the initial proposed finding of no significant hazards consideration.

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

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: March 30, 1998, as supplemented on October 27, 1998.

Brief description of amendment: The amendment revises the definition of logic system functional tests, and revises test frequency requirements for certain instrumentation.

Date of issuance: December 11, 1998.

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

Amendment No.: 248.

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

Date of initial notice in Federal Register: April 22, 1998 (63 FR 19978).

The October 27, 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 December 11, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 12, 1998, as supplemented on October 12, 1998. The October 12, 1998, letter provided clarifying information that did not change the initial proposed no sigificant hazards consideration determination.

Brief description of amendments: The amendments revise TS 3/4.6.1.3, ''Containment Air Locks,'' to change the action statements for an inoperable air lock. The amendments also revise TS Bases 3/4.6.1.2, ''Containment Leakage,'' to correct an editorial error and TS Bases 3/4.6.1.3, ''Containment Air Locks,'' to provide additional details regarding the air locks.

Date of issuance: December 2, 1998.

Effective date: December 2, 1998.

Amendment Nos: 215 and 195.

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

Date of initial notice in Federal Register: September 9, 1998 (63 FR 48265).

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

No significant hazards consideration comments received: No.

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

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: December 31, 1997, as supplemented by letter dated September 11, 1998.

Brief Description of amendments: The amendments revised the Technical Specifications (TSs) to change the intermediate range neutron flux reactor trip setpoint and allowable value, and delete the reference to the reactor trip setpoints in TS 3.10.3, ''Special Test Exceptions—Physics Tests,'' and TS 3.10.4, ''Special Test Exceptions—Reactor Coolant Loops.''

Date of issuance: December 8, 1998.

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

Amendment Nos.: Unit 1—140; Unit 2—132.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

Date of initial notice in Federal Register: February 11, 1998 (63 FR 6998).

The September 11, 1998, letter provided clarifying information that did not change December 31, 1997, application or the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama.

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: September 20, 1996 (TS 96–09).

Brief description of amendments: The amendments change the Technical Specifications to clarify the types of work shifts that are acceptable when considering the requirements to ensure overtime is not heavily used on a routine basis by unit staff.

Date of issuance: December 7, 1998.

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

Amendment Nos.: 240 and 230.

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

Date of initial notice in Federal Register: November 4, 1998 (63 FR 59596).

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

No significant hazards consideration comments received: No.

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

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: August 22, 1998, as supplemented on August 27 and October 8, 1998 (TS 96–08). The August 27, 1998, amendment request superseded the original (August 22, 1998) request in its entirety.

Brief description of amendments: The amendments revise the Sequoyah Nuclear Plant Technical Specifications by extending the allowed outage time for the SQN emergency diesel generators from 72 hours to 7 days.

Date of issuance: December 16, 1998.

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

Amendment Nos.: 241 and 231.

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

Date of initial notice in Federal Register: October 9, 1996 (61 FR 52969), superseded by a second notice on September 9, 1998 (63 FR 48270). The October 8, 1998, letter provided clarifying information that did not change the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 28, 1998, as supplemented October 16, 1998. The October 16, 1998, letter was administrative in nature and did not change the initial no significant hazards consideration determination.

Brief description of amendments: The amendments revise the Technical Specifications to change the Emergency Diesel Generator section to be consistent with station procedures associated with steady-state conditions.

Date of issuance: December 10, 1998.

Effective date: December 10, 1998.

Amendment Nos.: 216 and 197.

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

Date of initial notice in Federal Register: September 9, 1998 (63 FR 48272).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 10, 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 23rd day of December 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–34440 Filed 12–29–98; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–40826; File No. SR–NASD–98–80]

Self-Regulatory Organizations; Notice of Filing of Proposed Rule Change and Amendment No. 1 Thereto by the National Association of Securities Dealers, Inc. Relating to the Issuance of Temporary Cease and Desist Orders

December 22, 1998.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act''),[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on October 28, 1998, the National Association of Securities Dealers, Inc. (''NASD'' or ''Association''), through its regulatory subsidiary, NASD Regulation, Inc. (''NASD Regulation'') filed with the Securities and Exchange Commission (''SEC'' or ''Commission'') the proposed rule change as described in Items I, II, and III below, which Items have been prepared by NASD Regulation. The Association amended the proposal on December 15 and 16, 1998.[3] The Commission is publishing this notice to solicit comments on the proposed rule change, as amended, from interested persons.

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.

[3] The first amendment to the proposal included changes to the evidentiary standard and the tenure of a temporary cease and desist order. *See* Letter from Alden S. Adkins, Senior Vice President and General Counsel, NASD, to Katherine A. England, Assistant Director, Division of Market Regulation (''Division''), Commission, dated December 15, 1998. On December 16, 1998, the NASD made further non-substantive changes to the proposed rule language at a meeting between Peter Geraghty, Assistant General Counsel, NASD Regulation, and Mandy S. Cohen, Special Counsel, and Anitra T. Cassas, Attorney, Division, Commission. *See* Memorandum entitled: Meeting with Staff of NASD regulation, dated December 17, 1998. The NASD also agreed to extend the public comment period to sixty days by letter dated December 21, 1998. *See* Letter from Alden S. Adkins, Senior Vice President and General Counsel, NASD, to Katherine A. England, Assistant Director, Divisions, Commission.

I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

The Association is proposing to create the Rule 9800 Series and to amend certain existing NASD Rules of the Association to establish procedures to enable the Association to issue temporary cease and desist orders. The proposed rule change also would grant the NASD authority to initiate non-summary proceedings when temporary or permanent cease and desist orders are violated. The text of the proposed rule change follows. Additions are *italicized*; deletions are [bracketed].[4]

8300. Sanctions

8301. Sanctions for Violation of the Rules

(a) Imposition of Sanctions

After compliance with the Rule 9000 Series, the Association may impose one or more of the following sanctions on a member or person associated with a member for each violation of the federal securities laws, rules or regulations thereunder, the rules of the Municipal Securities Rulemaking Board, or Rules of the Association, or may impose one or more of the following sanctions on a member or person associated with a member for any neglect or refusal to comply with an order, direction, or decision issued under the Rules of the Association:

(5) suspend or bar a member or person associated with a member from association with all members; [or]

(6) [impose any other fitting sanction.]*impose a temporary or permanent cease and desist order against a member or a person associated with a member; or*

(7) impose any other fitting sanction.

* * * * *

IM–8310–2. Release of Disciplinary Information

* * * * *

(d)(1) The Association shall release to the public information with respect to any disciplinary decision issued pursuant to the Rule 9000 Series imposing a suspension, cancellation or expulsion of a member; or suspension or revocation of the registration of a person associated with a member; or barring of a member or person associated with a member from association with all members; or imposition of monetary sanctions of $10,000 or more upon a member or person associated with a member; or containing an allegation of a violation of a Designated Rule; and may also release such information with respect to any disciplinary decision or group of decisions that involve a significant policy or enforcement determination where the release of information is deemed by the President of NASD Regulation, Inc. to be in the public interest. The Association also may release to the public information with respect to any disciplinary decision issued pursuant to the Rule 8220 Series imposing a suspension or cancellation of the member or a suspension of the association of a person with a member, unless the National Adjudicatory Council determines otherwise. The National Adjudicatory Council may, in its discretion, determine to waive the requirement to release information with respect to a disciplinary decision under those extraordinary circumstances where the release of such information would violate fundamental notions of fairness or work an injustice. *The Association also shall release to the public information with respect to any temporary cease and desist order issued pursuant to the Rule 9800 Series.*

* * * * *

(h) If a final decision of the Association is not appealed to the Commission, the sanctions specified in the decision (other than bars, [and] expulsions, *permanent cease and desist orders, and temporary cease and desist orders*) shall become effective on a date established by the Association but not before the expiration of 30 days after the date of the decision. Bars, [and] expulsions, *permanent cease and desist orders, and temporary cease and desist orders,* however, shall become effective upon issuance of the decision, unless the decision specifies otherwise. *An appeal to the Commission of a decision that imposes a permanent cease and desist order or a temporary cease and desist order shall not stay the effectiveness of such orders, unless the Commission specifies otherwise.*

9000. CODE OF PROCEDURE

9100. Application and Purpose

* * * * *

9120. Definitions

* * * * *

(x) ''Party''

With respect to a particular proceeding, the term ''Party'' means:

(1) in the Rule 9200 Series, [and] the Rule 9300 Series, *and the Rule 9800 Series,* the Department of Enforcement or a Respondent;

* * * * *

9200. DISCIPLINARY PROCEEDINGS

* * * * *

9240. Pre-Hearing Conference and Submission

9241. Pre-Hearing Conference

* * * * *

(c) Subjects to be Discussed

At a pre-hearing conference, the Hearing Officer *shall schedule an expedited proceeding if required by Rule 9290, and* may consider and take action with respect to any or all of the following:

* * * * *

[4] Language in proposed rules IM–8310–2, 9360, 9500, 9510, 9511, and 9513 includes changes proposed in File No. SR–NASD–98–56. *See* Securities Exchange Act Release No. 34–40378 (August 27, 1998), 63 FR 47058 (September 3, 1998). Language in proposed rule 9120 includes changes proposed in File No. SR–NASD–98–90. *See* Securities Exchange Act Release No. 34–40755 (December 7, 1998), 63 FR 68814 (December 14, 1998). For purposes of this notice, the proposed rule language in File Nos. SR–NASD–98–56 and 98–90 is treated as approved.