consultants, and staff. Persons desiring to make oral statements should notify Mr. Sam Duraiswamy, Chief of the Nuclear Reactors Branch, at least five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Chief of the Nuclear Reactors Branch prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Chief of the Nuclear Reactors Branch if such rescheduling would result in major inconvenience.

In accordance with subsection 10(d) Pub. L. 92–463, I have determined that it is necessary to close portions of this meeting noted above to discuss matters that relate solely to the internal personnel rules and practices of this Advisory Committee per 5 U.S.C. 552b(c)(2), and to discuss information the release of which would constitute a clearly unwarranted invasion of personal privacy per 5 U.S.C. 552b(c)(6).

Further information regarding topics to be discussed, 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 Mr. Sam Duraiswamy, Chief of the Nuclear Reactors Branch (telephone 301/415–7364), between 7:30 a.m. and 4:15 p.m. EST.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at *http://www.nrc.gov/ACRSACNW*.

Video teleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician, (301–415–8066) between 7:30 a.m. and 3:45 p.m. EST at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment facilities that they use to establish the video teleconferencing link. The availability of video teleconferencing services is not guaranteed.

The ACRS meeting dates for Calendar Year 1999 are provided below:

ACRS meeting No.	1999 ACRS meeting date
	January 1999—No meeting.
459	February 4–6, 1999.
460	March 10 (1:00 p.m.)–13, 1999.
461	April 7 (1:00 p.m.)–10, 1999.
462	May 5 (1:00 p.m.)–8, 1999.
463	June 2–4, 1999.
464	July 7–9, 1999.
	August 1999—No meeting.
465	September 1–3, 1999.
466	September 30–October 2, 1999.
467	November 4–6, 1999.
468	December 2–4, 1999.

Dated: November 13, 1998.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 98–30870 Filed 11–17–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Planning and Procedures

The ACRS Subcommittee on Planning and Procedures will hold a meeting on December 2, 1998, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, December 2, 1998—2:00 p.m.

The Subcommittee will discuss proposed ACRS activities and related matters. It may also discuss the status of appointment of a new member to the ACRS. 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.

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 in schedule, etc., that may have occurred.

Dated: November 13, 1998.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 98–30871 Filed 11–17–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 24, 1998, through November 5, 1998. The last biweekly notice was published on November 4, 1998 (63 FR 59584).

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 December 18, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's

Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: October 16, 1998.

Description of amendment request: The proposed amendments would lower the power level below which the turbine control valve (TCV) and turbine stop valve (TSV) closure scram signals and the end-of-cycle recirculation pump trip (EOC-RPT) signals are not in effect. The bypass setpoint (P_{bypass}) would be reduced from 30 percent rated power to 25 percent rated power. The licensee also proposes to delete the reference to turbine first stage pressure as a measure of core thermal power in the Technical Specifications. To ensure that the trip functions will not be inadvertently bypassed when they are required to be operable, a requirement would be added to periodically verify that TCV and TSV scram trip functions and the ECO-RPT trip functions are not bypassed at greater than or equal to 25 percent of rated 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) Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated:

The probability of an accident previously evaluated will not increase as a result of this change because the setpoint change does not alter any of the initiators of an accident or cause them to occur more frequently.

The consequences of an accident previously evaluated are not impacted. LaSalle Units 1 and 2 each have approximately 30 percent bypass capability. Therefore, a scram on TCV or TSV closure signals is not needed until 30 percent core thermal power is reached, as adequate steam bypass capacity is available. A lower P_{bypass} remains conservative with respect to this criterion.

LaSalle utilizes power and flow dependent thermal limits. The power dependent portion of these thermal limits is dependent on the P_{bypass} setpoint. These limits provide assurance that adequate fuel thermal-mechanical margin is maintained through adherence to the thermal limits Technical Specification requirements.

Revised thermal limits have been determined based on the results of GE transient analyses. Adhering to these thermal limits ensures that the consequences of an accident or transient would not be increased from the consequences under the approved 30 percent setpoint. Adjustments to the thermal limits were determined through use of the NRC-approved ODYN reactor dynamic model for the limiting Load Rejection Without Bypass and the Feedwater Controller Failure events.

The deletion of the reference to turbine first stage pressure and rewording the Technical Specifications Notes does not affect either accident initiators or plant equipment, as they are administrative changes.

Adding the periodic verification that the bypass channels are set correctly ensures that scrams or EOC-RPT will not be inadvertently bypassed when Thermal Power is greater than or equal to 25 percent of Rated Thermal Power. The statement that specification 4.0.2 applies to the 18 month interval is needed, since the notes are not standard surveillance requirements and the interval is consistent with other similar instrumentation to which 4.0.2 currently applies.

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

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

The setpoint change and proposed bypass verification notes ensure that the scrams for TSV closure and TCV fast closure, and EOC–RPT, will be enabled above 25 percent of rated thermal power, rather than above 30 percent of rated thermal power. This change results in simplified reload transient analyses and does not impact any other equipment.

No other physical modifications are being proposed by this submittal. The only plant operational impact is that between 25 percent and 30 percent power, the plant will now scram upon a turbine trip, which is an analyzed transient.

The remaining changes to Technical Specification wording are administrative in nature and consistent with other Technical Specifications.

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

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

LaSalle Units 1 and 2 each have approximately 30 percent bypass capability. Therefore, a scram on TCV or TSV closure signals in not needed until 30 percent core thermal power is reached, as adequate steam bypass capacity is available. However, reduction of this setpoint to 25 percent power actually aids the plant transient response between 25 percent and 30 percent power.

The new thermal limits reflect the revised setpoint and have been determined based on revised limiting transient analyses that have included the new P_{bypass} value. If a transient were to occur, the revised operating limits ensure that adequate margin would be available to preclude violation of the Minimum Critical Power Ratio (MCPR) safety limit and the fuel thermal-mechanical limits.

All other UFSAR [Updated Final Safety Analysis Report] events are either bounded by the analyses performed or are not impacted by the P_{bypass} change.

The wording changes to the Technical Specifications do not change the requirement for the bypass function and for maintaining the bypass function and thus do not affect the analyses discussed above.

The addition of the Notes periodically verifying the TCV and TSV Closure Trip Functions are not bypassed at greater than or equal to 25 percent Rated Thermal Power ensures the trip functions will not be inadvertently bypassed when required to be Operable.

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

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

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

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: July 22 and October 22, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to reflect the licensee's planned use of fuel supplied by Westinghouse. The

Westinghouse fuel has different design characteristics from the fuel currently in use. Accordingly, the following changes would need to be made to the TS: Figure 2.1.1–1, ''Reactor Core Safety Limits—Four Loops in Operation''; various core operating parameters specified by Surveillance Requirements 3.2.1.2, 3.2.1.3, and 3.2.2.2; Section 4.2.1, ''Fuel Assemblies''; and Section 5.6.5, ''Core Operating Limits Report (COLR).''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, addressing the three standards of 10 CFR 50.92(c):

First Standard

Implementation of this LAR [license amendment request] would not involve a significant increase in the probability or consequences of an accident previously evaluated. The revised Reactor Core Safety Limits Figure further restricts acceptable operation. Moving an uncertainty factor from the Improved Technical Specifications to the Core Operating Limits Report (COLR) does not exempt this factor from regulatory restrictions. COLR parameters are generated by NRC approved methods with the intent of ensuring that previously evaluated accidents remain bounding. The COLR is submitted to the NRC upon implementation of each fuel cycle or when the document is otherwise revised. No accident probabilities or consequences will be impacted by this LAR.

Second Standard

Implementation of this LAR would not create the possibility of a new or different kind of accident from any previously evaluated. The revised Reactor Core Safety Limits Figure further restricts acceptable operation. Moving an uncertainty factor from the Improved Technical Specifications to the COLR does not exempt this factor from regulatory restrictions. Since the parameter in question is not being deleted, the possibility of a new or different kind of accident from any previously evaluated does not exist.

Third Standard

Implementation of this LAR would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. Use of the ZIRLO™ cladding material has been reviewed and approved in Reference 1 (as listed in Chapter 2.1 of Topical Report DPC–NE–2009/DPC–NE–2009P, Duke Power Company Westinghouse Fuel Transition Report). ZIRLO™ cladding has been extensively used in Westinghouse nuclear reactors. The changes proposed in this LAR are necessary to ensure that the performance of the fission product barriers (cladding) will not be impacted following the replacement of one fuel design for another. No safety margin will be significantly impacted.

The NRC staff reviewed the licensee's analysis, and agrees 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: York County Library, 138 East Black Street, Rock Hill, South Carolina.

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: June 18, 1996. This notice supersedes the notice published on July 31, 1996 (61 FR 40015) in its entirety.

Description of amendment request: For Beaver Valley Power Station, Unit No. 1 (BVPS–1) only, the proposed amendment would revise Technical Specification (TS) 4.4.5 and associated Bases; the Bases for TS 3/4.4.6.2 would also be revised. The proposed changes are editorial in nature and are intended to provide consistency between the TSs and associated Bases. Index page XIX would be revised to reflect the revision of page numbers for TS Tables 4.4–1 and 4.4–2 due to shifting of text.

For Beaver Valley Power Station, Unit No. 2 (BVPS–2) only, the proposed amendment would implement a voltage-based repair criteria for steam generator tubes similar to the changes approved for BVPS–1 by License Amendment No. 198. The proposed changes are intended to reflect the guidance provided in NRC Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.'' The proposed changes would revise TSs 4.4.5 and 3.4.6.2 and associated Bases. TS Table 4.4–2 would be revised to reference TS 6.6 for reporting requirements.

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

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

Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate (TSP). Test data indicates that tube burst cannot occur within the TSP, even for tubes which have 100% throughwall electric discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Since tube-to-TSP proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition, main steamline break (MSLB) pressure differential. The Regulatory Guide (RG) 1.121 criterion requiring maintenance of a safety factor of 1.43 times the MSLB pressure differential on tube burst is satisfied by 7⁄8″ diameter tubing with bobbin coil indications with signal amplitudes less than 8.6 volts, regardless of the indicated depth measurement.

The upper voltage repair limit (V_{URL}) will be determined prior to each outage using the most recently approved NRC database to determine the tube structural limit (V_{SL}). The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{NDE}) and growth (V_{GR}) to establish V_{URL}. Using the Generic Letter (GL) 95–05 NDE and growth allowances for an example, the NDE uncertainty component of 20% and a voltage growth allowance of 30% per full power year can be utilized to establish a V_{URL} of 5.7 volts. The 20% NDE uncertainty represents a square-root-sum-of-the-squares (SRSS) combination of probe wear uncertainty and analyst variability. The degradation growth allowance should be an average growth rate or 30% per effective full power year, whichever is larger.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated MSLB outside of containment but upstream of the main steam isolation valve (MSIV) represents the most limiting radiological condition relative to the plugging criteria. In support of implementation of the revised plugging limit, analyses will be performed to determine whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary-to-secondary leakage would result in postulated site boundary and control room doses exceeding 10 CFR 100, 10 CFR 50 Appendix A, and GDC–19 [General Design Criterion-19] requirements, respectively. A separate calculation has determined the maximum allowable MSLB leakage limit in a faulted loop. This limit was calculated using the technical specification reactor coolant system (RCS) Iodine-131 activity level of 1.0 microcuries per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG–0800. The projected MSLB leakage rate calculation methodology prescribed in Section 2.b of GL 95–05 will be used to calculate the end-of-cycle (EOC) leakage. Projected EOC voltage distribution will be developed using the most recent EOC eddy current results and considering an appropriate voltage measurement uncertainty. The log-logistic probability of leakage correlation will be used to establish the MSLB leakrate used for comparison with the faulted loop allowable limit. Therefore, as implementation of the voltage-based repair criteria does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

The proposed changes to the BVPS–1 Index, Specifications and associated Bases and the proposed change to BVPS–2 Table 4.4–2 are editorial in nature. Therefore, these changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

Implementation of the proposed steam generator tube voltage-based repair criteria does not introduce any significant changes to the plant design basis. Use of the voltage-based repair criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations as no outside diameter stress corrosion cracking (ODSCC) is occurring outside the thickness of the tube support plates. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging limit has been applied (during all plant conditions).

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

The single through-wall crack lengths that result in tube burst at 1.43 times the MSLB pressure differential and the MSLB pressure differential alone are approximately 0.57 inch and approximately 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of approximately 0.41 inch long cracks at nominal leak rates and approximately 0.62 inch long cracks at the lower 95% confidence level leak rates. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during MSLB conditions, the leakage from the maximum permissible crack must preclude tube burst at MSLB conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for MSLB conditions using the lower 95% leakrate data. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection. Analyses have shown that only a small percentage of the TSPs are deflected greater than the TSP thickness during a postulated MSLB.

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

The proposed change to BVPS–1 Index, Specifications and associated Bases and the proposed change to BVPS–2 Table 4.4–2 are editorial in nature. These changes do not change the performance of plant systems, plant configuration or method of operating the plant.

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

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

The use of the voltage-based repair criteria at BVPS–2 maintains steam generator tube integrity commensurate with the criteria of RG 1.121. This guide describes a method acceptable to the Commission for meeting GDCs 14, 15, 30, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be repaired or removed from service. Upon implementation of the proposed criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the tube support plate elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences remain within the licensing basis.

In addressing the combined effects of loss-of-coolant-accident (LOCA) + safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse. There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to complete through-wall cracks during tube deformation or collapse.

The results of an analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation. Since the leak-before-break methodology is applicable to the reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes the pressurizer spray line break. Analysis results have demonstrated that no tubes were subject to deformation or collapse. No tubes have been excluded from application of the subject voltage-based steam generator tube repair criteria.

Addressing RG 1.83 considerations, implementation of the voltage-based repair criteria is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, the bobbin coil inspection will include 100% of the hot-leg TSP intersections and cold-leg intersections down to the lowest cold-leg TSP with known ODSCC, the determination of the TSPs having ODSCC will be based on the performance of at least 20% random sampling of tubes inspected over their full length, and rotating pancake coil inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

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

The proposed change to the BVPS–1 Index, Specifications and associated Bases and the proposed change to BVPS–2 Table 4.4–2 are editorial in nature. These changes will not reduce the margin of safety because they have no impact on any safety analysis assumptions.

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

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

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

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts &

Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Robert A. Capra.

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

Date of amendment request: October 15, 1998.

Description of amendment request: The proposed amendment would make several changes that are administrative in nature. The changes would (1) make editorial changes to delete obsolete material or material adequately described elsewhere, change action statement numbers, update the technical specification (TS) index pages, and make changes to be consistent with the guidance of the improved standard technical specifications (ISTS); (2) delete reporting requirements that duplicate reporting requirements contained in 10 CFR; and (3) relocate the requirement for meteorological monitoring instrumentation from the TS to the Licensing Requirements Manual.

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

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

a. This change deletes an expired Unit 1 license condition and a Unit 2 license requirement that is not required since it is redundant to the reporting requirements addressed in 10 CFR 50.73. Deleting these requirements does not involve any increase in the probability or consequences of an accident previously evaluated.

b. The reference to Specification 3.0.6 was omitted from Specification 3.0.1 in Unit 1 Amendment 213 and Unit 2 Amendment 90 and is being added to 3.0.1 to be consistent with the Improved Standard Technical Specifications of NUREG 1431. This does not involve any increase in the probability or consequences of an accident previously evaluated.

c. The Core Alteration definition has been updated to be consistent with the regulations and ISTS. The Offsite Dose Calculation Manual (ODCM) definition has been updated to be consistent with the change to Administrative Control 6.9.3. The Members of the Public definition has been changed to be consistent with 10 CFR 20.1003. This does not involve any increase in the probability or consequences of an accident previously evaluated.

d. Changing Table 3.3–6 Action Statement 36 to Action Statement 35 is an editorial change to eliminate redundant use of action statement numbers. This does not involve any increase in the probability or consequences of an accident previously evaluated.

e. The technical specification index is being revised to address removal of the Meteorological Monitoring specification and title and page number changes to the administrative control reporting requirements section. The Meteorological Monitoring specification is being relocated to the Licensing Requirements Manual (LRM). Relocating the Meteorological Monitoring requirements is in accordance with the guidance in the Commission's Final Policy Statement and revisions to 10 CFR 50.36 on the content of the technical specifications and the ISTS. The Meteorological Monitoring requirements do not meet any of the criteria, 1 thru 4 of 10 CFR 50.36 and can, therefore, be relocated from the Technical Specifications to the LRM. These changes do not involve any increase in the probability or consequences of an accident previously evaluated.

f. The exclusion area boundary is adequately described in each unit's UFSAR [Updated Final Safety Analysis Report], therefore, design feature 5.1 Site Location is also being modified by deleting the description of the exclusion area boundary. This does not involve any increase in the probability or consequences of an accident previously evaluated.

g. The change to refer to the Unit 1 Overpressure Protection System (OPPS) enable temperature in Specification 3.4.9.3 in lieu of specifying 275 °F was evaluated and found acceptable in the request for approval of Amendment 160. The deletion of the asterisk in Unit 2 Specification 3.9.8.1 was justified as part of the request for approval of Amendment 25. The inadvertent omission of the ACTION to take in the case that the temperature of the steam generator is precisely 50 °F above the cold leg temperature is being corrected. The cases of greater than and less than 50 °F are already included. These are editorial changes that do not involve any increase in the probability or consequences of an accident previously evaluated.

h. The administrative control reporting requirements have been modified to incorporate various ISTS requirements. This requires changing titles and eliminating requirements addressed elsewhere, removing reference to deleted sections, and replacing reference to the administrative control section reporting requirements in various specifications with reference to 10 CFR 50.4. The 1993 NRC final policy statement set forth the criteria for determination of those requirements to be included in TS. The reporting requirements being removed from the TS do not meet the criteria for inclusion in the TS; therefore, the reporting requirements have been modified to reflect those requirements provided in the ISTS. These are editorial changes that do not involve any increase in the probability or consequences of an accident previously evaluated.

i. The Technical Specification index has been modified to address the revised pages.

These changes have been determined to be editorial and administrative in nature, and as such, would not affect any accident assumptions or radiological consequences of an accident. Therefore, the proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The editorial changes, the elimination of reporting requirements which duplicate 10 CFR requirements and administrative improvements to incorporate the ISTS requirements are all changes that are administrative in nature. The proposed changes will not affect any plant system or structure, nor will they affect any system functional or operability requirements. Consequently, no new failure modes are introduced as a result of the proposed changes. Therefore, the proposed change will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The proposed amendment modifies reporting requirements and incorporates associated editorial changes that do not impact the UFSAR design basis or accident analyses assumptions. This change does not introduce any new operational modes or physical modifications to the plant; therefore, no action will occur that will involve a significant reduction in a margin of safety. In addition, the proposed change does not affect radiological release limits, monitoring equipment or operating practices. 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 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: B.F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: Robert A. Capra.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: September 23, 1998.

Description of amendment request: The proposed amendment would change Division III battery specific gravity acceptance criteria outlined in River Bend Station (RBS) Technical Specifications (TS). The change is required as a result of battery system design modifications which are scheduled to be implemented in April 1999 during refueling outage (RF) RF–8. During this time, the current Division III battery will be replaced. The new battery, which also will have a greater capacity rating, will be supplied with a nominal specific gravity of 1.215 at 77°F in contrast to the existing Division III battery supplied with a nominal specific gravity of 1.210 at 77°F. Since TS Section 3.8.6, Table 3.8.6–1 values for specific gravity are based on the manufacturer's nominal specific gravity, these values will need to be updated.

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

The system loads, voltage requirements, and inrush currents have been calculated in accordance with IEEE Std. 485, ''IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations.'' To support these design requirements at a capacity of 80%, a new battery must be installed. The nominal specific gravity of the new battery, as provided by the manufacturer of the battery, is 1.215 at 77°F.

A review of USAR Chapter 15, including Appendix 15A, was conducted to determine what accidents, if any, may be impacted by the proposed change to the Division III battery specific gravity.

USAR Sections 15.2, ''Increase in Reactor Pressure;'' 15.3, ''Decrease in Reactor Coolant System Flow Rate;'' and Section 15.6, ''Decrease in Reactor Coolant Inventory'' discuss accidents that involve the initiation of HPCS when reactor vessel level drops to the initiation point. The function of the HPCS System is to mitigate the consequences of an accident (i.e., to maintain reactor vessel coolant inventory after small breaks which do not depressurize the reactor vessel, or provide spray cooling heat transfer following larger breaks, Ref. USAR Section 6.3.1.2.1). The function of the Division III 125 Vdc Power System is to provide a highly reliable, continuous, and independent source of control and motive power for the HPCS System logic, HPCS diesel generator set control and protection, and all Division III related control (Ref. USAR Section 8.3.2.2.1). This is a support function for the HPCS System.

USAR Section 15.5, ''Increase In Reactor Coolant Inventory,'' postulates an inadvertent HPCS actuation resulting from operator error. The proposed changes to the Division III battery specific gravity cannot result in an inadvertent HPCS actuation/injection. The proposed changes to the allowable specific gravity values provided in Technical Specification 3.8.6 are in agreement with the manufacturer's nominal specific gravity. The revision simply ensures that the battery has sufficient capacity to meet the energy requirements of its critical loads. The proposed change does not create any new internally generated missiles, nor does it affect the High Energy Line Break Analysis or any other accident described in Chapter 15 of the USAR. Neither the function nor the operation of the Division III battery is impacted by the proposed change.

The replacement Division III battery will be supplied by the manufacturer with a nominal specific gravity of 1.215 at 77°F. The battery manufacturer's rated performance is based on the specific gravity of the battery being maintained near the nominal specific gravity. Since the Division III design basis calculation depends on the battery manufacturer's rated performance, battery parameters upon which that performance is based must be monitored. The current Technical Specification values for specific gravity are based upon a nominal specific gravity of 1.210 at 77°F. The proposed values accurately reflect the manufacturer's nominal specific gravity. Testing the Division III battery to the proposed values provides assurance that the HPCS functions supported by the 125 Vdc System will not be adversely affected by the Division III battery.

The proposed changes will not affect failure modes of existing equipment. The proposed changes do not affect the ability of any structures, systems or components to perform their safety functions. Therefore, no undue risk to the health and safety of the public has been created by the proposed changes, nor is there any change in the radiological consequences at the site boundary.

By incorporating the correct value for battery specific gravity verification in Table 3.8.6–1, the Technical Specifications will accurately reflect the new design basis value for the Division III battery specific gravity. This change allows the performance of the Division III battery to be verified against the correct design basis value, thus providing assurance that the Division III 125 Vdc power system function will remain as assumed in the accident analysis. Therefore, the proposed change cannot affect any accidents previously evaluated (probability or consequences). Consequently, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Since a battery's capacity decreases as specific gravity decreases below the manufacturer's nominal value, monitoring the battery's specific gravity is one means of ensuring that the battery will adequately supply the minimum energy required to support the system function assumed in the accident analysis.

All safety systems will continue to function as originally designed. The subject equipment will not function in a manner different than described in USAR Section 8.3.2.2. The functional and performance requirements of the Division III 125 Vdc System and its associated interfaces have not been altered. The proposed change simply ensures that the HPCS battery performance is verified against the correct design basis value. This value provides assurance that the HPCS System functions will not be adversely affected by the capacity of the battery. Therefore, the proposed changes do not create the possibility of occurrence of a new or different kind of accident from any previously evaluated.

3. This request does not involve a significant reduction in a margin of safety.

This proposed change updates the acceptance criteria of the current specific gravity for the Division III battery. This acceptance criteria is in accordance with manufactures recommendations. The design and license basis for the Division III systems and functions remain unchanged and the battery will continue to supply the 125 Vdc loads necessary to support these functions. This value will reflect the manufacturer's nominal specific gravity for the Division III battery. With the system functions supported as assumed in the accident analyses, the margin to safety remains unchanged.

As a result, the proposed change does not involve a significant reduction in a margin to safety.

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

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW, Washington, DC 20005.

NRC Project Director: John N. Hannon.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: October 8, 1998.

Description of amendment request: The proposed amendment would implement Boiling Water Reactor Owners Group (BWROG) Enhanced Option I–A (EIA) Reactor Stability Long Term Solution as documented in NEDO–32339, Revision 1, ''Reactor Stability Long-Term Solution, Enhanced Option I–A.'' The EIA long term solution has been accepted by the NRC in Safety Evaluation Report, ''Reactor Stability Long-Term Solution, Enhanced Option I–A Generic Technical Specifications (TS), NEDO–32339, Supplement 4.''

The proposed changes to the RBS TS will enable the full implementation of the Enhanced Option I–A (EIA) long term solution to the neutronic/thermal hydraulic instability issue. Specifically, the proposed change deletes the limits on power and flow conditions associated with the implementation of the guidance in General Electric Service Information Letter #380, Revision 1, ''BWR Core Thermal Hydraulic Stability'' (current TS 3.4.1, Figure 3.4.1–1 and RBS plant procedures), adds new specifications, to establish limits for Fraction of Core Boiling Boundary (FCBB) and the Period Based Detection System (PBDS), modifies the RPS instrumentation specification and the description of the contents of the Core Operating Limits Report (COLR) in current TS 5.6.5. The two new specifications require maintaining stability control and the availability of a stability detection system during operation in defined regions of the power and flow operating domain.

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

1. The proposed amendments do no involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments allow the implementation of the Enhanced Option I–A (EIA) long term solution to the neutronic/thermal hydraulic instability issue. Current TS restrictions on power and flow conditions, number of operating recirculation loops, and operator actions implemented to reduce the probability of neutronic/thermal hydraulic instability are eliminated and new stability requirements consistent with NEDO–32339–A, Supplement 4, Revision 1, are imposed.

While the proposed amendments permit operation in regions of the power and flow operating domain postulated to be susceptible to neutronic/thermal hydraulic instability, the implementation of the EIA solution ensures there is not a significant increase in the probability or consequences of an accident previously evaluated. Operation in these regions does not increase the probability of occurrence of initiators and precursors of other previously analyzed accidents. The proposed amendments permit the implementation of the features of the EIA solution which prevent neutronic/thermal hydraulic instability. The features include pre-emptive reactor scram upon entry into the regions of the power and flow operating domain most susceptible to neutronic/thermal hydraulic instability—the Exclusion Region. The EIA solution prevents neutronic/thermal hydraulic instability during operation in regions of the power and flow operating domain previously excluded from operation and therefore does not significantly increase the probability of a previously analyzed accident.

The EIA solution also requires implementation of stability control prior to entry into a region of the power and flow operating domain which is potentially susceptible, in the absence of stability control, to neutronic/thermal hydraulic instability. The modified rod block functions providing the restricted region entry alarm (RREA), boiling boundary limits, and PBDS functions are required on entry into the Restricted Region of the power to flow map. The boiling boundary limits, and Period Based Detection System (PBDS) functions are required on entry into the Monitored Region of the power to flow map. The EIA solution prevents or allows for detection and suppression of neutronic/thermal hydraulic instability during operation in these regions of the power and flow operating domain.

The EIA solution includes restrictions on power and flow conditions and actions associated with the modified APRM flow biased scram and RREA functions. Required actions include adherence to the boiling boundary limit stability control prior to entry and during operation in the region of the power and flow operating domain which is potentially susceptible to neutronic/thermal hydraulic instability—in the absence of stability control. In addition, the proposed amendments require operator actions based upon control room indications generated by a new PBDS. The PBDS is designed to provide alarm indication that conditions consistent with a significant degradation in the stability performance of the reactor have occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. The PBDS also provides analog indication of the highest and second highest successive period confirmation count for all of the LPRMs monitored. This provides the plant operators with continuous indication of reactor stability operating conditions. The PBDS system provides indication only and does not affect plant structures, systems, or components in any way that could increase the probability or consequences of an accident. Rather, the improved control room indications provide the operator with more accurate and timely information.

The EIA solution allows for the ''Setup'' of APRM flow biased scram and control rod block function. The EIA solution requires adherence to certain boiling boundary limit stability controls prior to selection by the operator of APRM flow biased scram and control rod block function ''Setup'' setpoints. This ''Setup'' function allows operation in a region of the power and flow operating domain potentially susceptible to neutronic/thermal hydraulic instability provided the additional limits of the flow control boiling boundary (FCBB) and PBDS are met. After exiting the region requiring the stability control to be met, the setpoints can be manually reset to their normal values. Stability controls are required to be in place when setpoints are ''Setup''. As a backup EIA feature, the APRM flow biased setpoints automatically reset to their normal values above a pre-determined flow condition. This automatic reset to the more conservative setpoints ensures that the pre-emptive reactor scram will prevent operation as a result of an anticipated operational occurrence in the region most susceptible to neutronic/thermal hydraulic instability should the operator not select the more conservative setpoints appropriate for operation following exit from the region requiring stability control. The FCBB, PBDS, and automatic reset of the APRM flow biased scram and control rod block function ''setup'' setpoints allow for the use of the ''setup'' feature and help ensure that there is not an increase in the probability or consequences of an accident.

Operation in the regions of the power and flow operating domain excluded by current TS 3.4.1 and Figure 3.4.1–1 can occur as a result of anticipated operational occurrences. In the absence of operator actions the severity of these anticipated operational occurrences may increase due to the potential occurrence of neutronic/thermal hydraulic instability as a result of operation in these regions. Upon entry, as a result of an anticipated operational occurrence, into the region most susceptible to neutronic/thermal hydraulic instability the pre-emptive reactor scram prevents neutronic/thermal hydraulic instability. Therefore, the consequences of an accident do not significantly increase while operating with stability control in place.

The required EIA features is designed to limit possible neutronic/thermal hydraulic instabilities and to detect and suppress further neutronic/thermal hydraulic instabilities. These features include: a pre-emptive automatic scram, the control rod block and alarms associated with entry into the region susceptible to neutronic/thermal hydraulic instabilities, automatic reset of APRM flow biased setpoints, PBDS, FCBB, and the required operator actions, including manual reactor scram. Therefore, the proposed amendments prevent the occurrence of neutronic/thermal hydraulic instability during operation or as a consequence of an anticipated operational occurrence and do not significantly increase the consequences of any previously analyzed accident.

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

The proposed amendments eliminate existing restrictions on power and flow conditions and impose alternative restrictions which permit the implementation of the EIA long term stability solution. The current restrictions on the power and flow conditions do not prevent entry into regions of the power and flow operating domain most susceptible to neutronic/thermal hydraulic instability and therefore the possibility of neutronic/thermal hydraulic instability exists in the absence of operator action. The required features of the EIA solution implement a pre-emptive scram upon entry into the region most susceptible to neutronic/thermal hydraulic instability, without operator action. The accessible operating domain allowed by the proposed amendments is essentially a subset of the power and flow operating domain currently allowed. Initial conditions are bounded by the current initiators and precursors of accidents and anticipated operational occurrences. Accordingly, no new accident of initiator is present. Therefore, the proposed amendments do not create the possibility of a new or different kind of accident from that previously evaluated.

Concurrent with the implementation of the proposed amendments, a modified Flow

Control Trip Reference (FCTR) card, EIA FCTR card, and a new Period Based Detection System (PBDS) will be installed as required by the EIA solution. The function of the EIA FCTR card is to aid the operator in the identification of entry into regions of the power and flow operating domain potentially susceptible to neutronic/thermal hydraulic instability in the absence of stability controls and to initiate a pre-emptive scram upon entry into the regions most susceptible to neutronic/thermal hydraulic instability. This is accomplished by altering the existing values of setpoints of the APRM flow biased scram and the control rod block functions generated by the EIA FCTR card.

The design of the EIA digital FCTR card is a functional equivalent of the original analog FCTR card. The Failure Modes and Effects Analysis (FMEA) for the card detailed in NEDC–32339P-A Supplement 2 found no single failure that would increase the consequences of an accident. The EIA FCTR card maintains the original basis for the NMS interface functions of the analog FCTR card it replaces. The plant specific environmental conditions (temperature, humidity, pressure, seismic, and electromagnetic compatibility) have been confirmed to be enveloped by the environmental qualification values for the EIA FCTR cards. Therefore, the potential for spurious scrams or common mode failures induced by environmental effects (e.g., electromagnetic interference) is considered negligible. The installation of the EIA FCTR card will therefore not create the possibility of a new or different kind of accident from any accident previously evaluated.

The function of the PBDS is to provide the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. This is accomplished by the installation of a new PBDS card in the Neutron Monitoring System in accordance with NRC approved BWROG and GE design. The PBDS card takes inputs from individual local power range monitors and provides analog indication of the highest and second highest successive period confirmation count, provides a Hi DR and Hi-Hi DR alarm, and INOP status indication to the operator in the control room. These displays can not create the possibility of a new or different kind of accident from any accident previously evaluated. The plant specific environmental conditions (temperature, humidity, pressure, seismic, and electromagnetic compatibility) have been confirmed to be enveloped by the PBDS environmental qualification values. Therefore, the installation of the PBDS card will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments permit the implementation of the EIA long term solution to the stability issue. Under certain conditions, existing BWR designs are susceptible to neutronic/thermal hydraulic instability. GDC 10 of 10 CFR 50, Appendix A, requires that specified acceptable fuel design limits not be exceeded during anticipated operational occurrences. General Design Criterion (GDC) 12 of 10 CFR 50, Appendix A, requires thermal hydraulic instability to be prevented by design or be readily and reliably detected and suppressed. When the design of the reactor system does not prevent the occurrence of neutronic/thermal hydraulic instability, instability is considered an anticipated operational occurrence. The proposed amendments and the associated design modifications provide automatic features and operational information to the Control Room that replace the existing BWROG Interim Corrective Actions (ICAs). Thus the EIA solution assures compliance with GDC-10 and GDC 12 by providing for reliable detection and suppression and by the prevention of neutronic/thermal hydraulic instability. This therefore precludes neutronic/thermal hydraulic instability from becoming a credible consequence of an anticipated operational occurrence. As a result the margins of safety are maintained.

Analyses performed by the BWROG indicate that neutronic/thermal hydraulic instability induced power oscillations could result in conditions exceeding the MCPR SL prior to detection and suppression by the current design of the Neutron Monitoring System and Reactor Protection System. To ensure compliance with GDC 12 the BWROG developed Interim Corrective Actions (ICAs) to enhance the capability of the operator to readily and reliably detect and suppress neutronic/thermal hydraulic instability. The BWROG ICAs also provided additional guidance for monitoring local power range monitors beyond the requirements of current TS 3.4.1 to ensure adequate margin to the onset of neutronic/thermal hydraulic instability. Reliance on operator actions to comply with GDC 12 was accepted on an interim basis by the NRC pending final implementation of a long term solution to the stability issue. The modified design of the Reactor Protection System (APRM flow biased scram) and stability control prior to entry into a region of the power and flow operating domain which is potentially susceptible, in the absence of stability control, to neutronic/thermal hydraulic instability implemented with the EIA solution prevents neutronic/thermal hydraulic instability. In addition, significant backup protection features, including the PBDS and specified operator actions, are required to be implemented. As a result, the margin to the onset of neutronic/thermal hydraulic instability provided by the existing TS requirements and BWROG ICAs recommendations is not reduced by the implementation of the EIA solution. The EIA solution assures compliance with GDC 12 by the prevention of neutronic/thermal hydraulic instability and therefore precludes neutronic/thermal hydraulic instability from becoming a credible consequence of an anticipated operational occurrence. The consequences of anticipated operational occurrences will not increase and the margin to the MCPR SL will not decrease upon implementation of the EIA solution. Therefore, the proposed amendment does not involve a reduction in a margin of safety.

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

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW, Washington, DC 20005.

NRC Project Director: John N. Hannon.

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

Date of amendment request: October 1, 1998.

Description of amendment request: The proposed change modifies Technical Specification (TS) 3.3.3.7.3 and Surveillance Requirement 4.3.3.7.3 for the broad range gas detection system. A change to Technical Specification Basis 3/4.3.3.7 has been included to support this change. This change to the TS is necessary for the installation of a new, more reliable broad range gas detection system.

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

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

Response: No.

The broad range gas detection system has no effect on the accidents analyzed in Chapter 15 of the Final Safety Analysis Report. It's only effect is on habitability of the control room, which will be enhanced by installation of the new monitoring system and this change to the Technical Specifications. Qualitative analysis based on a quantitative risk assessment has shown that the impact on operator incapacitation and subsequent core damage risk of the periodic automatic background/reference spectrum check is negligible and that the probability of malfunction of the BRGMs due to a slowly increasing toxic chemical concentration is negligible.

Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of

accident from any accident previously evaluated?

Response: No.

The proposed Technical Specification change in itself does not change the design or configuration of the plant. The new broad range toxic gas monitoring system performs the same function as the old system, but it accomplishes this function with increased reliability.

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

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

Response: No.

The broad range gas detection system has no effect on a margin of safety as defined by Section 2 of the Technical Specifications. Its only effect is on habitability of the control room, which will be enhanced by installation of the new monitoring system and this change to the Technical Specifications.

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

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn, 1400 L Street NW, Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

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: July 30, 1998 (LAR–222).

Description of amendment request: The proposed amendment will change the Improved Technical Specifications (ITS) to add a new Required Action for the existence of breaches in the Control Complex Habitability Envelope (CCHE) that are in excess of allowances. A new surveillance requirement for the performance of a periodic integrated leak test of the CCHE boundary on a 24-month frequency would also be added. Changes to the current Ventilation Filter Test Program (VFTP) are proposed to adopt current standards for laboratory testing, change acceptable values of control room emergency ventilation flow rate and filter differential pressure, and add the Auxiliary Building Ventilation Exhaust Filters to the VFTP. Conforming changes to the ITS Bases are also included.

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.

Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The Control Room Emergency Ventilation System (CREVS) and the Control Complex Habitability Envelope (CCHE) are designed to limit the radiation dose to the control room operating staff following a design basis accident. Since these systems are only effective in limiting dose following an accident, the existence of limited breaches in the CCHE, the performance of periodic leak tests, and changes to the Ventilation Filter Test Program (VFTP) would not increase the probability of occurrence of any evaluated event. The features of the CREVS and the Control Complex emergency filters, or the CCHE have no direct function in mitigating the offsite consequences of any evaluated accident. The Auxiliary Building exhaust filters are not credited with reducing offsite doses, however, if available would filter releases from the Auxiliary Building. Adding them to the VFTP will not increase the consequences calculated for any evaluated accident.

The proposed changes are consistent with the revised control room operator dose calculations as presented in the Control Room Habitability Report dated July 1998. Since all calculated doses are within 10 CFR Part 50, Appendix A GDC 19 limits there is no significant increase in consequences.

It is conceivable that the existence of additional breaches in the CCHE could result in an increase in operator dose, however the low probability of a catastrophic reactor accident, the relatively short time allowed for breaches to be open in excess of approved dose calculation assumptions, and the ability to close breaches expeditiously makes the risk increase insignificant.

The changes to the ITS Bases improve information on the operation and function of CREVS, and establish that CREVS operability is dependent on maintaining CCHE integrity. The inclusion of this information reinforces the importance of maintaining the CCHE boundary, and will help to ensure the CREVS is capable of performing its intended safety function.

The Control Room Habitability Report, dated July 1998, provided with this LAR presents the methodology used and the results of the operator dose calculations for the Maximum Hypothetical Accident, toxic gas release, and other design basis accidents. The report provides the information needed for NRC review of LAR 222, Revision I and the associated unreviewed safety question. This evaluation concludes that the current level of CCHE integrity provides adequate protection for the control room operator.

Based on the foregoing, the proposed amendment does not significantly increase the probability or consequence of an accident previously evaluated.

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

Neither performance of periodic CCHE leak tests nor changes to the existing VFTP can create the possibility of a new or different kind of accident. During the period of time when CCHE breaches are greater than the design calculation, there exists the possibility that control room dose from an analyzed accident may be greater than specified in General Design Criterion 19. This condition will not however create the possibility of a new or different kind of accident. Since CREVS and the emergency filtration units function to provide protection following a radiological accident the changes proposed to improve their performance cannot create a new or different kind of accident. Changes to the Bases to provide better information on determining CREVS and CCHE operability cannot create the possibility of a new or different kind of accident.

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

The proposed amendment does not involve a significant reduction in a margin of safety. Neither performance of periodic CCHE leak tests nor changes to the existing VFTP can create a reduction in the margin of safety. The changes to both of these programs will result in improved assurance that the CREVS and CCHE will perform as expected if required for operator protection. Changes to the Bases of the CREVS Technical Specification which clarify the conditions necessary for operability will improve understanding of the requirements for maintaining control room habitability, and will not create a reduction in the margin of safety. The existence of additional breaches in the CCHE for short periods of time does not significantly increase the risk of control room operator exposure to airborne radioactivity or toxic gas. There is no change in the risk to the public since the CCHE has no direct function in mitigating the offsite consequences of any evaluated accident. Any event that could create these exposures has an extremely low probability of occurrence, and while the potential for higher operator exposure exists if additional breaches are open, the short duration allowed would not significantly increase the risk of exposure. Therefore, for the reason stated above the existing margin of safety would not be reduced.

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

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

Local Public Document Room location: Coastal Region Library, 8619

W. Crystal Street, Crystal River, Florida 34428.

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: September 30, 1998 (LAR–238).

Description of amendment request: The proposed amendment will correct the reactor coolant system (RCS) leakage detection capability of the Reactor Building atmosphere gaseous radioactivity monitor described in the Improved Technical Specification Bases and the Final Safety Analysis Report (FSAR). These documents currently identify that the gaseous radioactivity monitor is capable of detecting a one gallon per minute (gpm) RCS leak within one hour. The licensee has determined that it would take approximately 14 hours for this instrument to detect a one gpm RCS leak using currently accepted assumptions. The capability of other monitors to detect a one gpm RCS leak within one hour is not affected by this change.

The licensee cited several factors which contribute to the difficulty in reliably detecting RCS leakage increases of one gpm within one hour using a gaseous radioactivity monitor. These include the relatively long half-life of Xe-133 (primary nuclide of detection), fluctuations in background levels of radioactivity, the existence of minor RCS leaks, improved performance of nuclear fuel, and improved primary water chemistry control. Based on RCS radioactivity concentrations assumed in the Environmental Report, half-lives of the most abundant gaseous nuclides, and background radioactivity levels, the licensee indicated a one gpm leak can conservatively be detected in approximately 14 hours by the gaseous monitor. The licensee has determined that this change to the licensing basis is an unreviewed safety question.

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 function of the RM–A6 gaseous radioactivity monitor is to detect leakage from the RCS that may develop as a result of a flaw in a pressure boundary component. The previously identified capability to detect a one gpm leak within one hour would have provided an earlier warning of a small RCS leak than the actual detection capability now identified. However, RCS loss of coolant accidents evaluated in the FSAR cover the full spectrum of break sizes up to and including a complete severance of the largest RCS piping. The results of these analyses demonstrate that the consequences of such leaks are acceptable.

No other equipment relies on the capability of the RM–A6 gaseous monitor's ability to detect RCS leakage to perform its function. Likewise, no accident analyses rely on RCS leak detection for successful mitigation. Identifying the detector's actual capability to detect an RCS leak will not increase the probability of occurrence of an RCS leak. Detection time for an RCS leak was a consideration in granting a partial exemption to General Design Criterion 4. However, the capability of the RCS piping to resist propagation of a flaw from a leak into a break was based on material fracture analysis and material properties, not on the ability to detect low levels of leakage.

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

No. The function of the RM–A6 gaseous radioactivity monitor is to detect RCS leakage that may develop from a flaw in a pressure boundary component. The monitor is a passive component that provides an indication of possible leakage for further operator evaluation. Identifying that a longer response time is required for the monitor to detect a small leak will not create the possibility of a new or different kind of accident. Existing analyses for small and large break loss of coolant accidents provide an evaluation of the full spectrum of RCS break sizes.

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

No. The RM–A6 gaseous radioactivity monitor is included in plant technical specifications as one of two containment atmosphere RCS leak detection instruments required to be operable to satisfy a limiting condition for operation. If the RM–A6 particulate monitor is not operable, then the response time of the containment atmosphere monitor will be increased. RCS piping analyses have demonstrated that the propagation of a small primary loop leak into a pipe break would not occur rapidly. NRC acceptance of the applicable analyses included significant safety factors for the propagation of flaws into pipe breaks which were based on low probability stress combinations of normal plus safe shutdown earthquake loads. Considering the actual detection capability of the RM–A6 gaseous monitor and the existence of other diverse leak detection capabilities, detection of a leak in a relatively short period of time is anticipated. In the event an RCS leak developed into a pipe break, current accident analyses would bound the effects of the pipe break on and off site. Therefore, the possibility of increased time to detect an RCS leak does not represent 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: 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: October 16, 1998 (LAR–229).

Description of amendment request: The proposed amendment would change the Crystal River Unit 3 (CR–3) Final Safety Analysis Report (FSAR), Improved Technical Specifications (ITS) and ITS Bases to resolve an Unreviewed Safety Question (USQ). This USQ was created by changing the normal standby position of valves DHV–34 and DHV–35 (low pressure injection (LPI) pump suction valves from borated water storage tank) from normally open to normally closed. Maintaining these valves normally closed is necessary to ensure assumptions used in fire protection analyses remain valid. The proposed amendment would also add new ITS surveillance requirements for verifying on a periodic basis that the LPI system components and piping, and the building spray suction piping, are full of water.

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?

Valves DHV–34 and DHV–35 are located in the suction lines between the borated water storage tank (BWST) and the low pressure injection (LPI) and building spray (BS) pumps. These valves are maintained normally closed, and are designed to automatically open upon receipt of a reactor coolant system (RCS) low-low pressure signal of 500 psig or a reactor building (RB) high pressure signal of 4 psig from the engineered safeguards actuation system (ESAS). The designed full stroke time of these valves is within the assumptions of the accident analyses performed for the specific design basis accidents that require the LPI and/or BS systems for accident mitigation. This is the original design basis for these valves. Therefore, the valves are fully capable of performing their intended safety functions while being maintained normally closed.

The failure of one of these valves to open does not impact the mitigation of previously analyzed accidents that require the operation of the LPI and/or BS systems, and cannot increase the probability of these accidents occurring. No RCS or secondary system pressure boundaries are compromised, no release paths for radioactive materials are created, and no challenge to any safety limit or acceptance limit are created by maintaining these valves normally closed.

A single, active failure causing one of these valves to fail to open upon demand would render one train of LPI and BS unavailable for accident mitigation. However, the accident analyses have already accounted for the possibility of only one train of LPI and BS being available, and the consequences of previously evaluated accidents would therefore remain unchanged.

Undetected voiding in the LPI piping and components, and BS suction piping, is highly unlikely to occur. Based on the design and physical layout of the LPI system and BS system, and the monitoring of the systems performed on a periodic basis, any potential for LPI piping and components and BS suction piping voiding will be quickly and easily recognized and corrected. Therefore, since voiding is not likely to occur, the consequence of previously evaluated accidents would not be significantly increased by the proposed change.

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

Failure of either valves DHV–34 or DHV–35 to open upon demand on an ESAS signal will not create the possibility of a new or different kind of accident. The LPI system and BS system are maintained in a standby condition during normal plant operations, and automatically actuate only after an accident has occurred to mitigate the effects of the initiating accident. No RCS or secondary system pressure boundaries are compromised, no release paths for radioactive materials are created, and no challenges to any safety limit or acceptance limit are created by maintaining these valves normally closed. Additionally, the possibility of undetected voiding in the LPI piping and components, and BS suction piping, is not likely to occur by maintaining these valves normally closed. Therefore, maintaining valves DHV–34 and DHV–35 normally closed will not be an initiator of a new or different kind of accident from previously evaluated accidents.

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

Maintaining valves DHV–34 and DHV–35 normally closed will not create a reduction in the margin of safety. Maintaining valves DHV–34 and DHV–35 normally closed will ensure the capability to safely shut down the reactor under certain postulated fire scenarios, but will result in an extremely small increase in the probability of failure of one train of LPI and BS to perform its safety functions. Based on use of the CR–3 Probabilistic Safety Analysis (PSA) model, and assuming the failure of either valve DHV–34 or DHV–35 to open, the impact on the core-damage frequency was estimated and determined to slightly increase from 7.38 E–6 to 7.41 E–6 per year. This increase (3 E–8 or 0.4%) is in the range considered acceptable in Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis,'' dated July 1998.

Maintaining these valves normally closed will not result in undetected voiding in the LPI piping and components, and BS suction piping, as a result of performance of periodic pressure monitoring. If voiding occurs, the Improved Technical Specifications specify the actions required to restore the affected systems to operable status, including correcting the external leakage creating the observed pressure decay. Therefore, the proposed monitoring will ensure the margin of safety is not reduced.

Based on these benefits and risks, there is no discernible change in the risk to the public in mitigating the offsite consequences of any evaluated accident since the failure of one train of LPI and/or BS for any reason is bounded by the assumptions of the accident analyses. Failure of valve DHV–34 or DHV–35 to open upon demand results in extremely low increases in the potential for reactor core damage. Therefore, the existing margin of safety will not be reduced.

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

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

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

NRC Project Director: Frederick J. Hebdon.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: October 15, 1998.

Description of amendment request: The proposed amendment request would revise the TMI–1 Updated Final Safety Analysis Report (UFSAR) Chapter 14 postulated accident analysis radiological dose consequences resulting from application of revised atmospheric dispersion factors (X/Q) at the Technical Specification Section 5.1.1 defined exclusion area boundry (EAB) and low population zone (LPZ).

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. The proposed amendment has no effect on structures, systems or components. More extensive and recent meteorological data have been utilized for atmospheric dispersion factor (X/Q) determination for both EAB and LPZ. An evaluation of the design basis accidents with revised EAB and LPZ X/Q values results in increases in UFSAR Chapter 14 EAB and LPZ dose consequences which remain well within the guidelines of 10 CFR Part 100.

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 amendment has no impact on any plant structures, systems or components. The proposed change revises the atmospheric dispersion factors for EAB and LPZ used in the existing UFSAR Chapter 14 accident analyses, based on more extensive meteorological data. These changes only effect the postulated dose consequences of currently analyzed accidents. 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 proposed amendment has no impact on structures, systems or components. The proposed revisions to the EAB and LPZ X/Q values are based on recent more extensive meteorological data and Regulatory Guide 1. 145 methods. The increased X/Q values provide a more accurate assessment of meteorological conditions which result in postulated dose consequences which remain well within the guidelines of 10 CFR Part 100. 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, Pitman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cecil O. Thomas.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: October 19, 1998.

Description of amendment request: The proposed Technical Specification change request would add operability and surveillance requirements for the remote shutdown system similar to those in NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants'' Section 3.3.18 entitled ''Remote Shutdown System''.

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

1. Operation of 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. The proposed amendment adds operability and surveillance requirements for the existing TMI–1 remote shutdown system similar to those contained in NRC NUREG–1430, ''Standard Technical Specifications—Babcock & Wilcox Plants''. The addition of these requirements to Technical Specifications provides further assurance of remote shutdown system operability in the event that operators must place and maintain the unit in a safe shutdown condition from outside the control room. The function and operation of the remote shutdown system has not changed. Therefore, this activity has no affect on the probability of occurrence or 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 amendment has no impact on any plant structures, systems or components. The function and operation of the remote shutdown system has not changed. 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 proposed amendment provides additional assurance of remote shutdown system operability. The function and operation of the remote shutdown system has not changed. 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.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: October 19, 1998.

Description of amendment request: The proposed change to the TMI–1 Technical Specification would revise the limit on reactor coolant system activity to a maximum allowable of 1.0 microcurie/gram dose equivalent I–131. The proposed revision provides an allowable reactor coolant system specific activity limit base on once-through steam generator (OTSG) inspection results performed each refueling outage.

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. The proposed amendment has no effect on structures, systems or components. The existing steam line break criteria are maintained. This change only accounts for radiological consequences resulting from a revised maximum allowable reactor coolant system (RCS) specific activity limit of 1.0 µCi/gm. The new radiological consequences of the revised MSLB accident, which also incorporate more conservative values for atmospheric dispersion, are below 10 CFR 100 limits and 10 CFR 50, Appendix A, GDC–19 limits for the control room. The use of revised atmospheric dispersion factors for other TMI–1 accident analysis is addressed in a separate license amendment request submittal. 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 amendment has no impact on any plant structures, systems or components. OTSG tube structural integrity is maintained. 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 proposed amendment has no impact on structures, systems or components. OTSG tube structural integrity is maintained. The existing TMI–1 Technical Specification Section 3.1.4.1 Bases state that the limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will be well within the Part 100 limit following associated design basis accidents postulated in conjunction with an assumed steady state primary-to-secondary steam generator tube leakage of 1.0 gpm. This margin of safety is preserved since resulting does consequences incorporating more conservative values for atmospheric dispersion remain well within the Part 100 limit. 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, Pitman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: October 23, 1998.

Description of amendment request: The proposed amendment would allow implementation of a feedwater leakage control system to address leakage through the primary containment feedwater penetration isolation valves.

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 implements a method of providing a qualified sealing system for the primary containment feedwater penetration isolation valves. This water sealing function, i.e., the FWLCS, constitutes a new operating mode of the Residual Heat Removal (RHR) system. The FWLCS introduces new piping that constitutes an extension of the reactor coolant system (RCS); however, such piping is designed to the same requirements as other RCS piping and as such introduces no significant increase in the probability of any accident previously evaluated. Notwithstanding, a postulated line break in any of the new FWLCS piping would not, by itself, introduce any new effects or consequences not already bounded by postulated line-break or LOCA events previously evaluated in the USAR. Since the proposed change does not affect any parameters or conditions that contribute to the initiation of any accidents previously evaluated, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak-tight integrity of the primary containment designed to mitigate the consequences of a loss-of-coolant accident (LOCA). Once the FWLCS mode has been initiated and a water seal for the seating surfaces of the primary containment feedwater penetration isolation valves has been established (within one hour after the accident), post-LOCA primary containment atmosphere will be prohibited from leaking through the feedwater penetrations and thus bypassing the secondary containment.

Calculations of post-accident DBA LOCA doses affected by this change use accepted ICRP 30 dose conversion factors and take credit for suppression pool scrubbing. Suppression pool scrubbing is effective in reducing iodine release but has no assumed effect on the removal of noble gases. Since the methodology and assumptions for scrubbing are acceptable to the NRC per the guidance in SRP Section 6.5.5 and the values for decontamination factors are conservative, considerable margin is preserved within the analysis. However, these calculations show increases in some of the previously evaluated post-accident doses when compared with dose calculations performed as part of the initial plant licensing basis. Although some of the newly calculated post-accident doses are larger than those that were previously approved, the increases remain small enough to be within the acceptance limits given in 10 CFR 50, Appendix A, GDC 19 and in 10 CFR 100.11.

Since all of the newly calculated post-accident doses resulting from the proposed addition of a water sealing system for the feedwater primary containment penetration isolation valves are below the 10 CFR 50, Appendix A, GDC 19 and 10 CFR 100.11 acceptance limits, IP has concluded that the proposed change does not result in a significant increase in the consequences of an accident previously evaluated.

2. The proposed change institutes a new operating mode of the RHR system (the FWLCS mode). When this mode is established, it will reduce primary containment atmosphere leakage to the environment in the event of a LOCA. Flow diverted from the RHR system to the FWLCS has been evaluated, and has been determined to have no adverse impact on the capability of the RHR system to perform its intended safety functions. Further, the additional piping added for the FWLCS is designed to appropriate requirements for the RCS, thus ensuring that RCS integrity is maintained per design. Sufficient isolation between the RCS and the RHR low-pressure piping will also be maintained per the FWLCS design. Thus, no safety functions are altered or impacted as a result of this change. Installing, operating, or testing the components that support the FWLCS mode has no influence on, nor does it contribute to the possibility of a new or different kind of accident or malfunction from those previously analyzed. Because the USAR analysis already assumes leakage through the feedwater primary containment penetrations following a design basis LOCA, and the subject change does not affect the type of accident(s) that are postulated to occur, the proposed change does not present the possibility of an accident of a different type. Additionally, the change in dose analysis methodology does not create an accident or malfunction of a different type since it only involves the analysis of the effects of accidents or malfunctions previously evaluated in the USAR.

Based on the above, IP has concluded that the proposed change will not create the possibility of a new or different kind of accident not previously evaluated.

3. The margin of safety impacted by the proposed change involves the dose consequences of postulated accidents which are directly related to the primary containment leakage rate, specifically those consequences associated with dose attributable to leakage through the feedwater lines which are secondary containment bypass leakage paths.

Although considerable conservatisms were included in the reanalysis, this reanalysis identified some dose values that increased above the previously licensed values as well as some dose values that decreased below the previously licensed values. However, all of the radiation dose consequences resulting from the proposed change will continue to be below the 10 CFR 50, Appendix A, GDC 19 and 10 CFR 100.11 acceptance criteria.

Except for providing a method of sealing the feedwater primary containment penetration isolation valves (and therefore the method of performing periodic leakage testing of these components) no other change in the method of primary containment leakage testing or secondary containment bypass leakage path testing is being proposed. All other primary and secondary containment bypass leakage testing will continue to be performed in accordance with existing Technical Specification requirements. Adequate programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment, systems and components penetrating the primary containment, and for all secondary containment bypass leakage paths.

As a result, IP has concluded that the proposed change will not result in a significant reduction in a margin of safety.

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

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

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, IL 62525.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: September 28, 1998.

Description of amendment request: The proposed amendment request would resolve an unreviewed safety question (USQ) and amend the operating license to allow manual override capability for the containment isolation actuation signal to reactor coolant system letdown isolation valves.

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 modification does not change the probability of any accident previously evaluated since it does not change any mode of normal operation. Neither the accident signal (CIAS) nor the override feature is an initiator of an analyzed event. The consequences of an accident are also not changed significantly due to the fact that design and administrative controls ensure that previous accident analyses are bounding. The associated isolation valves will operate as they have in the past in response to an accident signal. There is no single failure that would prevent the letdown isolation function from occurring. The CIAS override feature can only be used if operators have verified that an UHE is the event which has taken place and safety functions are being met.

This ensures that no significant fuel failures will occur due to the event and the consequences of overriding CIAS will not adversely impact radiological conditions in the auxiliary building.

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

The proposed modification does not create any failure mode which could impact the operation of the RCS or associated systems in a manner that would create a new or different kind of accident. With respect to the letdown isolation function, the plant will operate as it previously has and will respond the same way, automatically, to an accident signal. No new accidents have been identified.

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

The procedural restrictions associated with the use of the CIAS override feature will ensure that existing analyses addressing the consequences of an UHE will be bounding and that safety functions will be maintained as defined in EOPs. The radiological consequences of letdown restoration in the auxiliary building will be similar to normal operating conditions and will be bounded by that assumed in the EEQ analysis. RCS inventory and pressure control will be maintained within the established procedural limits.

Letdown restoration capability already exists after ESF reset. The modification permits letdown restoration to occur earlier than it would previously have been possible.

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: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, NW, Washington, DC 20005–3502.

NRC Project Director: William H. Bateman.

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

Date of amendment request: October 15, 1998.

Description of amendment request: The proposed Technical Specification (TS) changes involve revising TS Section 3/4.10 to include a new Special Test Exception allowing the reactor to be considered in operational condition (OPCON) 4 (cold shutdown) during inservice leak or hydrostatic testing with a reactor coolant water temperature greater than 200°F and less than or equal to 212°F. This is an exception to certain OPCON 3 (hot shutdown) requirements, including primary containment. The proposed TS changes will permit unrestricted access to the primary containment for the performance of required inspections.

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 Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes do not make any physical alterations or modifications to plant systems or equipment. The proposed TS changes will permit the performance of inservice leak or hydrostatic testing, with the reactor in OPERATIONAL CONDITION (OPCON) 4 (COLD SHUTDOWN) and the average reactor coolant temperature greater than 200°F and less than or equal to 212°F. The probability of a leak in the reactor coolant pressure boundary during inservice leak or hydrostatic testing is not increased by considering the reactor in OPCON 4 with reactor coolant temperatures greater than 200°F and less than or equal to 212°F. The inservice leak and hydrostatic testing is performed water solid or near water solid. The stored energy in the reactor core will be very low and the potential for failed fuel and a subsequent increase in reactor coolant activity above TS limits is minimal. In addition, Secondary Containment will be operable and capable of handling airborne radioactivity from leaks that could occur during the performance of inservice leak or hydrostatic testing. Requiring the Secondary Containment to be operable will ensure that potential airborne radioactivity from leaks will be filtered through the Standby Gas Treatment System (SGTS), thereby limiting any radioactivity releases to the environment.

In the event of a large primary system leak, the reactor vessel would rapidly depressurize allowing the low pressure Emergency Core Cooling System (ECCS) subsystems to operate. The capability of the systems that are required for OPCON 4 would be adequate to keep the core flooded under this condition. Small system leaks would be detected by leakage inspections before significant inventory loss has occurred. This is an integral part of the hydrostatic testing program.

Therefore, the proposed TS changes will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed TS changes do not make any physical alterations or modifications to plant systems or equipment. The proposed TS changes do not adversely impact the operation of any plant equipment. Allowing the reactor to be considered in OPCON 4 during hydrostatic or inservice leak testing, with a reactor coolant temperature greater than 200°F and less than or equal to 212°F, is an exception to certain OPCON 3 (HOT SHUTDOWN) requirements, including primary containment integrity. The hydrostatic or inservice testing is performed water solid, or near water solid. The stored energy in the reactor core will be very low and the potential for failed fuel and a subsequent increase in coolant activity above TS limits is minimal. In addition, the Secondary Containment will be operable and capable of handling airborne radioactivity from leaks that could occur during the performance of hydrostatic or inservice leakage testing.

The inservice leak or hydrostatic test conditions remain unchanged. The potential for a system leak remains unchanged since the reactor coolant system is designed for temperatures exceeding 500°F with similar pressures. There are no alterations of any plant systems or components that cope with the spectrum of accidents.

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

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

The proposed TS changes do not make any physical alterations or modifications to plant systems or equipment. The proposed changes will permit the performance of inservice leak and hydrostatic testing with a reactor coolant temperature greater than 200°F and less than or equal to 212°F and the reactor in OPCON 4. Since the reactor vessel head will be in place, Secondary Containment integrity will be maintained, and all systems required in OPCON 4 will be operable in accordance with the applicable TS requirements. The proposed TS changes will not have any significant impact on any design basis accident or safety limit. The hydrostatic or inservice leak testing is performed water solid, or near water solid. The stored energy in the reactor core is very low and the potential for failed fuel and a subsequent increase in coolant activity would be minimal. In the event of a large primary system leak, the reactor pressure vessel would rapidly depressurize and the low pressure ECCS subsystems would function as designed to maintain adequate reactor core coverage. This would ensure that the fuel would not exceed peak clad temperature limits.

Also, requiring Secondary Containment integrity will assure that potential airborne radioactive material can be filtered through the SGTS. This will assure that any offsite doses remain well within the limits of 10 CFR 100 guidelines. Small system leaks would be detected by inspections before significant inventory loss could occur.

Therefore, this proposed TS change will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: October 19, 1998.

Description of amendment request: The proposed amendment would eliminate restrictions imposed by Technical Specification (TS) 3.0.4 for the Filtration, Recirculation and Ventilation System (FRVS) during fuel movement and core alteration activities. Specifically, TS Limiting Conditions for Operation (LCOs) 3.6.5.3.1 and 3.6.5.3.2 would each be revised to add a note stating that the provisions of TS 3.0.4 are not applicable for initiation of handling of irradiated fuel in the secondary containment and core alterations provided that the plant is in Operational Condition 5, with reactor water level equal to or greater than 22 feet 2 inches.

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

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

The proposed TS change does not involve any physical changes to plant structures, systems or components (SSC). FRVS will continue to function as designed. FRVS is an Engineered Safety Feature (ESF) designed to mitigate the consequences of an accident, and therefore, can not contribute to the initiation of any accident. For refueling accidents, the current design basis analysis of FRVS credits only the iodine removal capability of the FRVS ventilation unit and neglects the considerable iodine removal capability of the FRVS recirculation units. In addition, this proposed TS change will not increase the probability of occurrence of a malfunction of any plant equipment important to safety, since the time limits imposed by the current FRVS LCO Action Statements are not affected by these proposed changes. The proposed changes merely allow entry into the FRVS LCO Action Statement in order to support refueling activities.

Therefore, the proposed TS changes, which would permit the initiation of core alterations and handling of irradiated fuel with only one operable FRVS ventilation unit and four operable FRVS recirculation units for a limited seven day period under specific refueling conditions, would not result in the increase of the consequences of an accident previously evaluated.

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

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

The proposed TS changes do not involve any physical changes to plant SSC. The design and operation of the FRVS is not changed from that currently described in the [Updated Final Safety Analysis Report] UFSAR. FRVS will continue to function as designed to mitigate the consequences of an accident. No changes of any kind are being made to FRVS, or its support or supported systems. Deleting the restrictions imposed by TS 3.0.4 as proposed in this TS change request eliminates a compliance restriction imposed by the current TS. Since the current TS already provide a seven day period to perform refueling activities with inoperable FRVS ventilation and recirculation units, the proposed changes would not introduce plant operation in a configuration that is not already permitted in the TS. Therefore, there is no possibility that implementing this proposed TS change would create a different type of malfunction to the FRVS than any previously evaluated. In addition, the proposed TS changes do not alter the conclusions described in the UFSAR regarding operation of FRVS.

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

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

The proposed TS change involves the elimination of TS 3.0.4 restrictions imposed on the FRVS LCO. The TS 3.0.4 requirements impose an unnecessary challenge to performing refueling activities when the FRVS LCO Action Statements already sufficiently define the remedial measures to be taken. The time limits imposed by the current FRVS LCO Action Statements are not affected by these proposed changes. The FRVS LCO will retain sufficient configuration controls to appropriately maintain the capability of FRVS to mitigate design basis refueling accidents, no new FRVS configurations will be permitted by the proposed changes, and there will be no reduction in any margin of safety resulting from this proposed TS change. Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: September 18, 1998.

Description of amendment request: The proposed amendment would revise the VCSNS Technical Specifications (TS) to address the Best Estimate Analyzer for Core Operations—Nuclear (BEACON) core power distribution monitoring and support system. The BEACON system provides continuous core monitoring capabilities to augment the flux mapping system when rated thermal power (RTP) is greater than 25%. The proposed amendment would also make editorial changes to TS 3.3.3.2 and 4.3.3.2.c to delete the reference to F_{xy}.

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

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

The proposed change allows the Power Distribution Monitoring System (PDMS) to be used for measuring power distribution limits when Thermal Power is greater than 25% RTP. This includes relocating manufacturing and measurement uncertainty values from the Technical Specification to the COLR [core operating limit report]. Also included in this change is the addition of a new specification and bases section for the Power Distribution Monitoring System (PDMS). The Technical Specification Power Distribution Limits are not being changed; only the method in which they are measured is being changed. The probability of an accident is not significantly increased. The measurement of power distribution limits and the location of manufacturing and measurement uncertainty values are not initiators of any analyzed event. The change will not affect the consequences of any analyzed event. The power distribution limits will still be measured and verified to be within limits as required by the current Technical Specification Surveillance. The cycle-specific core operating limits, although not in Technical Specifications, will be followed in the operation of VCSNS. The actions as required by current Technical Specifications, when or if limits are exceeded are not being changed. This change will not significantly affect the assumptions relative to the mitigation of accidents.

Each accident analysis addressed in the VCSNS Final Safety Analysis Report will be examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC-approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be performed per requirements of 10 CFR 50.59, ensures that future reloads will not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

The proposed change allows the Power Distribution Monitoring System (PDMS) to be used for measuring power distribution limits when Thermal Power is greater than 25% RTP. This includes relocating manufacturing and measurement uncertainty values from the Technical Specification to the COLR. Also included is the addition of a new specification and bases section for the Power Distribution Monitoring System. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. No hardware is being added to the plant as part of the change. The cycle specific variables are calculated using the NRC-approved methods and submitted to the NRC to allow the Staff to continue to trend the values of these limits. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded. The change will not introduce any new accident initiators. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change allows the Power Distribution Monitoring System (PDMS) to be used for measuring power distribution limits when Thermal Power is greater than 25% RTP. The margin of safety presently provided by current Technical Specifications remains unchanged. Only the method in which the power distribution measurements are obtained is being changed. This method is verified by Westinghouse, and reviewed and approved by the NRC. Appropriate measures exist to control the values of the manufacturing and measurement uncertainties. The proposed amendment continues to require operation within the core limits, as obtained from NRC-approved reload design methodologies. Appropriate actions required to be taken when or if limits are violated remain unchanged.

Future changes to measurement and manufacturing uncertainties located in the current Technical Specification will be evaluated per the requirements of 10 CFR 50.59. Since the 10 CFR 50.59 process does not allow any reduction in the margin of safety, prior NRC approval is required prior to a reduction in the margin of safety. If the evaluation of the changes [does] not result in [an] unreviewed safety question, prior NRC approval will not be required. Additionally, the VCSNS Technical Specifications require that all revisions of the plant COLR be submitted to the NRC upon issuance.

Therefore, the 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: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: October 12, 1998.

Description of amendment request: The proposed amendments would revise Section 6, ''Administrative Controls,'' of the current Units 1 and 2 Technical Specifications (TS) to recognize additional management positions associated with the steam generator replacement project and providing them the ability to approve procedures regarding this project.

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

(1) The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The proposed changes have no impact on the probability of an accident. The change being proposed is administrative in nature and involves no physical alteration of the plant or changes to setpoints or operating parameters. The change will provide an appropriate level of review and approval of procedures related to the FNP steam generator replacement without impacting the operational attention of the current on-site plant management. There is no change in the FNP design basis as a result of this change and, as a result, does not involve a significant increase in the consequences of an accident previously evaluated.

(2) The proposed changes to the TS do not increase the possibility of a new or different kind of accident than any already evaluated in the FSAR. No new limiting single failure or accident scenario has been created or identified due to the proposed changes. Safety-related systems will continue to perform as designed. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) The proposed changes do not involve a significant reduction in the margin of safety. Adding individuals with the appropriate knowledge base to the list of individuals who can approve procedures, which may affect plant nuclear safety, is administrative in nature. There is no impact on the accident analyses. The training and experience requirements for the newly designated management positions are similar to those requirements for other FNP management positions. Therefore the established level of procedure review and approval is not adversely impacted. In addition, these changes allow FNP management to remain focused on plant operations. Thus the proposed 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 amendment request involves no significant hazards consideration.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama.

NRC Project Director: Herbert N. Berkow.

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: September 28, 1998.

Description of amendment request: The proposed amendment would modify the requirements applicable when one or more trains of fuel handling building exhaust air or control room makeup and cleanup filtration are inoperable, and eliminate the need to enter Technical Specification 3.0.3 when multiple trains of these systems are inoperable. In addition, the proposed changes would align the actuating instrumentation and logic system required actions with those that are applicable to the systems. Finally, an administrative change is proposed to remove a footnote that is no longer applicable to the facility.

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

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

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

(a) Assuring that the Specifications define consistent allowed outage times when the same safety function is addressed in multiple Specifications,

(b) Allowing a system to remain inoperable when appropriately restrictive administrative controls are placed on operations that could result in a challenge to the safety function of the system,

(c) Providing an appropriately short Allowed Outage Time for inoperability needed to permit required maintenance and testing that affects all trains of a system,

(d) Redefining system operability and associated actions in a manner consistent with the system design and function,

(e) Aligning a system to the actuated condition on the loss of an actuation channel,

(f) Using consistent terminology throughout the Specifications.

The proposed changes do not represent significant increases in the probability or consequences of an accident because:

(a) The alignment of the action times between actuating system and actuated system operability requirements do not affect the probability or consequences since inoperability of the actuated system has the same effect as inoperability of the actuating system. Since the changes proposed to the actuating system action times will reflect those of the actuated system action times, no change to the allowed outage time applicable to the safety function addressed and fulfilled by both, will occur.

(b) Administrative controls to prevent the conduct of operations that could lead to a challenge to the safety function of the system when the actuation system is inoperable, assures that the design bases functions of the system will not be challenged. Therefore, the probability or consequences of an event previously identified have not been significantly changed.

(c) Allowing up to 12 hours to recover from the inoperability of all three trains of Control Room Ventilation or two or more trains of Fuel Handling Building HVAC does not represent a significant change to the probability of an accident because the inoperability of these ventilation systems are not identified as precursors to a design basis event. The low likelihood of a design basis accident during the limited period of allowed inoperability of these systems does not represent a significant increase in the consequences of an accident.

(d) The redefinition of plant operability requirements into functional trains rather than individual components does not affect the required system functional operability. Therefore, this change does not represent an increase in the probability or consequences of an accident previously identified.

(e) The alignment of the Control Room Ventilation System to the same configuration it would be placed in from an actuation of the inoperable radiation monitoring channel places the system in the design condition. This alignment would result in maintaining the control room envelope pressurized and increases the protection afforded to the operators.

(f) The change in terminology does not change any requirements or actions in the Specification. Therefore this change does not represent an increase in the probability or consequences of any accident previously evaluated.

Based on the above discussion, the individual changes do not represent an increase in the probability or consequences of any accident previously evaluated.

In addition to the changes proposed to controls over Control Room Ventilation, Fuel Handling Building HVAC, and associated actuation logic, an administrative change is proposed to remove the footnote at the bottom of page 3/4 7–20. Since the footnote no longer has meaning or relevance to the operation of the facility, its removal does not increase the probability or consequences of any accident previously evaluated.

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

The proposed changes make the existing Specifications internally consistent, manually align a system to the actuated position, provide an alternative measure that assures [that] a safety function which is unavailable is not required to [be] perform[ed], provide an extended period of allowance for all trains of a system to be inoperable, and redefines system operability to reflect its functional design. The proposed changes do not introduce any new equipment into the plant or significantly alter the manner in which existing equipment will be operated. The systems affected by the proposed changes are not identified as contributing causal factors in design basis accidents, their function is to assist in mitigation of accidents postulated to occur. Since the proposed changes do not allow activities that are significantly different from those presently allowed, no possibility exists for a new or different kind of accident from those previously evaluated.

In addition to the changes proposed to controls over reactivity changes, an administrative change is proposed to remove the footnote at the bottom of page 3/4 7–20. Since the footnote does not perform any function and will never again apply to plant operations, its removal cannot create the possibility of a new or different kind of accident from those previously evaluated.

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

The proposed changes do not involve a significant reduction in a margin of safety because the ability of the Fuel Handling Building HVAC and Control Room Ventilation Systems will be maintained. The margin of safety is defined by the ability of the systems to limit the release of radioactive materials and limit exposures to operators respectively following a postulated design basis accident. The only aspect of the proposed change that can be postulated to have any effect on a margin of safety is the proposed allowance for all trains of Control Room Ventilation or Fuel Handling Building HVAC to be inoperable for a limited period. The low probability of a design basis event that would require the system to perform its safety function during the limited period allowed by the proposed action assures that the change does not involve a significant change in a margin of safety. Therefore, the proposed changes do not significantly affect these operating restrictions and the margin of safety which support the ability to make and maintain the reactor in a safe shutdown and limit the release of radioactive material is not affected.

In addition to the changes described above, an administrative change is proposed to remove the footnote at the bottom of page 3/4 7–20. Since the footnote is no longer applicable to the facility, its removal 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 standards of 10 CFR 50.92© 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, NW, Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

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: September 29, 1998.

Description of amendment request: The licensee proposes to use a revised methodology to calculate mass and energy release following a postulated large-break loss-of-coolant accident. The amendment request also included proposed changes to the Updated Final Safety Analysis Report.

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

1. The proposed change does not involve a significant increase in the probability or

consequences of an accident previously evaluated.

This proposal updates the design large break loss of coolant accident (LBLOCA) analysis and methodology described in the UFSAR to support replacement of Westinghouse Model E Original Steam Generators (OSG) with Westinghouse Delta-94 Replacement Steam Generators (RSG).

A safety analysis has been performed, including evaluation of existing analyses and performance of bounding or confirming calculations, to determine effects of the proposed changes.

Analysis of mass and energy releases and resultant containment pressure and temperature response for the RSG concluded a small reduction in peak pressure and temperature for the RSG compared to the OSG. Thus, the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated.

Changes to the LBLOCA model caused by installation of the RSGs and associated changes in analysis methodology result in no change in radiological consequence as delineated in 10 CFR 100 and the Standard Review Plan (NUREG–0800). Consequences of this design basis accident have not increased.

Thus, changes in the LBLOCA design basis event analysis associated with replacement of OSGs with RSGs do 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.

This proposal updates the design basis large break loss of coolant accident (LBLOCA) analysis and methodology described in the Updated Final Safety Analysis Report (UFSAR) to support replacement of OSGs with RSGs.

Fit, form, and design function of RSG equipment is not significantly changed from OSG equipment. Analyses of LBLOCA mass and energy releases and resultant containment system response indicates that performance with RSGs remains within the existing design limits. Thus, the proposed 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.

A safety analysis has been performed, including evaluations of existing analyses and performance of bounding and/or confirming calculations, to determine the effect of the proposed changes. Results of these analyses demonstrate that the proposed license amendment and operation of STP Units with Delta-94 steam generators installed will not produce post-accident Containment pressures or temperatures exceeding existing Technical Specification limits. Consequently, there are no effects on dose analyses due to design basis LBLOCA performance of the RSGs. Radiological consequences of the postulated accident did not change, and all results remain within the acceptance criteria of 10 CFR 100 and the Standard Review Plan (NUREG–0800).

Thus, the change in LBLOCA analysis results and methodology descriptions in the UFSAR associated with replacement of Model E steam generators with Delta-94 steam generators do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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, NW, Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

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: September 30, 1998.

Description of amendment request: The proposed amendment would change the Updated Final Safety Analysis Report and revise the offsite dose licensing basis to account for operation of the existing steam generators at reduced feedwater inlet temperatures, and to account for operation of the new replacement steam generators. The calculated offsite dose consequences would increase for the main steamline break, reactor coolant pump shaft seizure, and rod cluster control assembly ejection accidents. The proposed increases in offsite doses are minimal and all doses remain below the dose limits for their respective accidents, as specified by 10 CFR Part 100 and the Standard Review Plan (NUREG–0800).

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.

This document updates the facilities' radiological design basis, as described in the Updated Final Safety Analysis Report, to address both a reduction in allowed nominal feedwater temperature for Model E steam generators from 440 °F to 420 °F and the replacement of Model E steam generators with Delta-94 steam generators. Therefore, these changes do not change the probability of an accident previously evaluated.

A safety analysis has been performed, including evaluations of existing analyses and performance of bounding and/or confirming calculations, to determine the impact of the proposed changes. Effects on the dose analyses due to the accompanying physical changes to the plant are slight. However, some improvements were made to the analytical models used in the analyses. These improvements were responsible for the majority of the increase in offsite doses. While the radiological consequences of some postulated accidents increased, all results remain within the acceptance criteria, as defined in 10 CFR 100 and the Standard Review Plan (NUREG–0800).

The radiological consequences of the postulated accidents remain within their respective acceptance criteria with the use of the revised analysis methodologies. Therefore, the change to allow operation of the Model E steam generators at a reduced feedwater temperature of 420°F and the replacement of Model E steam generators with Delta-94 steam generators do 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.

This document updates the facilities' radiological design basis, as described in the Updated Final Safety Analysis Report, to address both a reduction in allowed nominal feedwater temperature for Model E steam generators from 440 °F to 420 °F and the replacement of Model E steam generators with Delta-94 steam generators. Since the proposed changes to the Updated Final Safety Analysis Report are analytical in nature, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

A safety analysis has been performed, including evaluations of existing analyses and performance of bounding and/or confirming calculations, to determine the impact of the proposed changes. Effects on the dose analyses due to the accompanying physical changes to the plant are slight. However, some improvements were made to the analytical models used in the analyses. These improvements were responsible for the majority of the increase in offsite doses. While the radiological consequences of some postulated accidents increased, all results remain within the acceptance criteria, as delineated in 10 CFR 100 and the Standard Review Plan (NUREG–0800), for the respective accidents.

The radiological consequences of the postulated accidents remain within their respective acceptance criteria with the use of the revised analysis methodologies. Therefore, the change to allow operation of the Model E steam generators at a reduced feedwater temperature of 420 °F and the replacement of Model E steam generators with Delta-94 steam generators do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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.

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

Date of amendment request: October 27, 1998.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.8.2.3, ''Electrical Power Systems—DC Distribution—Operating,'' and the associated bases. The surveillance requirements for battery testing would be revised.

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

The Davis-Besse Nuclear Power Station (DBNPS) has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are adversely affected by the proposed changes to station battery testing methodology and frequency.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are adversely affected by the proposed changes in station battery testing methodology and frequency. The proposed changes do not alter the source term, containment isolation, or allowable radiological releases. The proposed changes are consistent with the most recent IEEE Standard 450–1995, ''IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,'' and the ''Improved Standard Technical Specifications for Babcock and Wilcox Plants,'' NUREG–1430, Revision 1.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. The batteries are not an initiator or contributor to the initiation of an accident. No new accident scenarios, transient precursors, failure mechanisms, or limiting faults are introduced as a result of the proposed changes.

3. Not involve a significant reduction in a margin of safety because the proposed TS changes do not significantly reduce or adversely affect the capabilities of any plant structures, systems or components. These changes increase the effectiveness and frequency of the battery tests being performed. Therefore, there is not 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 Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: October 27, 1998.

Description of amendment request: The proposed amendment would relocate a Technical Specification (TS) surveillance requirement from TS Section 3/4.6.5.1, ''Shield Building-Emergency Ventilation System'' to TS Section 3/4.6.5.2, ''Shield Building Integrity.'' Administrative and bases changes would also be made.

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiator is affected by the proposed changes to the Technical Specifications (TS) Index; TS Definition 1.6, ''Shield Building Integrity''; TS 3/4.6.5.1, ''Emergency Ventilation System''; TS 3/4.6.5.2, ''Shield Building Integrity''; TS Bases 3/4.6.5.1, ''Emergency Ventilation System''; or TS Bases 3/4.6.5.2, ''Shield Building Integrity.''

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions and assumptions are significantly affected by the above proposed changes. The proposed change to relocate existing TS Surveillance Requirement (SR) 4.6.5.1.d.4 to TS 3/4.6.5.2, and the subsequent application of the Limiting Condition for Operation (LCO) of TS 3/4.6.5.2 should the Emergency Ventilation System (EVS) be unable to produce the required negative pressure in the annulus space due to an opening in the ventilation boundary, would allow 24 hours to restore the capability of maintaining the required negative pressure in the annulus. The current SR 4.6.5.1.d.4 and associated TS LCO 3.6.5.1 would require entry into TS 3.0.3, thereby allowing only one hour for restoration before commencing plant shutdown. The allowed outage time of 24 hours is reasonable considering the limited leakage design of containment and the low likelihood of a Design Basis Accident (DBA) occurring during this time period. The proposed changes are consistent with the guidance of the ''Improved Standard Technical Specifications for Combustion Engineering Plants,'' NUREG–1432, Revision 1 and the ''Improved Standard Technical Specifications for Westinghouse Plants,'' NUREG–1431, Revision 1. The ''Improved Standard Technical Specifications for Babcock and Wilcox Plants,'' NUREG–1430, Revision 1 does not contain guidance for shield building integrity because the DBNPS is the only Babcock and Wilcox-type plant with the containment vessel/annulus space/shield building design. The proposed changes do not alter the drawdown capability of the EVS. Since the likelihood of a DBA occurring during this 24 hour period is low and the containment is of a low leakage design, the radiological consequences of a previously evaluated accident are not significantly increased. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. No new accident scenarios, transient precursors, failure mechanisms, or limiting failures are introduced as a result of the proposed changes.

3. Not involve a significant reduction in a margin of safety because the proposed TS changes do not significantly reduce or significantly adversely affect the capabilities of any plant structures, systems or

components. The capability of the shield building/EVS to respond when necessary and to maintain a negative pressure will not be significantly changed by these proposed TS changes. Accordingly, there is not 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 Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: October 28, 1998.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 6, ''Administrative Controls.'' Several requirements would be modified and/or relocated to the Updated Safety Analysis Report.

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions or assumptions are affected by the proposed changes to Technical Specification (TS) 6.5.1.6 ''[Station Review Board] Responsibilities''; TS 6.8.4.d, ''Radioactive Effluents Control Program''; TS 6.10, ''Record Retention''; TS 6.11, ''Radiation Protection Program''; TS 6.12, ''High Radiation Area''; and TS 6.15, ''Offsite Dose Calculation Manual (ODCM).''

These changes proposed to TS 6.5.1.6, TS 6.8.4.d, TS 6.10, and TS 6.15 are administrative changes that improve or update the content of TS Section 6.0, ''Administrative Controls.''

The change proposed to TS 6.11 would relocate its content to the DBNPS Updated Safety Analysis Report, thereby removing it from the TS consistent with the NRC's NUREG–1430, Revision 1, ''Improved Standard Technical Specifications for Babcock and Wilcox Plants.''

The changes proposed to TS 6.12 are based upon the current revision to 10 CFR Part 20, ''Standards for Protection Against Radiation,'' as published in the **Federal Register**, dated August 15, 1994, and TS approved by the NRC for the San Onofre Nuclear Generating Station Units 2 and 3 in Operating License Amendments 127 and 116, respectively. The changes to TS 6.12 also provide for the use of alternative methods for controlling access to high radiation areas and state-of-the-art radiation protection monitoring methods, such as closed circuit television and telemetry.

Under the proposed changes, the TS would continue to satisfy the applicable requirements of 10 CFR 50.36(c)(5).

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes. As described above, these changes are administrative changes or are proposed pursuant to the current revision to 10 CFR Part 20, ''Standards for Protection Against Radiation.'' The proposed changes do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. As described above, these changes are administrative changes or are proposed pursuant to the current revision to 10 CFR Part 20, ''Standards for Protection Against Radiation.''

3. Not involve a significant reduction in a margin of safety because the proposed changes are administrative changes or are proposed pursuant to the current 10 CFR Part 20 requirements. These proposed changes do not reduce or adversely affect the capabilities of any plant structures, systems or components.

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 Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: October 27, 1998.

Description of amendment request: This proposed amendment request would modify Technical Specification (TS) 4.2.b, ''Steam Generator Tubes,'' to redefine the plugging limits for the Westinghouse Hybrid Expansion Joint sleeves (HEJs) and Westinghouse Laser Welded Sleeves (LWSs). Additional administrative changes are also proposed. The proposed changes are as follows:

1. TS 4.2.b.3.c.1 would be changed to correct an oversight from a previous amendment. The current TS 4.2.b.2.c.1 makes reference to TS 3.4.a.1.C. This reference is no longer valid because TS 3.4.a.1.C became TS 3.4.d as a result of TS Amendment 123. This change corrects an oversight from a previous amendment and is administrative.

2. TS 4.2.b.4.a would be revised to specify the updated revision of WCAP–14685 and the addendum to WCAP–13088.

3. TS 4.2.b.4.b would be revised to specify the corrected value for the plugging limit of the Westinghouse mechanical HEJ sleeves. The plugging limit would change from 24 percent to 23 percent or more sleeve wall degradation.

4. TS 4.2.b.4.e would be revised to specify the corrected value for the plugging limit of Westinghouse laser welded sleeves. The plugging limit would change from 25 percent to 23 percent or more sleeve wall degradation.

The associated bases pages for TS Section 4.2 would also be modified to reflect the above changes.

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.

The analysis of change in plugging limits was performed in accordance with RG 1.121 and ASME B&PV Code and, therefore, all required safety factors are met. The plugging limit or allowed degraded wall thickness value is not used in any accident analyses; therefore, this change has no significant effect on any previously evaluated accidents. The change does not significantly increase the probability or consequences of an accident previously evaluated.

Because the maximum primary-to-secondary differential pressure parameter has changed, the conventional analysis techniques originally used to qualify the required weld width under predicted the shear stress in the LWS and LWR [laser weld repair] of HEJ welds. Consequently, a verification program using experimental analysis, as allowed by Section III of the ASME B&PV Code, was performed to show that the weld remains in compliance with the ASME B&PV Code. Using a different analysis technique to verify that the previously approved weld width for LWS and LWR of HEJs is still accurate does not increase the probability or consequences of an accident previously evaluated.

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

Recalculating the allowable sleeve wall degradation and plugging limits and verifying the acceptability of the 0.015 inch weld width ensures that currently approved conditions are maintained. Requiring tubes to be plugged at a smaller sleeve wall degradation value does not result in any new or different conditions which could create a new or different accident.

Verification of the currently approved weld width using a different analysis technique does not have a physical effect on any plant equipment or operating parameters and, therefore, can not create a new or different kind of accident.

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

These TS changes are being made to ensure that the current margins of safety are maintained. This is accomplished by reducing the allowable sleeve wall degradation and plugging limit. Verifying the required, minimum weld width by an allowed, alternate analysis technique, as described by ASME B&PV Code, ensures that an adequate margin of safety is maintained and there is not a significant reduction in the margin of safety.

The minor administrative changes do not impact the technical content or implementation of the TS and therefore can not create a significant hazard.

The changes to the steam generator tube and sleeve plugging limits are necessary because of an increase in the normal operating differential pressure between the primary and secondary coolant systems. The differential pressure was increased as a result of the effects of extensive tube plugging on primary to secondary heat transfer. Since, per Regulatory Guide 1.121, the safety factor for mimimum acceptable wall thickness for steam generator tubes is based on normal operating pressures, it was found necessary to recalculate the plugging limits.

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

Local Public Document Room location: 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.

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

Date of amendment request: October 23, 1998.

Description of amendment request: The amendment would revise Technical Specification 3.5.1, ''Emergency Core Cooling Systems—Accumulators,'' to increase the allowed outage time for the accumulators from 1 hour to 24 hours if an accumulator is inoperable for reasons other than not meeting its boron concentration requirements.

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

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

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The overall protection system performance will remain within the bounds of the accident analyses documented in Chapter 15 of the Updated Safety Analysis Report (USAR), WCAP–10961–P, and WCAP–11883, since no hardware changes are proposed. The impact of the increase in the accumulator AOT on core damage frequency for all the cases evaluated in WCAP–15049 is within the acceptance limit of 1.0E–06/yr for a total plant CDF less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049 for the accumulator AOT increase meet the criterion of 5E–07 in Regulatory Guide DG–1065 for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049, design basis accumulator success criteria are not considered necessary to mitigate large break LOCA events, and was only included in the WCAP–15049 evaluation as a worst case data point. In addition, WCAP–15049 states that the NRC has indicated that an ICCDP greater than 5E–07 does not necessarily mean the change is unacceptable.

The safety injection accumulators are credited in Section 15.6.5 of the Updated Safety Analysis Report for large and small break LOCA. There will be no effect on these analyses, or any other accident analysis, since the analysis assumptions are unaffected and remain the same as discussed in Section 15.6.5. Design basis accidents are not assumed to occur during allowed outage times covered by the Technical Specifications. As such, the ECCS Evaluation Model equipment availability assumptions made in Section 15.6.5 remain valid.

The safety injection accumulators will continue to function in a manner consistent with the above analysis assumptions and the plant design basis. As such, there will be no degradation in the performance of, nor an increase in the number of challenges to, equipment assumed to function during an accident situation.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs. Therefore, this change will not increase the probability of an accident or malfunction.

The corresponding increase in CDF due to the proposed change to increase the AOT of the accumulators from one hour to 24 hours is not significant. Pursuant to the guidance in Section 3.5 of NEI 96–07, Revision 0, ''Guidelines for 10 CFR 50.59 Safety Evaluations,'' the proposed increase in AOT does not ''degrade below the design basis the performance of a safety system assumed to function in the accident analysis,'' nor does it ''increase challenges to safety systems assumed to function in the accident analysis such that safety system performance is degraded below the design basis without compensating effects.''

Therefore, it is concluded that this change does not increase the probability of occurrence of a malfunction of equipment important to safety.

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 does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change does not involve any change to the installed plant systems or the overall operating philosophy of WCGS.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049 evaluation, the plant design will not be changed with this proposed Technical Specification AOT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator AOT increase has a very small impact on core damage frequency. The WCAP–15049 evaluation demonstrates that the small increase in risk due to increasing the accumulator AOT is within the acceptance criteria provided in Draft Regulatory Guide DG–1065. No new accident or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore, the possibility of a new or different malfunction of safety related equipment is not created.

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

The proposed change does not involve a significant reduction in a margin of safety. There will be no change to the Departure from Nucleate Boiling Ratio (DNBR) Correlation Limit, the design DNBR limits, or the safety analysis DNBR limits discussed in Bases Section 2.1.1.

The basis for the accumulator LCO, as discussed in Bases Section 3/4.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of the WCAP–15049 evaluation, the proposed change will allow plant operation in a configuration outside the design basis for up to 24 hours, instead of 1 hour, before being required to begin shutdown. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming design basis accumulator success criteria is necessary to mitigate the event, has a very small impact on plant risk. Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049 evaluation, the impact of increasing the accumulator AOT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049 evaluation. It is therefore concluded that the proposed change does not involve a significant reduction in the margin of safety as described in Technical Specification Bases Section 3/4.5.1.

As discussed previously, the performance of the accumulators will remain within the assumptions used in the large and small break LOCA analyses, as presented in USAR Section 15.6.5. Also, there will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

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

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

NRC Project Director: William H. Bateman.

Yankee Atomic Electric Company, Docket No. 50–029, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of amendment request: October 15, 1998.

Description of amendment request: The licensee proposes to extend the interval of submission of Effluent and Waste Disposal Reports from semi-annual to annual pursuant to 10 CFR 50.36a(a)(2). This action would require a change to Technical Specification (TS) 6.8.2.b, a reporting requirement, and textual changes in other parts of the TS to make the change consistent throughout.

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 changes to the Yankee Nuclear Power Station Defueled Technical Specifications proposed above are administrative in nature. The proposed changes are consistent with the revised 10 CFR 50.36a, ''Technical specifications on effluents from nuclear power reactors,'' which require the submittal of one Radioactive Effluent Release Report per year. Furthermore, the NRC has already concluded in issuing the 10 CFR 50.36a rule change that implementation of the proposed technical specifications changes would not result in a reduction to the public health and safety or common defense and security.

As such, the changes:

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

The administrative nature of the changes do not affect the operation of YNPS in the permanently defueled condition. Furthermore, the changes do not result in a change to the plant design, configuration, or operating procedures. Because the physical plant is not affected, and the only change is the frequency with which reports are submitted to the NRC, the probability of an accident previously evaluated is not increased and the radiological consequences of an accident previously evaluated are not increased.

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

The changes described do not modify the design, configuration, or operating procedures for any plant systems or components. The accident analyses for the facility are not affected by the proposed changes. The changes do not introduce any new failure mechanisms. Therefore, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Will not involve a significant reduction in the margin of safety.

The changes described are administrative in nature. The changes do not modify the design, configuration, or operating procedures for any plant systems or components. The changes do not affect the facility's accident analyses. Radioactive effluent release limits remain unchanged. The submittal of reports to the NRC is an administrative function and is not included in the bases of any Technical Specifications to define or establish a margin of safety. Therefore, the proposed changes do not reduce the margin of safety as defined in the bases of any Technical Specifications.

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

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Project Director: Seymour H. Weiss.

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.

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of amendment request: October 16, 1998.

Description of amendment request: The amendment would change Technical Specification (TS) 3.3.1, ''Reactor Protective System Instrumentation—Operating'' and TS 3.3.2, ''Reactor Protective System Instrumentation Shutdown'' to clarify an inconsistency between TS wording and the design basis as described in the TS Bases and the Updated Final Safety Analysis Report.

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

Expiration date of individual notice: November 27, 1998.

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

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: August 17, 1998.

Brief description of amendments: The amendments revise Technical Specification 5.2.2.f regarding the senior reactor operator licensing requirement for the operations manager.

Date of issuance: November 4, 1998.

Effective date: November 4, 1998.

Amendment Nos.: 204 and 234.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments revise the facility's Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48258) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 4, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 19, 1998.

Brief description of amendment: This amendment revises Section 5.4.8 of the Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR) such that it incorporates the use of a freeze seal as a temporary part of the reactor coolant pressure boundary.

Date of Issuance: November 4, 1998.

Effective date: November 4, 1998.

Amendment No. 201.

Facility Operating License No. DPR–16. Amendment revised the UFSAR.

Date of initial notice in **Federal Register**: September 30, 1998 (63 FR 52307).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 4, 1998. .

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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

Date of application for amendment: July 23, 1998, as supplemented September 25, 1998. The September 25, 1998, supplement did not change the initial proposed no significant hazards consideration determination.

Brief description of amendment: The amendment establishes that the existing Safety Limit Minimum Critical Power Ratio in Technical Specification 2.1.A is applicable for Cycle 17.

Date of Issuance: November 5, 1998.

Effective date: November 5, 1998, to be implemented within 30 days.

Amendment No.: 202.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 26, 1998 (63 FR 45525).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 5, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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

Date of application for amendments: January 29, 1997, as supplemented February 11, 12, March 7, 10, 11, 19, 20, April 29, June 30, and July 10,1997, June 20, June 22, July 24, September 15, and October 1, 1998.

Brief description of amendments: The amendments change the design basis of the cooling water system emergency intake line flow capacity. The changes also reclassify the intake canal for use during a seismic event, which would be an additional source of cooling water available during a design-basis earthquake. The amendments also reflect the completion of license conditions that were implemented as part of interim amendments 128/120 dated March 25, 1997, to reflect compensatory measures taken by Northern States Power until a seismically qualified emergency cooling water source could be provided.

Date of issuance: November 4, 1998.

Effective date: November 4, 1998, with full implementation within 30 days. Implementation of the USAR update shall be no later than June 1, 1999, as stated in License Condition 3.

Amendment Nos.: 140 and 131.

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

Date of initial notice in **Federal Register**: October 1, 1998 (63 FR 52772). The October 1, 1998, submittal provided revised USAR pages reflecting the change to the cooling water system emergency intake design bases. This information was within the scope of the October 1, 1998, **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 4, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 6, 1998.

Brief description of amendments: The proposed changes modify the technical specifications (TS) to eliminate reference to shutdown cooling (SDC) system isolation bypass valve inverters. This allows the licensee to replace the inverters with transfer switches.

Date of issuance: October 26, 1998.

Effective date: October 26, 1998, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2—143; Unit 3—134.

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

Date of initial notice in **Federal Register**: September 23, 1998 (63 FR 50939).

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

No significant hazards consideration comments received: No.

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

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

Date of amendments request: December 30, 1997, as supplemented by letter dated April 9, 1998.

Brief Description of amendments: The amendments change the Technical Specifications to revise the surveillance requirements for the Auxiliary Building and Service Water Building batteries to remove the existing 1.75 volt minimum individual cell voltage associated with the ''service test'' acceptance criterion and replace it with a reference to the battery load profile specified in the Final Safety Analysis Report, Section 8.3.2.

Date of issuance: November 3, 1998.

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

Amendment Nos.: Unit 1—139; Unit 2—131.

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

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17234). The April 9, 1998, letter provided clarifying information that did not change the scope of the December 30, 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 November 3, 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 No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: August 6, 1998.

Brief description of amendment: Change Technical Specifications (TS) Surveillance and Bases Sections 3.3.2, ''ESFAS Instrumentation,'' and 3.7.5, ''AFW System'' to clarify the intent of the surveillance testing requirements for the turbine driven auxiliary feedwater pump, which is consistent with the wording and intent of the Westinghouse Improved TS.

Date of issuance: October 26, 1998.

Effective date: October 26, 1998.

Amendment No.: 13.

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

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

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

No significant hazards consideration comments received: None.

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

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

Date of application for amendments: November 6, 1996, as supplemented April 15, July 14, and October 16, 1998. The supplemental submittals contained clarifying information only, and did not change the initial no significant hazards consideration determination.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Sections 3.4.1.4, 4.4.1.4, 3.4.1.5, 3.4.1.6, 4.4.1.6.1, 4.4.1.6.2, 4.4.1.6.3, 3/4.4.2 and 3/4.4.3 for Unit 1, and 3.4.1.4, 4.4.1.4, 3.4.1.5, 3/4.4, 3.4.1.6, 4.4.1.6.1, 4.4.1.6.2, and 4.4.1.6.3 for Unit 2, modifying the requirements for isolated loop startup to permit filling of a drained isolated loop via backfill from the reactor coolant system through partially opened loop stop valves.

Date of issuance: October 30, 1998.

Effective date: October 30, 1998.

Amendment Nos.: 215 and 196.

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

Date of initial notice in **Federal Register**: December 4, 1996 (61 FR 64396).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 30, 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.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By December 18, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

Date of application for amendments: October 23, 1998, as supplemented October 26, 1998.

Brief description of amendments: The amendments clarify the conditions that constitute operable Individual Rod Position Indication (IRPI) system channels, provide for an allowed out of service time for inoperable IRPI indicator channels, and provide compensatory measures to be taken when any channel is determined to be inoperable.

Date of issuance: October 30, 1998.

Effective date: October 30, 1998.

Amendment Nos.: 139 and 130.

Facility Operating License Nos. DPR–42 and DPR–60. Amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated October 30, 1998.

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

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

NRC Project Director: Cynthia A. Carpenter.

Dated at Rockville, Maryland, this 10th day of November 1998.

For the Nuclear Regulatory Commission.

William H. Bateman,

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

[FR Doc. 98–30691 Filed 11–17–98; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination

SUMMARY: This notice provides supplemental information regarding implementation of the Nuclear Regulatory Commission's (NRC's) Final Rule on Radiological Criteria for License Termination (License Termination Rule, LTR) which was issued on July 21, 1997 (62 FR 39058). The information provided in this notice pertains to: (1) The end of the ''grandfathering period'' on August 20, 1998; (2) issuance of the draft regulatory guide on the LTR for interim use; (3) availability of the NRC's screening computer code (DandD, Version 1) for calculating screening values to demonstrate compliance with the dose limits in the LTR; (4) screening values for building surface contamination for beta/gamma radiation emitters; (5) NRC plans to hold public workshops to discuss issues related to the draft guidance and implementation of the LTR; (6) staff plans to develop a standard review plan (SRP) for decommissioning; and (7) status of NRC decommissioning guidance documents.

SUPPLEMENTARY INFORMATION:

1. End of the Grandfathering Period

Subpart E to 10 CFR Part 20 contains a provision, 20.1401(b)(3), that the criteria in the LTR do not apply to sites that submit a sufficient decommissioning plan (DP) or license termination plan (LTP) before August 20, 1998, provided the NRC approves the DP or the LTP before August 20, 1999, and the plan is in accordance with the criteria identified in the Site Decommissioning Management Plan (SDMP) Action Plan (57 FR 13389; April 16, 1992). The period from the effective date of the LTR, August 20, 1997 through August 20, 1998, is referred to as the ''grandfathering period,'' during which the criteria in the SDMP Action Plan could continue to be proposed. This notice reminds licensees that the grandfathering period has ended, and that all future requests to terminate a license must be in accordance with the provisions in Part 20, Subpart E. Note that the NRC review of the licensee plans submitted in accordance with 10 CFR 20.1401(b)(3), incorporating the SDMP Action Plan criteria, will continue through August 20, 1999.

2. Draft Regulatory Guide

The NRC has issued Draft Regulatory Guide DG–4006, ''Demonstrating Compliance with the Radiological Criteria For License Termination,'' for a two-year interim use period (i.e., July 8, 1998 through July 7, 2000). NRC has also issued draft NUREG reports in support of DG–4006 (the applicable draft NUREG reports are referenced in DG–4006). A notice of availability of the Draft Regulatory Guide was published in the **Federal Register** on August 4, 1998 (63 FR 41604).

3. Availability of NRC DandD Screening Code

On August 20, 1998, NRC issued a screening computer code DandD, Version 1. The DandD code, when used with default parameters, is an acceptable method for licensees to calculate screening values to demonstrate compliance with the unrestricted use dose limit in the LTR. The DandD code can be installed by downloading the self-extracting program file, setup.exe, accessed through the web site: ''http:/techconf.llnl.gov/radcri/java.html,'' clicking on ''dose assessment,'' and then on ''decontamination and decommissioning software.'' The installation instruction file ''readme.txt'' can also be downloaded, using the above web site, to help users installing the code. Important support documents (e.g., NUREG–1549, ''Decision Methods for Dose Assessment to Comply With Radiological Criteria for License Termination'' and NUREG/CR–5512, Vol. #3, ''Residual Radioactive