constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Tuesday, July 10, 2001, 3:00 p.m. until 5:00 p.m.

The Subcommittee will discuss proposed ACRS activities and related matters. 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. A portion of this meeting 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.

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. (EDT). 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: June 21, 2001.

James E. Lyons,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 01–16095 Filed 6–26–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting of the ACRS Subcommittee on Plant Operations; Notice of Meeting

The ACRS Subcommittee on Plant Operations will hold a meeting on July 9, 2001, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Monday, July 9, 2001—9:30 a.m. until 12:30 p.m.

The Subcommittee will continue its discussion of the Reactor Oversight Process. 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 and 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 engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore, can be obtained by contacting the cognizant ACRS staff engineer, Ms. Maggalean W. Weston (telephone: 301/415–3151) between 8:00 a.m. and 5:30 p.m. (EDT). 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 potential changes to the agenda, etc., that may have occurred.

Dated: June 21, 2001.

James E. Lyons,

Associate Director for Technical Support.

[FR Doc. 01–16096 Filed 6–26–01; 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 Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 4 through June 15, 2001. The last biweekly notice was published on June 12, 2001 (66 FR 31700).

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 Administrative 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By July 27, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by email to pdr@nrc.gov.

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

Date of amendments request: June 4, 2001.

Description of amendments request: The proposed license amendments revise, from 2 hours to 6 hours, the time period in Surveillance Requirement 3.6.1.6.1 for verifying that each suppression chamber-to-drywell vacuum breaker is closed after any discharge of steam to the suppression chamber from any source. In conjunction with this change, the Completion Time associated with Required Action B.1 for closing an open vacuum breaker is being revised from 8 hours to 4 hours.

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

The proposed changes provide additional time to verify that each vacuum breaker is closed and reduce the time allowed for closing an open vacuum breaker. The safety functions of the suppression chamber-to-drywell vacuum breaker valves are to relieve vacuum in the drywell following a postulated loss-of-coolant accident and to remain closed, except when the vacuum breakers are performing their intended design function, in order to ensure that no excessive bypass leakage occurs from the drywell to the suppression chamber. With a vacuum breaker not closed, communication between the drywell and suppression chamber airspaces could occur and, if a loss-of-coolant accident were to occur, there would be the potential for primary containment overpressurization due [to] steam leakage from the drywell to the suppression chamber without quenching. The vacuum breakers do not perform a safety function that initiates, or alters initiation of, an accident previously evaluated. Rather, the vacuum breakers function to mitigate the consequences of certain design basis accidents. Therefore, the proposed changes do not involve an increase in the probability of an accident previously evaluated or the method of performing their safety functions.

As noted above, the vacuum breakers function to mitigate the consequences of certain design basis accidents. The proposed changes to the Surveillance Requirement and Completion Time provide additional time to verify that each vacuum breaker is closed and reduce the time allowed for closing an open vacuum breaker; however, the proposed changes do not alter the safety functions of the vacuum breakers. When performing the surveillance to verify each vacuum breaker is closed, the expected result is the verification that the component is indeed closed. However, if this surveillance result is not obtained, the Technical Specifications limit the time allowed to close the vacuum breaker. Additional time is being provided to verify that each vacuum breaker is closed; however, the overall time allowed for closing and verifying closure of a vacuum breaker is not being increased. Since the overall time to take action for an open vacuum breaker has not been increased, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

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

The suppression chamber-to-drywell vacuum breakers are not an initiator of any design basis accident. Rather, the safety functions of the vacuum breaker valves are to relieve vacuum in the drywell following a loss-of-coolant accident and to remain closed when not relieving vacuum to ensure that no excessive bypass leakage occurs from the drywell to the suppression chamber. Neither safety function of these vacuum breakers is altered by the proposed changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not affect the ability of the suppression chamber-to-drywell vacuum breakers to perform their safety functions. Rather, as previously stated, the proposed changes provide additional time to verify that each vacuum breaker is closed and reduce the time allowed for closing an open or inoperable vacuum breaker. As a result, the overall time for taking action for an open vacuum breaker is unchanged. The vacuum breakers will continue to be verified closed every 14 days, as part of a required functional test of the vacuum breaker every 31 days, and following any activity involving the discharge of steam to the suppression chamber. If a vacuum breaker is found to be open and cannot be closed as required, plant shutdown will continue to be required within the same time requirements as currently specified in the Technical Specifications. Current Technical Specifications allow up to 10 hours to close an open vacuum breaker (i.e., 2 hours to perform the surveillance to verify vacuum breaker closure and, if necessary, 8 hours to close the vacuum breaker). The proposed change maintains the 10 hour limit by reducing the time to 4 hours to close an open or inoperable vacuum breaker while increasing the time to 6 hours to complete the surveillance to verify vacuum breaker closure. Thus, on this basis, the proposed license amendments will not change overall plant risk and do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Patrick M. Madden, Acting.

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

Date of amendment request: May 18, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.9.4 ''Containment Building Penetrations'' and the associated Bases to permit containment building penetrations to remain open, under administrative controls, during core alterations or the movement of irradiated fuel within the containment. Specifically, the licensee proposes: (1) Incorporating an alternate source term methodology in the fuel handling accident analysis; (2) revising TS 3.9.4 to remove portions of a note restricting the applicability of administrative controls with respect to containment penetrations; and (3) including the use of administrative controls on the equipment hatch and other penetrations that provide access from containment atmosphere to outside atmosphere.

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

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

The proposed changes modify TS requirements previously reviewed and approved by the NRC in improved Technical Specifications (ITS) and changes to ITS as described in TSTF [Technical Specification Task Force]-312. An alternate source term calculation has been performed for the HNP [Harris Nuclear Plant] that demonstrates that dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed change does not modify the design or operation of equipment used to move spent fuel or to perform core alterations[.]

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

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

Containment penetrations are designed to form part of the containment pressure boundary. The proposed change provides for administrative controls and operating restrictions for containment penetrations consistent with guidance approved by the NRC staff. Containment penetrations are not an accident initiating system as described in the Final Safety Analysis Report [FSAR]. The proposed change does not affect other Structures, Systems, or Components. The operation and design of containment penetrations in operational modes 1–4 will not be affected by this proposed change.

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

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

The proposed changes modify required Actions and Surveillance Requirements previously reviewed and approved by the NRC in improved Technical Specifications (ITS) and changes to ITS, TSTF–312. Additionally, the implementation of the alternate source term methodology is consistent with NRC Regulatory Guide 1.183. The proposed change to containment penetrations does not significantly affect any of the parameters that relate to the margin of safety as described in the Bases of the TS or the FSAR. Accordingly, NRC Acceptance Limits are not significantly affected by this change.

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

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

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

NRC Section Chief: Patrick M. Madden, Acting.

Duke Energy Corporation, et al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of amendment request: March 9, 2001.

Description of amendment request: The amendment will revise the cold leg elbow tap flow coefficients used in the determination of Reactor Coolant System (RCS) flow rate at Catawba Nuclear Station, Unit 2. No changes in Technical Specification are necessary for this amendment.

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 following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

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

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

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

First Standard

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. No component modification, system realignment, or change in operating procedure will occur which could affect the probability of any accident or transient. The revised cold leg elbow tap flow coefficients will not change the probability of actuation of any Engineered Safeguards Feature or other device. The actual Unit 2 RCS flow rate will not change. Therefore, the consequences of previously analyzed accidents will not change as a result of the revised flow coefficients.

Second Standard

The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No component modification or system realignment will occur which could create the possibility of a new event not previously considered. No change to any methods of plant operation will be required. The elbow taps are already in place, and are presently being used to monitor flow for Reactor Protection System purposes. They will not initiate any new events.

Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. The removal of some of the excess flow margin, which was introduced by the hot leg streaming flow penalties in later calorimetrics, will allow additional operating margin between the indicated flow and the Technical Specification minimum measured flow limit. The proposed changes in the cold leg elbow tap flow coefficients will continue to be conservative with respect to the analytical model flow predictions, since the proposed coefficients will continue to contain some hot leg streaming penalties from the calorimetric determined coefficients used in the average.

An increase in the RCS flow indication of approximately 1.0% will increase the margin to a reactor trip on low flow but will not adversely affect the plant response to low flow transients. Current UFSAR Chapter 15 transients that would be expected to cause a reactor trip on the RCS low flow trip setpoint are Partial Loss of Reactor Coolant Flow, Reactor Coolant Pump Shaft Seizure and Reactor Coolant Pump Shaft break transients. Three reactor trip functions provide protection for these transients, RCS low flow reactor trip, RCP undervoltage reactor trip and RCP underfrequency reactor trip. The transient analyses of these events assume the reactor is tripped on the low flow reactor trip setpoint. This is conservative and produces a more severe transient response since a reactor trip on undervoltage or underfrequency would normally be expected to trip the reactor sooner and therefore reduce the severity of these transients.

The RCS low flow reactor trip is currently set at 91% of the Technical Specification minimum measured flow of 390,000 gpm. The setpoint will not be revised as a result of this change, which means the transients relying on this function will behave in the same manner with the reactor trips occurring at essentially the same conditions as previously analyzed. Therefore, any small increase in the reactor trip margin gained by the small increase in the indicated RCS flow will not adversely affect the plant response during these low flow events.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

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

Attorney for licensee: Ms. Lisa F. Vaughn , Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard L. Emch, Jr.

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: May 23, 2001.

Description of amendment request: The amendment request proposes a change to the minimum critical power ratio safety limit (SLMCPR) and changes to the references for the analytical methods used to determine the core operating limits.

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?

The Minimum Critical Power Ratio (MCPR) safety limit is defined in the Bases to Technical Specification [TS] 2.1.1 as that limit which ''ensures that during normal operation and during AOOs [Anticipated Operational Occurrences], at least 99.9% of the fuel rods in the core do not experience transition boiling.'' The MCPR safety limit satisfies the requirements of General Design Criterion 10 of Appendix A to 10 CFR [Part] 50 regarding acceptable fuel design limits. The MCPR safety limit is re-evaluated for each reload using NRC [Nuclear Regulatory Commission]-approved methodologies. The analyses for RBS [River Bend Station] Cycle 11 have concluded that a two-loop MCPR safety limit of 1.08, based on the application of Framatome ANP Richland, Inc.'s [FRA–ANP] [(proprietary)] NRC-approved MCPR safety limit methodology, will ensure that this acceptance criterion is met. For single-loop operation, a MCPR safety limit of 1.10, also ensures that this acceptance criterion is met.

In addition to the MCPR safety limit, core operating limits are established to support the Technical Specification 3.2 requirements which ensure that the fuel design limits are not exceeded during any conditions of normal operation or in the event of any anticipated operational occurences (AOO). The methods used to determine the core operating limits for each operating cycle are based on methods previously found acceptable by the NRC and listed in TS section 5.6.5. A change to TS section 5.6.5 is requested to include the FRA–ANP methods in the list of NRC approved methods applicable to RBS. These NRC approved methods will continue to ensure that acceptable operating limits are established to protect the fuel cladding integrity during normal operation and in the event of an AOO.

The requested Technical Specification changes do not involve any plant modifications or operational changes that could affect system reliability or performance or that could affect the probability of operator error. The requested changes do not affect any postulated accident precursors, do not affect any accident mitigating systems, and do not introduce any new accident initiation mechanisms.

Therefore, these changes to the Minimum Critical Power Ration (MCPR) safety limit and to the list of methods used to determine the core operating limits do 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 kind of accident from any accident previously evaluated?

The ATRIUM–10 fuel to be used in Cycle 11 is of a design compatible with the co-resident GE–11. Therefore, the introduction of ATRIUM–10 fuel into the Cycle 11 core will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications. The proposed revised MCPR safety limits have accounted for the mixed fuel core and have been shown to be acceptable for Cycle 11 operation. Compliance with the criterion for incipient boiling transition continues to be ensured. The core operating limits will continue to be developed using NRC approved methods which also account for the mixed fuel core design. The proposed MCPR safety limits or methods for establishing the core operating limits do not result in the creation of any new precursors to an accident.

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

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

The MCPR safety limits have been evaluated in accordance with Framatome ANP Richland, Inc.'s NRC-approved cycle-specific safety limit methodology to ensure that during normal operation and during Anticipated Operational Occurrences (AOO's) at least 99.9% of the fuel rods in the core are not expected to experience transition boiling. On this basis, the implementation of this Framatome ANP Richland, Inc. methodology does not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: May 11, 2001.

Description of amendment request: This amendment revised the Technical Specifications to allow, on a one-time basis only, Entergy Nuclear Operations, Inc. to extend the allowed out-of-service time for the Residual Heat Removal Service Water (RHRSW) System from 7 days to 11 days. This amendment is only applicable during installation of the modification 00–12 to the ''B'' RHRSW Strainer.

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:

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

The CCDP [Conditional Core Damage Probability] due to this proposed change is calculated to be 4.33 E–8 (assuming no-risk significant SSC maintenance), which falls below the threshold probability of 1 E–6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide (Reference 2). The ICLERP [incremental conditional large early release probability] is calculated to be 8.85 E–8, which falls below the threshold probability of 1 E–7 for risk significance per Reference 2 [see application dated May 11, 2001].

This proposed change does not increase the consequences of an accident previously evaluated because all relevant accidents (LOCA) [loss-of-coolant accident] would result in the transfer of decay heat to the suppression pool. For this scenario, the same compliment of equipment will be available to achieve and maintain cold shutdown as is required by the current TS LCO [limiting condition for operation].

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

The proposed change does not physically alter the plant. As such, no new or different types of equipment will be installed. The new design for the RHRSW strainer packing gland will be evaluated under a separate 10 CFR 50.59 evaluation and is considered to be functionally equivalent for the purposes of this one-time-only proposed TS change.

The connection and use of a temporary hose for achieving limited containment heat removal in the event the ''A'' division of RHRSW is rendered inoperable for some reason is a contingency plan that is already addressed by current plant procedures.

Involve a significant reduction in a margin of safety.

The CCDP due to this proposed change is calculated to be 4.33 E–8 (assuming no-risk significant SSC maintenance). This value falls below the threshold probability of 1 E–6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide (Reference 2). The CLERP is calculated to be 8.85 E–8, which falls below the threshold probability of 1 E–7 for risk significance per Reference 2.

The consequences of a postulated accident occurring during the extended allowable out-of-service time are bounded by existing analyses, therefore, there is no significant reduction in a margin of safety.

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Richard P. Correia, Acting.

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

Date of amendment request: May 22, 2001.

Description of amendment request: The proposed change to Technical Specification (TS) 3/4.7.1.2, Emergency Feedwater (EFW) System expands and clarifies the current TS.

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

1. 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?

The administrative and more restrictive changes will not affect the assumptions, design parameters, or results of any accident previously evaluated. The accident mitigation features of the plant are not affected by these proposed changes. The proposed changes do not add or modify any existing equipment. The administrative change to test EFW pumps pursuant to the Inservice Test Program will ensure the EFW pumps are tested against the more restrictive of the data points required by either the safety analysis or the Inservice Test Program. Therefore, the proposed administrative changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

The less restrictive changes (allowing 7 days for an inoperable pump due to an inoperable steam supply, allowing 24 hours for an inoperable steam supply and one inoperable motor driven EFW pump, allowing 72 hours for two inoperable motor driven EFW pumps, performing Surveillance Requirements during other than shutdown conditions, allowing the use of actual actuation signals in addition to test signals, and delaying the requirement to complete Surveillance Requirement ''d'' to just prior to Mode 2) will not affect the assumptions, design parameters, or results of any accident previously evaluated. The accident mitigation features of the plant are not affected by these proposed changes. The proposed changes do not add or modify any existing equipment. Therefore, the proposed less restrictive changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

Therefore, this change does 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 kind of accident from any accident previously evaluated?

The proposed changes do not alter the design or configuration of the plant. There has been no physical change to plant systems, structures, or components. The proposed changes will not reduce the ability of any of the safety-related equipment required to mitigate Anticipated Operational Occurrences or accidents.

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

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

The proposed change to the LCO [Limiting Conditions for Operation] requiring three pumps and two flow paths be OPERABLE maintains the functionality of the EFW such that it is capable of performing its design function as assumed in the Final Safety Analysis Report. If the functionality of the system is not maintained, Technical Specifications require ACTIONs be taken, within specified time limitations, to restore EFW to OPERABLE status or shutdown the reactor. This action is consistent with the existing Technical Specifications and NUREG–1432.

The allowed outage time for one inoperable steam supply has been increased from 72 hours to 7 days in accordance with NUREG–1432. This is acceptable due to the redundant OPERABLE steam supply, the availability of redundant OPERABLE motor-driven EFW pumps, and the low probability of an event requiring the inoperable steam supply. This change is consistent with NUREG–1432 and has therefore been previously approved by the NRC [Nuclear Regulatory Commission].

The ACTION for an inoperable steam supply to the turbine-driven EFW pump steam turbine concurrent with one motor-driven EFW pump being inoperable will allow a 24 hour completion time. This change is acceptable based on the ability of the system to cool the reactor coolant system to shutdown cooling entry conditions following a loss of normal feedwater. The 24 hour completion time is reasonable based on the redundant OPERABLE steam supply to the turbine-driven EFW pump steam turbine, the OPERABLE motor-driven EFW pump, and the low probability of an event requiring the inoperable steam supply to the turbine-driven EFW pump.

The ACTION for an inoperable steam supply to the turbine-driven EFW pump steam turbine concurrent with both motor-driven EFW pumps being inoperable as proposed requires a unit shutdown be initiated immediately. This change is appropriate due to the seriousness of the condition and is acceptable due to the ability of the EFW system to support the unit shut down.

The ACTION for the EFW system inoperable for reasons other than those described in ACTION (a), (b), or (c) and able to deliver at least 100% flow to either steam generator as proposed will allow a 72 hour completion time. This change is acceptable based on the ability of the system to cool the RCS [Reactor Coolant System] to SDC [Shutdown Cooling] entry conditions following a design basis accident assuming no single active failure.

The ACTION for the EFW system inoperable for reasons other than those described in ACTION (a), (b), or (c) and able to deliver at least 100% combined flow to the steam generators as proposed requires a unit shutdown be initiated immediately. This change is appropriate due to the seriousness of the condition and is acceptable due to the ability of the EFW system to support the unit shut down.

The ACTION for the EFW system inoperable and unable to deliver at least 100% flow to the steam generators as proposed requires immediate action be taken to restore the ability to deliver at least 100% flow to the steam generators. The unit is in a seriously degraded condition in that the EFW system is unable to support a unit shutdown. This change is consistent with the intent of the current EFW Technical Specification and NUREG–1432.

Testing pursuant to Specification 4.0.5 (Inservice Testing Program) as proposed for Surveillance Requirement 'b' will ensure the EFW pumps are tested against the more restrictive of the data points required by either the safety analysis or ASME [American Society of Mechanical Engineers] Section XI.

The remaining changes to the EFW Technical Specification are consistent (other than format) with NUREG–1432 and have therefore been previously approved by the NRC.

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

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

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

NRC Section Chief: Robert A. Gramm.

Exelon Generation Company, LLC, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois.

Date of amendment request: February 28, 2001

Description of amendment request: The proposed amendments would revise the Technical Specifications to eliminate the requirement for at least one person qualified to stand watch to be present in the control room when nuclear fuel is stored in the spent fuel pool.

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

The Defueled Safety Analysis Report (DSAR) identifies three categories of events: spent fuel pool events (i.e., operational occurrences), fuel handling accidents in the fuel building, and radioactive waste handling accidents. There are no active controls in the control room that affect spent fuel pool equipment, or the handling of fuel or radioactive waste. Actions to mitigate the consequences of these events are taken outside the control room. Emergency response is not adversely affected by this proposed change because the control room is still available to the emergency response team and communication capability and timeliness will not be affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The configuration, operation and accident response of the systems, structures or components that support safe storage of the spent fuel are unchanged by the proposed TS change. Current site surveillance requirements ensure frequent and adequate monitoring of system and component functionality. Systems in the Spent Fuel Nuclear Island will continue to be operated in accordance with current design requirements and no new components or system interactions have been identified. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change. The proposed TS change does not have an adverse affect on any system related to safe storage of spent fuel. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

All design basis accident acceptance criteria will continue to be met. The margin of safety relative to the cooling of the spent fuel is unaffected by the proposed change as the SFP [spent fuel pool] parameters will continue to be monitored at the same frequency that they are monitored now. The ability of the shift crew to respond to abnormal or accident conditions is unaffected by the proposed change since all controls are located in the fuel building and any necessary communication will be handled by the DERO [Defueled Emergency Response Organization]. Therefore, it is concluded that the proposed TS change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. Robert Helfrich, Senior Counsel, Nuclear, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, Illinois 60515.

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: May 15, 2001.

Description of amendment request: The proposed amendment would revise refueling operation Technical Specification (TS) requirements for containment equipment hatch cover closure during core alterations and during movement of irradiated fuel both inside containment and in the spent fuel pool or cask pit. The proposed change would allow the containment equipment hatch cover to be off during core alterations and movement of irradiated fuel provided the Emergency Ventilation System is operable with the ability to filter any radioactive release. The proposed changes involve TS 3/4.9.4, Refueling Operations -Containment Penetrations, and TS 3/4.9.12, Refueling Operations—Storage Pool Ventilation, and associated Bases.

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:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. The amendment application proposes to revise DBNPS TS 3/4.9.4, Refueling Operations—Containment Penetrations, and its associated Bases, and TS 3/4.9.12, Refueling Operations—Storage Pool Ventilation, and its associated Bases. The proposed changes would provide for access to the containment through the containment equipment hatch during core alterations and movement of irradiated fuel, provided that an Emergency Ventilation System is operable with the ability to filter any radioactivity release through the containment equipment hatch. The proposed changes would also permit relying on the closing the containment personnel air lock by a designated individual to establish the negative pressure boundary for the Emergency Ventilation System servicing the storage pool. The use of a designated individual to close the containment personnel airlock is currently permitted by TS 3.9.4 for meeting containment closure requirements. Neither the containment equipment hatch nor the Emergency Ventilation System contributes to the initiation of any accident described in the DBNPS Updated Safety Analysis Report.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no equipment, accident conditions, or assumptions are affected which could lead to a significant increase in radiological consequences. The approved analysis for the fuel handling accident inside containment does not take credit for containment closure or Emergency Ventilation System filtering. This analysis results in a maximum calculated offsite does well within the limits of 10 CFR 100.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new or different accident initiators are introduced by these proposed means to mitigate the consequences of an accident.

3. Not involve a significant reduction in a margin of safety because there are no changes to the initial conditions contributing to accident severity or the resulting consequences. Consequently, there are no significant reductions in a margin of safety.

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Dade County, Florida

Date of amendment request: May 14, 2001.

Description of amendment request: The proposed amendments would delete Technical Specifications (TS) Figures 5.1–1, ''Site Area Map,'' and 5.1–2, ''Plant Area Map,'' and would replace TS 5.1, ''Site,'' with a site location description. Conforming changes are requested to delete TS 5.1.1, ''Exclusion Area,'' TS 5.1.2, ''Low Population Zone,'' and TS 5.1.3, ''Map Defining Unrestricted Areas and Site Boundary for Radioactive Gaseous and Liquid Effluents,'' from TS 5.1 and the TS Index. These changes conform to NUREG–1431, Rev. 1, Improved Standard TS for Westinghouse Plants, and the requirements of 10 CFR 50.36 (c)(4).

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

The proposed amendments are administrative in nature, removing sections and maps from the TS, which are located in other documents previously approved by NRC. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect TS that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the TS are administrative in nature and can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature and do not affect operating limits or functional capabilities of plant systems, structures and components. The addition of a site location description to the TS adds geographical information to the TS. Elimination of site and plant area maps from the TS would have no effect on margin of safety as they are located in other controlled plant documents. Thus, the changes proposed would not involve a significant reduction in margin of safety of the facility.

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.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Patrick M. Madden (Acting).

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: February 15, 2001.

Description of amendment request: The proposed amendment to the Cooper Nuclear Station (CNS) Operating License (OL) DPR–46 would (1) delete OL Condition 2.D, Additional Conditions for Protection of the Environment, and (2) remove the depiction of railroad tracks in Technical Specifications (TS) Figure 4.1–1, Site and Exclusion Area Boundaries and Low Population Zone.

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

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

Response: No.

OL [Operating License] Condition 2.D has become obsolete based upon it being satisfied or superceded by amendments to the FSAR [Final Safety Analysis Report] and OL. The previous FSAR and OL amendments which made it obsolete were reviewed and approved based on their individual Unreviewed Safety Question (USQ) evaluations or no significant hazards considerations. Since this proposed change does not physically alter any plant equipment or operating limitations, it therefore does not impact any previously evaluated accident initiator, nor change mitigating systems or features or operating limitations for accidents previously evaluated in the Updated Safety Analysis Report (USAR). Thus, it does not involve a significant increase in the probability or consequences of an accident previously evaluated. This is an administrative change.

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

Response: No.

This proposed change is administrative in nature. It does not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. No setpoints for parameters which initiate protective or mitigative action are being changed. As a result, no new failure modes are being introduced. There are no changes in the procedures or methods governing normal plant operation, nor are the procedures utilized to respond to plant transients altered as a result of this administrative change. This change does not impose any new or different requirements or eliminate any existing requirements. In addition, the change does not alter assumptions made in the safety analysis, nor does it impact the licensing basis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This proposed change is administrative in nature. It does not alter any accident analysis assumptions, conditions, or methodology. Since this proposed change does not physically alter plant systems, structures or components (SSC's), change mitigating systems, features, operating limitations, nor revise accident analysis assumptions, conditions or methodology, it 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.

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: October 19, 2000, as supplemented March 23 and April 9, 2001.

Description of amendment request: The proposed amendment would authorize the licensee to change the licensing basis to utilize the full scope of an alternative radiological source term for accidents as described in NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' and change the Technical Specifications to implement various assumptions in the Alternative Source Term analyses. The portion of this amendment request regarding operability requirements during core alterations and while moving irradiated fuel assemblies within the secondary containment, and which provided for selective application of the Alternative Source Term to the design-basis fuel handling accident was previously evaluated and issued as Amendment No. 237 on April 16, 2001.

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

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

The Alternative Source Term and those plant systems affected by implementing the setpoints and action levels specified in the analyses are not assumed to initiate design basis accidents. The Alternative Source Term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated the new source term is an input to evaluate the consequence. The implementation of the Alternative Source Term has been evaluated in revisions to the analyses of the limiting design bases accidents at DAEC [Duane Arnold Energy Center]. Based on the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the Alternative Source Term. This guidance is presented in NUREG 1465, 10 CFR 50.67, associated Regulatory Guide 1.183, and Standard Review Plan (SRP) Section, 15.0.1. Since secondary containment operability is not assumed for the fuel handling accident (FHA), the consequences of eliminating the requirements for secondary containment operability, secondary containment isolation valves/dampers, secondary containment instrumentation and the Standby Gas Treatment system during fuel movement or core alterations will not increase the effects of a FHA beyond those evaluated in the Alternative Source Term analysis. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

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

The Alternative Source Term and those plant systems affected by implementing the setpoints and action levels specified in the analyses do not initiate design basis accidents. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The changes proposed are associated with the implementation of a new licensing basis for DAEC. Approval of the basis change from the original source term developed in accordance with TID–14844 to a new alternative source term as described in NUREG–1465 is requested by this submittal. The results of the accident analyses revised in support of this submittal, and the requested Technical Specification changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies. Safety margins and analytical conservatisms have been evaluated and are satisfied. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound all postulated event scenarios. The dose consequences of these limiting events are within the acceptance criteria also found in the latest regulatory guidance. This guidance is presented in NUREG 1465, in the approved rulemaking for 10 CFR 50.67, and in the associated Regulatory Guide 1.183.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limit. Specifically, the margin of safety for these accidents is considered to be that provided by meeting the applicable regulatory limit, which, for most events, is conservatively set below the 10 CFR 50.67 limit. With respect to the control room personnel doses, the margin of safety (the difference between the 10 CFR 50.67 limits and the regulatory limit defined by 10 CFR50, Appendix A, Criterion 19 (GDC 19)) continues to be satisfied.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, they are considered to 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.

Attorney for licensee: Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Claudia M. Craig.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: May 30, 2001.

Description of amendment request: The proposed amendment would eliminate local suppression pool temperature limits from the Updated Safety Analysis Report as the basis for limiting suppression pool mechanical loads due to unstable steam condensation during safety relief valve actuations.

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

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

Eliminating the Local Suppression Pool Temperature Limits (LSPTLs) will not introduce new equipment or new equipment methods of operation, and will not alter existing system relationships. LSPTLs are not an accident initiator and does [sic] not affect other accident initiators. The integrity of fission product barriers do not rely on LSPTLs since mechanical loads on containment will not be exceeded and ECCS [emergency core cooling system] operation in the event of an accident will not be adversely affected as demonstrated and approved in Reference 6 [letter from G. Holahan (NRC) to R. Pinelli (Boiling Water Reactor Owners Group), ''Transmittal of the Safety Evaluation of General Electric Co. Topical Reports; NEDO–30832, Entitled 'Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge With Quenchers,' and NEDO–31695, Entitled 'BWR Suppression Pool Temperature Technical Specification Limits','' dated August 29, 1994].

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

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

Eliminating the LSPTLs will not introduce new equipment or new equipment methods of operation, and will not alter existing system relationships. Since containment integrity and ECCS operation will not be challenged, new or different kinds of accidents are not created.

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

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

Since LSPTLs are not required to limit mechanical loads on containment, the margin of safety associated with containment integrity is not significantly reduced. Since LSPTLs are not required to prevent steam binding of the ECCS pumps, the margin of safety associated with ECCS operation is not significantly reduced.

Therefore, the proposed amendment will 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.

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

NRC Section Chief: Claudia M. Craig.

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

Washington County, Nebraska

Date of amendment request: May 15, 2001.

Description of amendment request: The proposed changes would: (1) Replace the titles of Manager—Fort Calhoun Station and the Vice President with generic titles, (2) relocate the requirements for the Plant Review Committee (PRC) and the Safety Audit and Review Committee (SARC) to the Fort Calhoun Station (FCS) Quality Assurance Program, (3) relocate the requirements for procedure controls and records retention to the FCS Quality Assurance Program, (4) enhance and clarify the qualification and training requirements for individuals who perform licensed operator functions, (5) incorporate the Westinghouse/CENP definition of Azimuthal Power Tilt, and (6) eliminate specific mailing address and reporting requirements that are redundant to Title 10 of the Code of Federal Regulations (10 CFR).

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 changes: revise the FCS definition of Azimuthal Power Tilt, remove specific titles from the Technical Specifications, provide minor clarifications of the training requirements for plant staff, and indicate the change in title of the Licensed Senior Operator. This change also relocates the requirements for the Plant Review Committee (PRC) and the Safety Audit and Review Committee (SARC), procedure control, and records retention to the Fort Calhoun Station Quality Assurance Program as described in NRC Administrative Letter 95–06.

The proposed change includes an update to the definition of Azimuthal Power Tilt and adds the bases for the definition of Azimuthal Power Tilt to the bases section of Section 2.10.4 as recommended in ABB Combustion Engineering (CE) Infobulletin Number 97–07, dated December 31, 1997. As noted in the infobulletin, CE discovered a discrepancy in the definition for CE analog plants that use Combustion Engineering Core Operating Report (CECOR) for monitoring and surveillance purposes. Plants that use CECOR should use the same definition as the CE digital plants. This change will make the FCS definition and bases agree with the improved Standard Technical Specifications for CE digital plants, which have previously been approved by the NRC.

The proposed change would allow the use of generic personnel titles as provided in ANSI/ANS 3.1 and NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants,'' in lieu of plant-specific personnel titles. This change does not eliminate any of the qualifications, responsibilities or requirements for these positions, since the plant-specific personnel titles are currently identified in licensee controlled documents such as the Updated Safety Analysis Report (USAR) or the Quality Assurance Program. For example, Section 12 of the Updated Safety Analysis Report describes the management structure and reporting responsibilities of OPPD and provides an organizational chart to determine the corporate officer with responsibility for overall plant nuclear safety from other corporate officers within OPPD. Therefore, changing the terminology within the Technical Specifications, indicating this reporting responsibility does not involve a significant increase in the probability or consequences of an accident previously evaluated. Changing the periodicity of review for staff overtime is also considered an administrative change. This includes a change of the title of the Supervisor—Operations to Manager—Shift Operations, Licensed Senior Operator to Control Room Supervisor, and crewman to crewmember. The change to the number of Senior Operator License present during Core Alterations and the associated note is also considered clarifying in nature and not a change of intent.

The proposed change would update the qualification requirements for the Manager—Radiation Protection, the Shift Technical Advisors, and those individuals that perform the functions described in 10 CFR 50.54(m) to Regulatory Guide 1.8, Revision 3, and ANSI/ANS 3.1–1993. In the March 1987 revision to 10 CFR Part 55, the NRC included the requirement that those facility licensees that have made a commitment that is less than that required by the new rules must conform to the new rules automatically. OPPD had previously considered that commitments made to comply with the requirements of NUREG–0737 and the standards applied through the Institute of Nuclear Power Operations (INPO) accreditation process were equivalent to the guidance provided in Regulatory Guide 1.8, Revision 3. The proposed change provides enhancement to the current requirements and clarifies the qualifications and training requirements for licensed personnel. This provides additional assurance that these personnel are properly trained and qualified for their positions and conforms with the guidance of NRC Regulatory Issues Summary 2001–01. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would relocate specific requirements for SARC, PRC, procedure control, and records retention to the Fort Calhoun Station Quality Assurance Program (Appendix A, of the FCS USAR). This proposed revision does not change or eliminate responsibilities or requirements for these programs. The management level and expertise of personnel who are PRC or SARC members is not being changed. The review of plant operations, procedures control, and record retention is still required to be in compliance with the Fort Calhoun Station Quality Assurance (QA) Program. Any changes in the QA Program which reduce the effectiveness of the program must be approved by the NRC in accordance with 10 CFR 50.54(a)(4). These changes meet the criteria as described in NRC Administrative Letter 95–06. Therefore, the proposed relocation of these programs to the QA Program does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would also remove the requirements prescribing specific submittal addresses, titles, and reporting periods. For example, the requirement to submit License[e] Event Reports within 30 days is replaced with a citation referencing 10 CFR 50.73. This is in agreement with 10 CFR 50.73 and 10 CFR 50.4(f). Additionally, an administrative requirement prescribing the submittal of a Special Maintenance Report is being deleted, as it is redundant to the requirements of 10 CFR 50.73. Therefore, the proposed changes 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.

The proposed changes revise organizational and administrative requirements contained within the Administrative Controls section of the TS. The proposed change to the definition of Azimuthal Power Tilt is as recommended in CE Infobulletin 97–07 for CE analog plants that use CECOR for monitoring and surveillance purposes and will have no affect on accidents previously evaluated. The proposed changes do not revise any equipment setpoints, change the manner in which any plant equipment is operated, or propose any new operating modes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes revise organizational and administrative requirements contained within the Administrative Controls section of the TS. The proposed change to the definition of Azimuthal Power Tilt has no affect on the margin of safety. The proposed changes do not revise any equipment setpoints, change the manner in which any plant equipment is operated, or propose any new operating modes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: May 17, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to permit an increase in the allowable leak rate for the main steam isolation valves (MSIVs) and to delete the MSIV Sealing System (MSIVSS). These changes are based on the use of an alternate source term and the guidance provided in Regulatory Guide 1.183, ''Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.''

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

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

As described in Section 6.7 of the Hope Creek Updated Final Safety Analysis Report (UFSAR), the MSIVSS limits the leakage of fission products through the MSIVs following a design-basis accident large break Loss of Coolant Accident (LOCA). The system is manually actuated following a LOCA. The licensee has proposed to remove the MSIVSS from the plant and to delete the associated requirements from the TSs. In addition, the TSs would be revised to increase the allowable MSIV leak rate. The MSIVSS lines and main steamline drain valves that are connected to the main steam piping will be capped and welded closed to ensure primary containment integrity is maintained. The welding and post-weld examination procedures will be in accordance with the American Society of Mechanical Engineers Code, Section III requirements. The welded caps will be periodically tested as part of the Containment Integrated Leak Rate Test. MSIV leakage and operation of the MSIVSS do not affect the precursors for accidents analyzed in Chapter 15 of the Hope Creek UFSAR. In addition, the proposed changes do not adversely affect other structures, systems, or components important to safety. Therefore, there is no increase in the probability of occurrence of an accident previously evaluated as a result of the proposed changes.

The licensee's submittal states that the radiological consequences associated with the proposed changes have been analyzed based on the results of revised offsite and control room operator dose calculations for a LOCA, which is the most limiting Hope Creek design-basis accident. The current design-basis analysis for the radiological consequences associated with a LOCA is shown in Hope Creek UFSAR Sections 6.4.7 and 15.6.5.5. The revised analysis was performed using an alternate source term in accordance with the requirements in 10 CFR 50.67 and the guidance in Regulatory Guide 1.183. The dose calculations assess the effects of the proposed increase in allowable MSIV leak rate and take no credit for the MSIVSS. In addition, the calculations assume an unfiltered control room inleakage design-basis value that is higher than the current design basis value to address control room habitability issues associated with NEI 99–03. The revised analysis was performed in accordance with the current accepted methodology discussed in Regulatory Guide 1.183 and the radiological consequences were evaluated in terms of Total Effective Dose Equivalent (TEDE) dose as per the acceptance criteria specified in 10 CFR 50.67. The Regulatory Guide 1.183 methodology is not exactly comparable to the current Hope Creek design basis analysis which is in terms of whole body and thyroid doses. The results of the licensee's analysis associated with the proposed changes indicate that the post-LOCA doses will result in an increase in the dose exposures for the control room, the Exclusion Area Boundary (EAB), and the Low Population Zone (LPZ), compared to the current design basis analysis. However, the revised post-LOCA doses will remain below the TEDE dose acceptance criteria for the control room, EAB, and LPZ, as specified in 10 CFR 50.67. The methodology and guidance provided in Regulatory Guide 1.183 has been developed for the purpose of performing design basis radiological consequence analyses using an alternate source term such that meeting the 10 CFR 50.67 acceptance criteria demonstrates adequate protection of public health and safety. Therefore, the proposed changes do not involve a significant increase in the consequences of any accident previously evaluated.

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

The proposed change to increase the allowed MSIV leakage rate does not affect the operability the MSIVs and will not inhibit the capability of the MSIVs to perform their function of isolating the primary containment as assumed in the Hope Creek accident analyses in UFSAR Chapter 15. The proposed change to delete the MSIVSS does not introduce any new modes of plant operation and, as previously discussed, the design-basis LOCA analysis was reanalyzed without taking credit for the operation of MSIVSS. The affected main steam piping will be welded and/or capped closed to assure that the primary containment integrity, isolation, and leak testing capability are not compromised. Based on the above considerations, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

As previously discussed, the results of the licensee's analysis associated with the proposed changes indicate that the post-LOCA doses will result in an increase in the dose exposures for the control room, the EAB, and the LPZ, compared to the current design basis analysis. Since there will be an increase in dose exposure, the margin of safety will be decreased. However, the revised post-LOCA doses will remain below the TEDE dose acceptance criteria for the control room, EAB, and LPZ, as specified in 10 CFR 50.67. Meeting the 10 CFR 50.67 acceptance criteria demonstrates adequate protection of public health and safety. An acceptable margin of safety is inherent in these acceptance criteria. Therefore, there is no significant reduction in the margin of safety as a result of the proposed changes.

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.

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

NRC Section Chief: James W. Clifford.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: March 5, 2001.

Description of amendment request: The proposed amendment would (1) change the Security Plan provision that a member of the security force escort all vehicles, other than designated licensee vehicles, and to delete the related Security Training and Qualification Plan task, (2) change the requirement of the Security Plan that all areas of the protected area be illuminated to a minimum of 0.2 footcandle, and (3) change the frequency of protected area patrols in the Security Plan.

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

1. Do the proposed changes involve a significant increase in the probability or

consequences of an accident previously evaluated?

The proposed changes involving security activities do not reduce the ability for the security organization to prevent radiological sabotage and therefore do not increase the probability or consequences of a radiological release previously evaluated.

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

The proposed changes involve functions of the security organization concerning vehicle control, protected area illumination, and protected area patrol frequency. Analysis of the proposed changes has not indicated nor identified a new or different kind of accident from any previously evaluated.

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

Analysis of the proposed changes show that they affect only the functions of the Security organization and have no impact upon nor cause a significant reduction in margin of safety for plant operation. The failure points of key safety parameters are not affected.

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.

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

NRC Section Chief: James W. Clifford.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: May 30, 2001 (ULNRC–04481).

Description of amendment request: The proposed amendment changes the technical specifications to remove the phrase ''and the charging flow control valve full open'' from Limiting Condition for Operation 3.5.5, Required Action A.1, and Surveillance Requirement 3.5.5.1 for the reactor coolant pump seal injection flow.

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 emergency core cooling system (ECCS) analysis models the reactor coolant pump (RCP) seal injection flow path as a hydraulic flow resistance. The proposed change clarifies that RCP seal injection flow is a function of system conditions. The seal injection flow rate can vary during operation, but the hydraulic flow resistance is fixed by positioning the manual seal injection throttle valves. The resistance does not change if the valve adjustments are not changed. Thus, RCP seal injection flow variation due to changing reactor coolant system (RCS) backpressure following a loss of coolant accident (LOCA) is explicitly accounted for as a result of modeling the RCP seal injection flow path resistance.

The proposed change does not impact the way the RCP seal injection flow should be established per the safety analysis and does not affect RCP seal integrity. The seal injection flow resistance only affects ECCS flow. Since ECCS flow occurs after an accident, the proposed change cannot impact the probability of an accident.

Overall ECCS performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The ECCS will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the proposed change are [still] maintained.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [Final Safety Analysis Report].

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The proposed change will not affect the normal method of plant operation. No performance requirements will be affected.

Since the proposed change continues to assure that the assumed ECCS flow is available after a large break LOCA, no new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result [of the proposed change]. There will be no adverse effect or challenges imposed on any safety-related system as a result of this request.

The proposed change does not alter the design or performance characteristics of the ECCS. It simply corrects the description of how to properly set the position of the RCP seal injection throttle valves in support of the ECCS flow balance assumptions.

Therefore, the proposed 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.

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. There will be no impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor (F_Q) nuclear enthalpy rise hot channel factor (FN/DH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: April 11, 2000, as supplemented by letters dated August 28, 2000, November 20, 2000, and April 11, 2001.

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) 3.7, 3.10, and 3.22, as well as the Bases of TS 3.4, 3.8, 3.10, 3.19, and 3.22. The proposed changes would implement an alternate accident source term methodology previously approved by NRC. Implementation of the alternate source term could permit a number of plant changes that have been proposed, including: Permitting a slight atmospheric pressure in containment for a short time following a loss-of-coolant accident (LOCA), deletion of automatic function requirements and setpoints for containment particulate and gas monitors, deletion of the requirement to filter fuel building and containment purge exhaust during refueling, and a number of other related operational and configuration requirements.

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

The proposed TS changes allow relaxation of containment integrity requirements during refueling operations by allowing the personnel airlock, equipment access hatch and certain penetrations to remain open during fuel movement in containment. The changes also eliminate the requirement to filter the exhaust from containment or the fuel building during refueling operations. Also proposed is a relaxation of the current containment design basis acceptance criteria to allow an interval of four hours following the design basis LOCA until containment is depressurized to subatmospheric conditions. We have reviewed the proposed TS changes relative to the requirements of 10 CFR 50.92 and determined that a significant hazards consideration is not involved. Specifically, operation of Surry Power Station with the proposed changes will not:

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

The probability remains unaffected since the accident analyses involve no change to a system, component or structure that affects initiating events for any of the accidents evaluated. The consequences of the reanalyzed events is expressed in terms of the TEDE [total effective dose equivalent] dose, which is not directly comparable to either the thyroid or whole body doses reported in existing analyses. However, even taking this comparison into consideration, any dose increase is not significant. Furthermore, the revised analysis results meet the applicable TEDE dose acceptance criteria for alternative source term implementation.

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

The implementation of the proposed changes does not create the possibility of an accident of a different type than was previously evaluated in the SAR [Safety Analysis Report]. The proposed Technical Specifications changes allow relaxation of these current requirements: (1) maintaining subatmospheric containment conditions following a LOCA; (2) filtration of containment & fuel building exhaust during fuel movement; (3) maintaining the personnel airlock, equipment access hatch & penetrations closed during fuel movement and (4) operability of containment purge isolation during refueling. These changes do not alter the nature of events postulated in the UFSAR [Updated Final SAR] nor do they introduce any unique precursor mechanisms. Therefore, there is no possibility for accidents of a different type than previously evaluated.

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

The implementation of the proposed changes does not reduce the margin of safety. The radiological analysis results, even though compared with the revised TEDE acceptance criteria, meet the applicable limits. These criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting the stated limits demonstrates adequate protection of public health and safety. It is thus concluded that the margin of safety will not be reduced by the implementation of the changes.

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

Attorney for licensee: Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch.

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to pdr@nrc.gov.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: November 30, 2000.

Brief description of amendments: The amendments revise TS 5.5.13, ''Diesel Fuel Oil Testing Program,'' to relocate the specific American Society for Testing and Materials (ASTM) Standard reference from the Administrative Controls Section of TS to a licensee-controlled document, i.e., the Diesel Fuel Oil Program in the Technical Requirements Manual (TRM). In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil in lieu of the ''clear and bright'' test.

Date of issuance: June 13, 2001.

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

Amendment Nos.: 122, 122, 116, and 116.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 21, 2001.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 31, 2000.

Brief description of amendments: Revised Technical Specification (TS) Surveillance Requirement (SR) 4.5.1.d.1, concerning the operability of the Automatic Depressurization System, and relocated the existing requirements in TS SR 4.5.1.d.1 and TS SR 4.5.1.d.2.c to the Technical Requirements Manual.

Date of issuance: June 12, 2001.

Effective date: As of date of issuance and shall be implemented within 30 days.

Amendment Nos.: 152 and 116.

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

Date of initial notice in **Federal Register:** October 18, 2000 (65 FR 62389).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 12, 2001.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Beaver County, Pennsylvania

Date of application for amendment: November 8, 2000, as supplemented on February 6, and May 7, 2001.

Brief description of amendment: The amendment changed the technical specifications associated with the deletion of TS 3/4.4.1.6, ''Reactor Coolant Pump—Startup.''

Date of issuance: June 13, 2001.

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

Amendment No: 238.

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

Date of initial notice in **Federal Register:** December 27, 2000 (65 FR 81917).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 2001.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of application for amendment: January 19, 2001, as supplemented April 20 and May 9, 2001.

Brief description of amendment: The amendment would change the TSs to extend surveillance intervals associated with the emergency diesel generator (EDG) engines and station batteries that are currently required to be completed beginning June 27, 2001. The license amendment would allow these requirements to be performed during the next refueling outage, but no later than December 31, 2001. This would preclude the need for a mid-cycle shutdown of the Unit.

Date of issuance: June 11, 2001.

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

Amendment No.: 234.

Facility Operating License No. DPR–74: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** March 21, 2001 (66 FR 15926). The April 20 and May 9, 2001, supplemental letters, did not change the scope of the proposed action and did not change the Nuclear Regulatory Commission's (NRC's) preliminary no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 11, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: January 13, 2000, as supplemented March 7, March 30, and May 4, 2001.

Brief description of amendment: The amendment revises the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs) 3.6, ''Containment'' to add Limiting Condition for Operation (LCO) and Allowed Outage Times (AOT) for containment isolation devices. In addition, the amendment provides additional information, clarification, and uniformity to the bases of the associated TSs.

Date of issuance: June 8, 2001.

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

Amendment No.: 155.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 21, 2001 (66 FR 11061). The March 7, March 30, and May 4, 2001, letters, provided clarifying information that was within the scope of the original application, did not change the NRC staff's initial proposed no significant hazards consideration determination, and did not expand the amendment beyond the scope of the original notice (66 FR 11061).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 8, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: January 26, 2001, as supplemented by letter dated March 13, 2001.

Brief description of amendment: The amendment changes Technical Specification Surveillance Requirement 3.7.9.2, ''Ultimate Heat Sink (UHS),'' by increasing the maximum allowable temperature of Lake Michigan water from 81.5 °F to 85 °F.

Date of issuance: June 4, 2001.

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

Amendment No.: 202.

Facility Operating License No. DPR–20. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 7, 2001 (66 FR 13800).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 4, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 3, 2000, as supplemented by letters dated November 17, 2000, and February 14, 2001.

Brief description of amendment: The amendment deletes Section 3.D, ''License Term,'' from the Fort Calhoun Station, Unit No. 1 operating license.

Date of issuance: June 6, 2001.

Effective date: June 6, 2001, to be implemented within 30 days from the date of issuance.

Amendment No.: 199.

Facility Operating License No. DPR–40: The amendment revised the operating license.

Date of initial notice in **Federal Register:** January 10, 2001 (66 FR 2019).

The November 17, 2000, and February 14, 2001, supplemental letters provided clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 6, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 6, 2001 and supplemented by letter dated April 20, 2001.

Brief description of amendments: The amendments proposed to revise the San Onofre Nuclear Generating Station, Units 2 and 3 Technical Specification Surveillance Requirements 3.8.1.2, 3.8.1.3, 3.8.1.9, 3.8.1.10, and 3.8.1.19 to assure that an emergency diesel generator automatic voltage regulator (AVR) is operable and regularly tested. AVR operability would be demonstrated by conducting SR 3.8.1.2 and 3.8.1.3 within the past 60 days, and any one of SR 3.8.1.9, 3.8.1.10, or 3.8.1.19 within the past 24 months.

Date of issuance: June 8, 2001.

Effective date: June 8, 2001, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2–179; Unit 3–170.

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

Date of initial notice in **Federal Register:** May 2, 2001 (66 FR 22032).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 8, 2001.

No significant hazards consideration comments received: No.

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

Date of amendments request: August 17, 2000, as supplemented by letter dated April 2, 2001. The April 2, 2001, letter requested a new implementation date, but did not change the August 17, 2000, application and the initial proposed no significant hazards consideration determination.

Brief description of amendments: The amendments eliminate the need for the licensee to perform periodic response time testing of selected reactor trip system and engineered safety feature actuation system equipment as defined in Westinghouse report WCAP–14036–P–A, Revision 1, ''Elimination of Periodic Protection Channel Response Time Tests.''

Date of issuance: June 7, 2001.

Effective date: As of the date of issuance and shall be implemented on Unit 1 entry in Mode 3 for Cycle 18 following the 2001 fall refueling.

Amendment Nos.: 149 and 141.

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

Date of initial notice in **Federal Register:** January 10, 2001 (66 FR 2023). The supplement dated April 2, 2001, provided clarifying information that did not change the scope of the August 17, 2001, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 7, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 11, 2001.

Brief description of amendments: The amendments revise TS 5.5.17, ''Containment Leakage Rate Testing Program,'' to add an exception to Regulatory Guide 1.163 related to visual examination of containment concrete surfaces.

Date of issuance: June 6, 2001.

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

Amendment Nos.: 122 and 100.

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

Date of initial notice in **Federal Register:** May 2, 2001 (66 FR 22033).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 6, 2000.

Description of amendment request: These amendments revised the Technical Specifications (TS) to allow four residual heat removal suppression pool cooling subsystems to be inoperable for 8 hours.

Date of issuance: June 8, 2001.

Effective date: June 8, 2001.

Amendment Nos.: 241, 272, and 230.

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

Date of initial notice in **Federal Register:** November 29, 2000 (65 FR 71139).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 8, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 3, 2001.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.3.6, ''Containment Ventilation Isolation Instrumentation,'' to modify the Note for Required Action B.1 such that it applies only to * * * Required Action and associated Completion Time of Condition A not met * * * This change is the result of the discovery of an error which occurred when the TSs were converted to the improved TS with issuance of License Amendment Nos. 64 and 64, for Comanche Peak Steam Electric Station, Units 1 and 2, on February 26, 1999.

Date of issuance: June 4, 2001.

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

Amendment Nos.: 86 and 86.

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

Date of initial notice in **Federal Register:** May 2, 2001 (66 FR 22034).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 4, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: September 27, 2000, as supplemented November 21 and December 18, 2000, and February 2, March 2, and May 21, 2001.

Brief description of amendment: These amendments add Technical Specification (TS) 3.7.14, TS 4.7.14, TS 3.7.15, TS 4.7.15, Figure 3.7.15–1, and Figure 3.7.15–2; and revise TS 5.3.1 and TS 5.6.1.1. The purpose of these amendments is to increase the limit on the fuel enrichment from the current limit of 4.3 weight percent U^{235} to a maximum of 4.6 weight percent U^{235}, establish TS Limiting Conditions for Operations for the Spent Fuel Pool (SFP) boron concentration and fuel storage restrictions, and eliminate the value of uncertainties in the calculation for K_{eff} in the SFP criticality calculation.

Date of issuance: June 15, 2001.

Effective date: As of the date of issuance and shall be implemented by December 21, 2001.

Amendment Nos.: 227 and 208.

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

Date of initial notice in **Federal Register:** December 13, 2000 (65 FR 77929). The December 18, 2000, February 2, March 2, and May 21, 2001, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination, or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 15, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 19th day of June 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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NUCLEAR REGULATORY COMMISSION

Preliminary Impact Assessment of Nuclear Industry Consolidation onNRC Oversight: Request for Comments

AGENCY: Nuclear Regulatory Commission (NRC).

ACTION: Request for comments.

SUMMARY: Economic deregulation of the electric utility industry has resulted in consolidation and restructuring of the nuclear power industry. The transformation of the once strictly regulated industry has led to separation of the generation, transmission and distribution sectors, corporate mergers and asset transfers, acquisitions by outright purchase, and a general transition to a nationwide competitive market. There have also been numerous nuclear power plant license transfer applications, which the NRC staff must review and approve before a license can be transferred to a new entity.

The NRC staff has identified and performed a preliminary assessment of the impacts of nuclear industry consolidation on the NRC and whether the NRC needs to change its regulations, policies, processes, guidance, or organizational structure to continue to meet its strategic public health and safety goals. The initial object of this effort is to identify impacts that need to be considered further.

The NRC staff has identified a number of consolidation and a few deregulation-related impacts on NRC oversight of the nuclear industry, grouped them by category, and performed preliminary impact assessments. The individual assessments follow this notice.

The NRC staff requests comments and suggestions from stakeholders on the identified issues and the preliminary impact assessments. The NRC staff will consider all comments received. A public workshop will be held at NRC Headquarters in the October/November 2001 timeframe to discuss the regulatory oversight issues attendant to industry consolidation, the staff's preliminary impact assessments, and the comments received from the stakeholders. Notice of this workshop will be published at a later date. Commenters should indicate their interest in attending and participating in this workshop.

The product of this effort will be staff recommendations of impacts that the Commission needs to consider further.

DATES: The comment period ends August 27, 2001. Comments received after this date will be considered if it is practical to do so, but the staff guarantees consideration only of comments received on or before this date.

ADDRESSES: Mail written comments to Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Comments may also be sent by completing the online comment form at *http://www.nrc.gov/NRC/REACTOR/CONSOLIMPACT/index.html.*

Deliver comments to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

For further information contact Herbert N. Berkow, Mail Stop O 8 H–12, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555; telephone (301) 415–1485 and e-mail at HNB@NRC.GOV.

Dated at Rockville, Maryland, this 20th day of June 2001.

For the Nuclear Regulatory Commission.

Herbert N. Berkow,

Director, Project Directorate II, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

Industry Consolidation Preliminary Impact Assessments

Categorization of Industry Consolidation Issues

Category 1 Plant Operational Safety

Issue 1.a Possible Cost-cutting Initiatives
Issue 1.b Technology-related Issues
Issue 1.c Spent Fuel Storage and Transportation
Issue 1.d Low-Level Radioactive Waste Management
Issue 1.e Emergency Preparedness
Issue 1.f Reliable Off-site Power

Category 2 Licensing

Issue 2.a License Transfer Process
Issue 2.b New License Applications, Site Approvals, and Reactivations of Deferred Plants
Issue 2.c License Renewal
Issue 2.d NRC Organizational Structure

Category 3 Inspection, Enforcement, and Assessment

Issue 3.a NRC Reactor Oversight Process
Issue 3.b Other NRC Inspection Programs
Issue 3.c NRC Enforcement Program
Issue 3.d NRC Allegation Program

Category 4 Decommissioning

Category 5 External Regulatory Interfaces

Category 6 Fuel Cycle Facilities

Category 7 Financial

Issue 7.a Foreign Ownership
Issue 7.b License Fee Structure
Issue 7.c Insurance
Issue 7.d Joint and Several Regulatory Responsibility
Issue 7.e Bankruptcy Protection
Issue 7.f Financial Qualifications

Category 8 Non-NRC Regulatory Considerations

Issue 8.a Grid Stability/Reliability
Issue 8.b Antitrust Considerations

Issue Category: 1. Plant Operational Safety

Issue: 1.a Possible Cost-Cutting Initiatives

Discussion

In a more consolidated, economically deregulated market, the nuclear power industry will be faced with new pressures to operate more efficiently. Cost controls could result in shorter outages (and thus longer run times), increased use of on-line maintenance, power uprate amendments, increased use of risk-informed technology and decisions and other changes that would result in lower costs and increased productivity.