controversy, preceded by discovery under the Commission's rules and the designation, following argument of only those factual issues that involve a genuine and substantial dispute, together with any remaining questions of law, to be resolved in an adjudicatory hearing. Actual adjudicatory hearings are to be held on only those issues found to meet the criteria of section 134 and set for hearing after oral argument.

The Commission's rules implementing section 134 of the NWPA are found in 10 CFR Part 2, Subpart K, ''Hybrid Hearing Procedures for Expansion of Spent Fuel Storage Capacity at Civilian Nuclear Power Reactors'' (published at 50 FR 41662 dated October 15, 1985). Under those rules, any party to the proceeding may invoke the hybrid hearing procedures by filing with the presiding officer a written request for oral argument under 10 CFR 2.1109. To be timely, the request must be filed within ten (10) days of an order granting a request for hearing or petition to intervene. The presiding officer must grant a timely request for oral argument. The presiding officer may grant an untimely request for oral argument only upon a showing of good cause by the requesting party for the failure to file on time and after providing the other parties an opportunity to respond to the untimely request. If the presiding officer grants a request for oral argument, any hearing held on the application must be conducted in accordance with the hybrid hearing procedures. In essence, those procedures limit the time available for discovery and require that an oral argument be held to determine whether any contentions must be resolved in an adjudicatory hearing. If no party to the proceeding timely requests oral argument, and if all untimely requests for oral argument are denied, then the usual procedures in 10 CFR Part 2, Subpart G apply.

For further details with respect to this action, see the application for amendment dated July 24, 2001, which is available for public inspection at the Commission's PDR, 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 System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 21st day of June 2002.

For the Nuclear Regulatory Commission.

Karen R. Cotton,

Project Manager, Section 1, Project Directorate II, Division of Licensing Project Management.

[FR Doc. 02–16097 Filed 6–24–02; 8:45 am]

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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 May 31, 2002, through June 13, 2002. The last biweekly notice was published on June 11, 2002 (67 FR 40019).

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 Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By July 25, 2002, 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,[1] which is available at the Commission's PDR, 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 System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* 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 Staff, or may be delivered to the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent 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 PDR, 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 System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

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[1] The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.741(d) and subparagraphs (d)(1) and (2), regarding petitions to intervene and contentions. Those provisions are extant and still applicable to petitions to intervene. Those provisions are as follows: ''In all other circumstances, such ruling body or officer shall, in ruling on—

(1) A petition for leave to intervene or a request for hearing, consider the following factors, among other things:

(i) The nature of the petitioner's right under the Act to be made a party to the proceeding.

(ii) The nature and extent of the petitioner's property, financial, or other interest in the proceeding.

(iii) The possible effect of any order that may be entered in the proceeding on the petitioner's interest.

(2) The admissibility of a contention, refuse to admit a contention if:

(i) The contention and supporting material fail to satisfy the requirements of paragraph (b)(2) of this section; or

(ii) The contention, if proven, would be of no consequence in the proceeding because it would not entitle petitioner to relief.''

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendments request: May 15, 2002 (102–04701).

Description of amendments request: The amendments would revise Limiting Condition for Operation (LCO) 3.9.3, ''Containment Penetrations,'' of the Technical Specifications. The amendments would (1) modify LCO 3.9.3.b on one door in each air lock being closed and (2) add a note to LCO 3.9.3 about containment penetration flow paths providing direct access from the containment to the outside atmosphere may be unisolated under administrative controls. The amendments would allow the containment air lock and other penetrations to be open during core alterations or movement of irradiated fuel assemblies within containment.

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 amendment[s] to Technical Specification (TS) 3.9.3[,] ''Containment Penetrations,'' would allow the personnel air locks and other containment penetrations to remain open during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The position of the personnel air locks and other containment penetrations (open or closed) are not an initiator of any accident.

The fuel handling accident contained in the Updated Final Safety Analysis Report [for Palo Verde], Revision 11[,] assumes that the personnel air locks, containment penetrations, and the equipment hatch are open and the entire airborne radioactivity reaching the containment [from the damaged fuel] is released to the outside environment. Using these assumptions, the current analysis results in off site doses that are well within guideline values specified in 10 CFR [Part] 100[,] ''[R]eactor Site Criteria[,]'' and calculated control room doses within the acceptance criteria specified in General Design Criteria 19[,] ''Control Room.''

Therefore, the proposed amendment request to allow the personnel air locks and [other] containment penetrations to be open during CORE ALTERATIONS [or] movement of irradiated fuel assemblies in containment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment[s] to TS 3.9.3[,] ''Containment Penetrations,'' allowing the personnel air locks and other containment penetrations to be open during CORE ALTERATIONS [or] movement of irradiated fuel in containment does not involve a physical alteration of the plant (no new or different type of equipment will be installed). It does[,] however, involve a minor change in the methods governing normal plant operation during refueling. This minor change in personnel air lock and containment penetration control does not create the possibility of a new or different kind of accident. [Containment penetration control is not an initiator of an accident.] The fuel handling accident [(FHA)] analysis contained in the Updated Final Safety Analysis Report, Revision 11[,] already assumes that the personnel air locks, [other] containment penetrations, and the equipment hatch are open and the entire airborne radioactivity released in containment following a FHA is transported to the outside environment. This analysis results in off site doses that are well within guideline values specified in 10 CFR [Part] 100[,] ''Reactor Site Criteria[,]'' and calculated control room doses within the acceptance criteria specified in General Design Criteria 19[,] ''Control Room.''

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

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

The proposed amendment[s] to TS 3.9.3[,] ''Containment Penetrations,'' allowing the personnel air locks and other containment penetrations to be open during CORE ALTERATIONS [or] movement of irradiated fuel in containment remains bounded by previously determined radiological dose consequences for a FHA inside containment. The previously analyzed dose consequences assumes that the personnel air locks, containment penetrations, and the equipment hatch are open and the entire airborne radioactivity released in containment is transported to the outside environment. The results of this analysis were determined to be within the limits of 10 CFR [Part] 100[,] ''Reactor Site Criteria[,]'' and [* * *] meets the acceptance criteria of NUREG–0800[, ''] Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants[,''] Section 15.7.4[,] ''Radiological Consequences of Fuel Handling Accidents.'' The calculated control room doses are within the acceptance criteria specified in General Design Criteria 19[,] ''Control Room.'' There are no changes in the assumptions made about the positions of the containment openings and penetrations. Therefore, there is no change in the analysis results and the proposed amendment request does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) 5.5.3, ''Post Accident Sampling System (PASS),'' and thereby eliminate the requirements to have and maintain the PASS at Fermi 2. The changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated May 23, 2002.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the [Three Mile Island, Unit 2] TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes within the containment building. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS. Therefore, this change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment would delete Section 2.F of the Operating License which requires reporting violations of the requirements in Section 2.C of the Operating License. The licensee stated that the requirements in Section 2.F are adequately addressed by the reporting requirements identified in 10 CFR 50.72 and 10 CFR 50.73, and therefore, Section 2.F is not required. The proposed amendment would also delete License Conditions 2.C.(19), 2.C.(20) and 2.C.(21), which pertain to historical actions that have been met.

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

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

This License Amendment request involves administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed changes. The three License Conditions proposed for deletion pertain to actions that have been completed and are no longer applicable. The reporting requirements in Section 2.F of the Operating License are not required because they are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or contained in the specific License Condition (2.C.(10)). 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 have no impact on the design, function or operation of any plant structure, system or component. The changes are administrative in nature and do not affect plant equipment or accident analyses. License Conditions 2.C.(19), 2.C.(20) and 2.C.(21) can be deleted because they are no longer applicable. The reporting requirements in the Fermi 2 Operating License can be deleted because they are either adequately addressed in 10 CFR 50.72 and 10 CFR 50.73, or are included in the specific License Condition (2.C.(10)). Therefore, these changes cannot create a new failure mode, nor can they create the possibility of a new or different kind of accident than any accident previously evaluated.

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

The proposed changes do not relax the bases for any limiting condition of operation nor do they affect the design or operation of any fission product barrier. The changes are administrative in nature and result in the deletion of obsolete License Conditions and reporting requirements that are adequately addressed elsewhere. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan, Section Chief.

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

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment would change the Fermi 2 Technical Specifications (TSs) to allow a one-time deferral of the Type A primary containment integrated leak rate test. Specifically, TS 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' would be revised to extend the current interval for performing the containment Type A test to 15 years.

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 License Amendment involves a one-time extension of the testing frequency for the primary containment 10 CFR [Part] 50, Appendix J, Type A test. The current 10-year test interval would be extended on a one-time basis to no longer than 15 years. The proposed Technical Specification (TS) change does not involve a physical plant change or a change in the manner in which the plant is operated or controlled. The primary containment is designed to provide an essentially leak tight barrier against an uncontrolled release of radioactivity to the environment resulting from postulated design basis accidents. As such, the primary containment and the testing requirements do not affect accident initiation; therefore, the proposed TS change does not involve a significant increase in the probability of an accident previously evaluated.

Type B and C containment local leak rate testing will continue to be performed at the frequency required by the TS. As documented in NUREG–1493, ''Performance-Based Containment Leakage Test Program,'' industry experience has shown that Type B and C tests have identified about 97 percent of containment leakage paths, and only about 3 percent have been detected by a Type A test. NUREG–1493 also concluded, in part, that reducing the frequency of Type A containment leakage rate test to once per 20 years would result in an imperceptible increase in risk. The Fermi 2 risk-based assessment of the proposed extension supports this conclusion. The design and construction of the primary containment, combined with the containment inspection program in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, and the Maintenance Rule program per 10 CFR 50.65 requirements, provide a high degree of confidence that the containment will not degrade in a manner that is detectable only by Type A testing. Additionally, the inherent feature of Boiling Water Reactor containments which provides on-line containment monitoring capability, allows for early detection of gross containment leakage during power operation.

Based on the above, the proposed change does not significantly increase the probability or consequences of any accident previously evaluated.

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

The primary containment is designed to contain energy and fission products during and following design basis accidents. The containment and testing requirements, invoked to periodically demonstrate the integrity of the containment, ensure the plant's ability to mitigate the consequences of an accident; however, the containment and testing do not involve accident initiation. In addition, the proposed change to the Type A test frequency does not involve a physical change to the facility. The change does not affect the operation of the plant such that a new failure mode involving the possibility of a new or different kind of accident is created. Therefore, the proposed change does not create the potential for a new or different kind of accident from any accident previously evaluated.

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

The NUREG–1493 generic study on the effects of extending containment leakage testing found that reducing the Type A test frequency to once per 20 years resulted in an imperceptible increase in risk to the public. The NUREG study concluded that Type B and C testing detect most potential containment leakage. The extension of [the] Type A test interval to 15 years has a minimal effect on leakage detection capability. The TS allowed leakage limit is not impacted by this change, and the frequency of local Type B and C testing will not be altered as a result of this extension. Additionally, the containment inspection program provides a high degree of assurance that the containment will not degrade in a manner only detectable by Type A testing. On-line containment monitoring provides additional assurance for detecting gross containment leakage during power operation. The combination of all these factors ensures that the safety margin will be maintained. Therefore, the proposed changes will not result in 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to eliminate the response time testing requirements for certain instrumentations in TS 3.3.1.1 and TS 3.6.1.1, based on NRC-approved licensing topical report, NEDO–32291-A, ''System Analyses for Elimination of Selected Response Time Testing Requirements,'' dated October 1995, and its Supplement 1, dated October 1999. This licensing topical report shows that other periodic tests required by TSs, such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, provide adequate assurance that instrument response times are within acceptance limits. Therefore, the proposed change to delete the specific response time testing requirements does not change the response time assumptions in the Updated Safety Analysis Report. Only the methodology of time response verification would be changed.

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

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

Response: No.

The proposed amendment to the Technical Specifications does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The same Reactor Protection System (RPS) and Primary Containment Isolation Instrumentation instrumentation [sic] is used, and the response time assumptions in [the] Updated Final Safety Analysis Report (UFSAR) Chapter 15 analysis remain unchanged. Only the methodology of time response verification is changed. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. Therefore, the proposed amendment does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This change does not alter the performance of the Reactor Protection System (RPS) or Primary Containment Isolation Instrumentation systems. All RPS and Primary Containment Isolation Instrumentation channels will still have an initial response time verified by test before initially placing the channel in operational service and after any maintenance that could affect response time. Changing the method of periodically verifying instrument response for certain RPS and Primary Containment Isolation Instrumentation channels (assuring equipment operability) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the channel characteristic. Implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Implementation of NEDO 32291–A methodologies for eliminating selected response time testing does not involve a significant reduction in the margin of safety. The current response time limits are based on the maximum values assumed in the plant safety analyses. The analyses conservatively time testing does not affect the capability of the associated systems to establish the margin of safety. The elimination of the selected response perform their intended function within the allowed response time used as the basis for plant safety analyses. Plant and system response to an initiating event will remain in compliance within the assumptions of the safety analyses, and therefore, the margin of safety is not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan, Section Chief.

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

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to revise the requirements for system operability during movement of recently irradiated fuel assemblies in the secondary containment. Specifically, the Applicability of TS 3.3.7.1, ''Control Room Emergency Filtration (CREF) System Instrumentation,'' 3.7.3, ''Control Room Emergency Filtration (CREF) System,'' and 3.7.4, ''Control Center Air Conditioning (AC) System,'' during movement of recently irradiated fuel assemblies would be deleted.

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

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

This License Amendment involves changes in the requirements for the operability of the CREF system, CREF system instrumentation, and Control Center Air Conditioning (AC) system. The functions of these systems provide configurations for mitigating the consequences of radiological accidents; however, they do not involve the initiation of any previously analyzed accident. Therefore, the proposed changes cannot increase the probability of any previously evaluated accident.

The analysis of the Fuel Handling Accident (FHA) concludes that radiological consequences are within the regulatory acceptance criteria. The FHA analysis includes evaluations of the radiological consequences resulting from a limiting drop of a fuel assembly, using the Alternative Source Term (AST) and the Regulatory Guide 1.25 methodologies, over the reactor core. The radiological consequences associated with this scenario, assuming no mitigation credit for the CREF System, have been shown to satisfy the regulatory acceptance criteria. Therefore, the proposed changes do not significantly increase the radiological consequences of any previously evaluated accident.

Based on the above, the proposed changes do not significantly increase the probability or consequences of any 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 do not alter the design function or operation of the systems involved. The CREF system will still provide protection to control room occupants in the case of a significant radioactive release. The revised Technical Specification (TS) requirements are supported by the FHA analysis. The radiological consequences of a FHA under the proposed TS requirements are well below the regulatory limits. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. The original Licensing Basis for the FHA took no credit for CREF system mitigation. Therefore, the proposed changes do not create the potential for a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to the Fermi 2 TS requirements are supported by the design basis analysis and are established such that the radiological consequences are below the regulatory guidelines. Safety margins and analytical conservatisms are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed TS requirements continue to ensure that the radiological consequences at both the control room and the exclusion area and low population zone boundaries are below the corresponding regulatory guidelines; therefore, the proposed changes will not result in 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan, Section Chief.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: May 7, 2002.

Description of amendment request: The proposed amendment would change Technical Specifications (TSs) 2.2, ''Limiting Safety System Settings'' and 3/4.3, ''Instrumentation'' to more accurately reflect the existing plant design for the Reactor Protection System, the Engineered Safety Features Actuation System, and the Radiation Monitoring System instrumentation and to provide consistency within TS Tables 2.2–1, 3.3–1, and 4.3–1. Specifically, the proposed amendment would make the following changes:

(1) The Reactor Coolant Pump Speed—low functional unit, also known as the Underspeed—Reactor Coolant Pumps functional unit, which is not credited by the facility accident analysis, would be deleted from the TSs.

(2) The mode applicability for the Wide Range Logarithmic Neutron Flux Monitor functional unit would be revised consistent with a previously approved license amendment (Millstone Unit No. 2 License Amendment No. 38, dated April 19, 1978).

(3) The Safety Limits And Limiting Safety System Settings TS would be revised for completeness and consistency with the Reactor Protection System Instrumentation TS to include those functional units which do not have specific trip or allowable values.

(4) The Reactor Protection System Instrumentation TS would be revised to include operability requirements for the Reactor Protection System Logic functional unit.

(5) The Reactor Protection System Instrumentation TS would be revised to

include operability requirements for the Reactor Trip Breakers functional unit.

(6) The Engineered Safety Feature Actuation System Instrumentation TS would be revised to include operability, trip setpoint, and surveillance requirements for the Automatic Actuation Logic, as applicable, associated with the Safety Injection, Containment Spray, Containment Isolation, Main Steam Isolation, Enclosure Building Filtration, Containment Sump Recirculation, Loss of Power, and Auxiliary Feedwater functional units.

(7) The Engineered Safety Feature Actuation System Instrumentation TS action statement for the Auxiliary Feedwater manual actuation functional unit would be revised such that the required actions are consistent with the applicability of the TS.

(8) The Engineered Safety Feature Actuation System Instrumentation table, which identifies Engineered Safety Features Trip Values, would be revised for completeness and consistency to include those functional units which do not have specific trip or allowable values.

(9) The Radiation Monitoring Instrumentation TS would be revised to include a new surveillance requirement which would verify that the response time for the control room isolation function is consistent with facility accident analysis assumptions.

(10) The Noble Gas Effluent Monitor (high range) (Unit 2 stack) functional unit would be relocated within the applicable TS as a process monitor, consistent with its current (and original) design function.

(11) The Remote Shutdown Instrumentation TS would be revised consistent with standard practices for TS format such that the action statement would not be entered unless the minimum channels of remote shutdown instrumentation that are required to be operable, as defined by this specification, are not maintained.

(12) The Remote Shutdown Instrumentation TS would be revised by extending the restoration period for an inoperable channel of remote shutdown instrumentation from 7 days to 31 days.

The TS Bases would also be revised, as applicable, to reflect these changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The staff's review is presented below:

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

The proposed changes would not alter the way any structure, system, or component functions and would not alter the manner in which the plant is operated. There are no hardware changes associated with the proposed changes. Therefore, the Reactor Protection System, the Engineered Safety Features Actuation System, and the Radiation Monitoring System instrumentation would continue to perform within the bounds of the previously performed accident analyses. The proposed changes to the operability requirements would not affect the instrumentation's ability to mitigate the design-basis accidents. The design-basis accidents would remain the same postulated events described in the Millstone Unit No. 2 Final Safety Analysis Report, and the consequences of these events will not be affected. Therefore, the proposed changes would not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes would not alter the plant configuration (no new or different type of equipment would be installed) or require any new or unusual operator actions. The proposed changes would not alter the way any structure, system, or component functions and would not alter the manner in which the plant is operated. The proposed changes would not introduce any new failure modes. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes would not reduce the margin of safety since the changes have no impact on any accident analysis assumption. The proposed changes would not decrease the scope of equipment currently required to be operable or subject to surveillance testing, nor would the proposed changes affect any instrument setpoints or equipment safety functions. The proposed changes would not alter the operation of any component or system, nor would the proposed changes affect any safety limits or safety system settings which are credited in a facility accident analysis. Therefore, the proposed changes would not result in a reduction in a margin of safety.

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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: April 18, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications to increase the boron concentration in the spent fuel pool from 730 ppm to 850 ppm, reduce the Boraflex credit from 50 percent to 40 percent, and change the storage criteria, fuel enrichment, and burnup requirements for Region 2A of this 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. Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

No, based upon the following:

Dropped Fuel Assembly

There is no significant increase in the probability of a fuel assembly drop accident in the spent fuel pools when considering the degradation of the Boraflex panels in the spent fuel pool racks coupled with the presence of soluble boron in the spent fuel pool water for criticality control. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water, and the quantity of Boraflex remaining in the racks has no effect on the probability of such a drop accident.

The criticality analysis showed that the consequences of a fuel assembly drop accident in the spent fuel pools are not affected when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron.

Fuel Misloading

There is no significant increase in the probability of the accidental misloading of spent fuel assemblies into the spent fuel pool racks when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron in the pool water for criticality control. Fuel assembly placement and storage will continue to be controlled pursuant to approved fuel handling procedures to ensure compliance with the Technical Specification requirements. These procedures will be revised as needed to comply with the revised Region 2A requirements which would be imposed by the proposed Technical

Specification changes. These revised storage limits were developed with input from station personnel. Their awareness, in conjunction with any procedure changes as described above, will provide additional assurance that an accidental misloading of a spent fuel assembly should not occur.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate soluble boron concentration. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools. The McGuire Station UFSAR Chapter 16, ''Selected Licensee Commitments,'' provides for adequate monitoring of the remaining Boraflex in the spent fuel pool racks. If that monitoring identifies further reductions in the Boraflex panels which would not support the conclusions of the McGuire Criticality Analysis, then the McGuire TSs and design bases would be revised as needed to ensure that acceptable subcriticality are maintained in the McGuire spent fuel storage pools.

Significant Change in Spent Fuel Pool Temperature

There is no significant increase in the probability of either the loss of normal cooling to the spent fuel pool water or a decrease in pool water temperature from a large emergency makeup when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the spent fuel pool water. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

A loss of normal cooling to the spent fuel pool water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density that would result in a decrease in reactivity when Boraflex neutron absorber panels are present in the racks. However, since a reduction in the amount of Boraflex present in the Region 2A racks is considered, and the spent fuel pool water has a high concentration of boron, a density decreases causes a positive reactivity addition. However, the additional negative reactivity provided by the current boron concentration limit, above that provided by the concentration required to maintain k_{eff} less than or equal to 0.95 (1470 ppm), will compensate for the increased reactivity which could result from a loss of spent fuel pool cooling event. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a loss of normal cooling to the spent fuel pool will not be increased. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

A decrease in pool water temperature from a large emergency makeup causes an increase in water density that would result in an increase in reactivity when Boraflex neutron absorber panels are present in the racks. However, the additional negative reactivity provided by the current boron concentration limit, above that provided by the concentration required to maintain k_{eff} less than or equal to 0.95 (1470 ppm), will compensate for the increased reactivity which could result from a decrease in spent fuel pool water temperature. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a decrease in pool water temperature will not be increased. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

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

No. Criticality accidents in the spent fuel pool are not new or different types of accidents. They have been analyzed in Section 9.1.2.3 of the Updated Final Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments for fuel enrichments up to 4.75 weight percent U–235. Specific accidents considered and evaluated include fuel assembly drop, accidental misloading of spent fuel assemblies into the spent fuel pool racks, and significant changes in spent fuel pool water temperature. The accident analysis in the Updated Final Safety Analysis Report remains bounding.

The possibility for creating a new or different kind of accident is not credible. The amendment proposes to take credit for the soluble boron in the spent fuel pool water for reactivity control in the spent fuel pool while maintaining the necessary margin of safety. Because soluble boron has always been present in the spent fuel pool, a dilution of the spent fuel pool soluble boron has always been a possibility; however, a criticality accident resulting from a dilution accident was not considered credible. A spent fuel pool dilution evaluation * * * has demonstrated that a dilution of the boron concentration in the spent fuel pool water which could increase the rack k_{eff} to greater than 0.95 (constituting a reduction of the required margin to criticality) is not a credible event. The requirement to maintain a revised minimum boron concentration in the spent fuel pool water for reactivity control (at least 850 ppm) will have no effect on normal pool operations and maintenance. There are no changes in equipment design or in plant configuration. This revised requirement will not result in the installation of any new equipment or modification of any existing equipment. Therefore, the proposed amendment will not result in the possibility of a new or different kind of accident.

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

No. The proposed Technical Specification changes and the resulting McGuire Region 2A spent fuel storage operating limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those revised limits are based on a plant specific criticality analysis * * * based on the ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology'' * * * The Westinghouse methodology for taking credit for soluble boron in the spent fuel pool has been reviewed and approved by the NRC * * * This methodology takes partial credit for soluble boron in the spent fuel pool and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

(1) k_{eff} shall be less than 1.0 if fully flooded with unborated water which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level; and

(2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

The criticality analysis utilized credit for soluble boron to ensure k_{eff} will be less than or equal to 0.95 under normal circumstances, and storage configurations have been defined using a 95/95 k_{eff} calculation to ensure that the spent fuel rack k_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including uncertainties, tolerances and accident conditions in the presence of spent fuel pool soluble boron. The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to exceeding a k_{eff} of 0.95 has been evaluated * * * and shown to be not credible. Accordingly, the required margin to criticality is not reduced.

Previous evaluations * * * have shown that the dilution of the spent fuel pool boron concentration from the conservative assumed initial boron concentration (2475 ppm) to the minimum boron concentration required to maintain $k_{eff} \leq 0.95$ (850 ppm) is not credible. The dilution analyses, along with the 95/95 criticality calculation which shows that the spent fuel rack k_{eff} will remain less than 1.0 when flooded with unborated water, provide a level of safety comparable to the conservative criticality analysis methodology* * *

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the facility's 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: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: April 19, 2002.

Description of amendment request: The proposed change revises Technical

Specification (TS) 5.5.10, ''Technical Specification (TS) Bases Control Program,'' to provide consistency with changes to 10 CFR 50.59 as published in the **Federal Register** (64 FR 53582) on October 4, 1999, that became effective March 13, 2001. The proposed changes to TS 5.5.10 are made to incorporate the change made in 10 CFR 50.59 to remove the phrase ''unreviewed safety question.'' The proposed changes to TS 5.5.10 are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF–364, Revision 0, as amended by the Westinghouse Owners Group (WOG) editorial change WOG–ED–24.

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

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

The proposed change deletes the reference to ''unreviewed safety question'' as defined in 10 CFR 50.59. Deletion of the definition of ''unreviewed safety question'' was approved by the NRC with the revision of 10 CFR 50.59. This change is administrative in nature. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the probability or consequences of any accident previously evaluated are not significantly affected. There is no increase in the radiological dose at the site boundary for any previously evaluated accident. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. These changes are considered administrative in nature and do not modify, add, delete, or relocate any technical requirements in the TS. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph 10 CFR 50.59(c)(2) continue to require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the revision to 10 CFR 50.59. 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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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 14, 2002

Description of amendment request: The proposed changes will amend the Operating License to revise the as-found safety function lift setpoint tolerances for the Safety and Relief Valves (S/RVs) for River Bend Station, Unit 1. The proposed amendment does not change the actual setpoint or the way the S/RVs are operated, would be limited to the lower tolerances and would not affect the upper limits, and would only apply to the as-found tolerance and not to the as-left tolerance which will remain unchanged. The as-found tolerances are used for determining operability and to increase sample sizes for testing. There will be no change to the valves as installed in the plant. The proposed amendment would also allow surveillance of the relief mode of operation of the S/RVs without physically lifting the disk of a valve off the seat at power.

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

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

Response: No.

These changes have no influence on the probability or consequences of any accident. The setpoint tolerance change does not [a]ffect the operation of valves that are installed in the plant or change the as-left tolerance which will remain at ±1%. The setpoint tolerances for valves that have been tested or refurbished are not being changed. The change only has an [a]ffect on increased sampling for operability and for IST [in-service testing] purposes. The change to the tolerance only affects the lower limit for opening the valve and does not change the upper limit which is the limit that protects from overpressurization.

There is no increase in the probability or consequences of any accident based on the changes to the remote actuation testing of the valves because the valve opening capability will continue to be bench tested and the actuator will be tested independently. The open and close capabilities will therefore be demonstrated satisfactorily.

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

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

Response: No.

No new or different accidents are created because the proposed changes do not change the configuration or operation of the plant in any way. The setpoint tolerance changes only affect the criteria that determines when a valve test is considered to be a failure and is limited to the lower limit. It does not change the criteria for the upper limit that protects against overpressurization.

The changes to the remote actuation testing continue to provide assurance that the valves have open and close capabilities and remain consistent with the intent of the present surveillance.

Therefore, the proposed change does 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?

Response: No.

The proposed changes do not change the configuration or operation of the plant in any way. The setpoint tolerance changes only affect the criteria that determines when a valve test is considered to be a failure and is limited to the lower limit. It does not change the criteria for the upper limit that protects against overpressurization.

The changes to the remote actuation testing continue to provide assurance that the valve has open and close capability and is consistent with the intent of the present surveillance.

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

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

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

NRC Section Chief: Robert A. Gramm.

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 14, 2002.

Description of amendment request: Entergy Operations, Inc. (Entergy) requests changes to the Degraded Voltage—Voltage basis and loss-of-coolant accident (LOCA) time delay allowable values (Technical Specification Table 3.3.8.1–1, Items 1.c and 1.e; and Items 2.c and 2.e) to reflect the results of new calculations performed in association with a design basis reconstitution.

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

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

Response: No.

The change in the degraded voltage protection voltage and time delay allowable values allows the protection scheme to function as originally designed. The proposed allowable values ensure that the Class 1E distribution system remains connected to the offsite power system when adequate offsite voltage is available and motor starting transients are considered. Replacement of the Division 1 and 2 degraded voltage relays provide operational flexibility to accommodate the proposed protection voltage allowable values, which are more conservative than the current limits. Calculations have demonstrated that adequate margin is present to support the decrease in the minimum allowable Division 3 degraded voltage. The small increase in the time delay allowable values more accurately reflects the actual load sequencing experienced during an accident condition. The proposed time delay continues to provide equipment protection while preventing a premature separation from offsite power. The diesel start due to a Loss of Coolant Accident signal is not impacted by this change. During an actual degraded voltage condition, the degraded voltage time delays will continue to isolate the Class 1E distribution system from offsite power before the diesel is ready to assume the emergency loads, which is the limiting time basis for mitigating system responses to the accident. For this reason, the existing Loss of Power / Loss of Coolant accident analysis continues to be valid.

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

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

Response: No.

The proposed change involves the revision of degraded voltage protection voltage and time delay allowable values to satisfy existing design requirements. Component replacement necessary to support these new values will be performed in accordance with plant procedures, which ensure adherence with all quality requirements. No additional failure mechanisms are introduced as a result of the changes to the allowable values.

Therefore, the proposed change does 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?

Response: No.

The proposed protection voltage allowable values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient voltage is available to the required equipment. The small increase in the time delay allowable values more accurately reflects the actual load sequencing experienced during an accident condition. The proposed time delay continues to provide equipment protection while preventing a premature separation from offsite power. The diesel start due to a Loss of Coolant Accident signal is not impacted by this change. During an actual degraded voltage condition, the degraded voltage time delays will continue to isolate the Class 1E distribution system from offsite power before the diesel is ready to assume the emergency loads.

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

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

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

NRC Section Chief: Robert A. Gramm.

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 14, 2002.

Description of amendment request: Entergy Operations, Inc. is proposing to revise the River Bend Station, Unit 1 (RBS), Administrative Technical Specifications (TSs) regarding containment leak rate testing. The proposed change will revise RBS Administrative TS 5.5.13 to add an exception to the commitment to follow the guidelines for Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' The exception is taken to the interval guidance in NEI 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Base Option of 10CFR50, Appendix J.'' The effect of this request will be a one-time extension of the interval between tests from 10 years to 15 years.

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

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

Response: No.

10CFR50, Appendix J was amended to incorporate provisions for performance-based testing in 1995. The proposed amendment to Technical Specification (TS) 5.5.13 adds a one-time extension to the current interval for Type A testing (i.e., the integrated leak rate test). The current interval of ten years, based on past performance, would be extended on a one-time basis to 15-years from the date of the last test. The proposed extension to the Type A test cannot increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences since research documented in NUREG–1493, ''Performance Based Containment Leak Rate Test Program,'' has found that, generally, fewer than 3% of the potential containment leak paths are not identified by Type B and C testing. A risk evaluation of the interval extension for RBS is consistent with these results. In addition, at RBS, the testing and containment inspections also provide a high degree of assurance that the containment will not degrade in a manner detectable only by a Type A test. Inspections required by the Maintenance Rule (10CFR50.65) and by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code are performed to identify containment degradation that could affect leaktightness.

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

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

Response: No.

The proposed extension to the interval for the Type A test does not involve any design or operational changes that could lead to a new or different kind of accident from any accidents previously evaluated. The test itself is not being modified, but is only intended to be performed after a longer interval. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

Therefore, the proposed change does 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?

Response: No.

The generic study of the increase in the Type A test interval, NUREG–1493, concluded there is an imperceptible increase in the plant risk associated with extending the test interval out to twenty years. Further, the extended test interval would have a minimal effect on this risk since Type B and C testing detect 97% of potential leakage paths. For the requested change in the RBS ILRT (integrated leak rate testing) interval, it was determined that the risk contribution of leakage will increase 0.32%. This change is considered very small and does not represent a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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 14, 2002.

Description of amendment request: The proposed modification of the River Bend Station Technical Specifications is to revise several of the Surveillance Requirements (SRs) pertaining to testing of the Division 3 standby diesel generator (DG) and manual transfer test for offsite circuits. The proposed change would modify specific restrictions associated with these SRs that prohibit performing required testing in Modes 1, 2, or 3. The affected SRs are SR 3.8.1.8, SR 3.8.1.9, SR 3.8.1.10, SR 3.8.1.11, SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.16, SR 3.8.1.17, SR 3.8.1.18, and SR 3.8.1.19.

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

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

Response: No.

The DG and its associated emergency loads are accident mitigating features, not accident initiating equipment. Therefore, there will be no impact on any accident probabilities by the approval of the requested amendment.

The design of plant equipment is not being modified by these proposed changes. As such, the ability of the DG to respond to a design basis accident will not be adversely impacted by these proposed changes. The capability of the DG to supply power in a timely manner will not be compromised by permitting performance of DG testing during periods of power operation. Additionally, limiting testing to only one DG at a time ensures that design basis requirements for backup power is met, should a fault occur on the tested DG. Therefore, there would be no significant impact on any accident consequences.

Based on the above, the proposed change to permit certain DG surveillance tests to be performed during plant operation will have no [a]ffect on accident probabilities or consequences. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

No new accident causal mechanisms would be created as a result of NRC [Nuclear Regulatory Commission] approval of this amendment request since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration with the exception of the plant mode in which the testing is conducted. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

Based on the above, implementation of the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the proposed change does 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?

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes to the testing requirements for the DG do not affect the operability requirements for the DG, as verification of such operability will continue to be performed as required. Continued verification of operability supports the capability of the DG to perform its required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Consequently, the performance of these fission product barriers will not be impacted by implementation of this proposed amendment.

In addition, the proposed changes involve no changes to setpoints or limits established or assumed by the accident analysis. On this and the above basis, no safety margins will be impacted.

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

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: May 30, 2002.

Description of amendment request: The proposed amendment would revise the requirements in several administrative programs in Technical Specification (TS) Section 6.0, ''Administrative Controls.'' Specifically, the proposed amendment would: (1) Replace the specific management titles for several organizational positions with generic titles, (2) replace the title of the Quality Assurance Program Description with a reference to the quality assurance program described or referenced in the Updated Final Safety Analysis Report (UFSAR), and (3) delete the functions of the Station Nuclear Safety and the Nuclear Facilities Safety Committees and the Vice President-Nuclear Power since their duties and responsibilities are described in the Quality Assurance Program Description. The proposed changes reflect the organizational integration at the Indian Point Energy Center.

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

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

The proposed change eliminates the redundant controls on elements of the managerial and administrative controls implemented by the quality assurance program described or referenced in the UFSAR. There are no changes proposed to the design, operation, maintenance or testing of the plant's systems, structures or components. Therefore, the assumptions of the operability or performance of systems, structures, or components in accident analyses are unchanged.

The adequacy of the managerial and administrative controls used to assure safe operation were previously accepted by the NRC [Nuclear Regulatory Commission] in their approval of the quality assurance program description. The changes to the existing controls were evaluated under 10 CFR 50.54 to ensure the changes would not reduce the commitments in the quality assurance program description previously accepted by the NRC. Therefore, there is no increase in the probability or in the consequences of an accident previously evaluated.

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

The proposed changes do not affect the design, operation, maintenance, or testing of a plant system, structure or component. No new or unanalyzed conditions can be created through the proposed replacement of specific administrative position titles with generic position titles, since the authority, responsibility and qualification for each required position are specified in the quality assurance program described or referenced in the UFSAR.

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

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

The proposed changes do not affect a design function or operation of any plant structure, system, or component. The change does not affect the method of ENO's [Entergy Nuclear Operations'] compliance with any regulation. The changes to the quality assurance program as described or referenced in the UFSAR were evaluated under 10 CFR 50.54 and it was determined that the changes do not reduce any commitments from the quality assurance program description that was previously evaluated and accepted by the NRC.

Therefore, the proposed changes do not result in a change to any of the safety analyses or 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: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Richard J. Laufer.

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 amendment request: March 19, 2002.

Description of amendment request: The proposed amendment would revise the method of controlling the fuel cycle unfavorable exposure time (UET) related to an anticipated transient without scram (ATWS) event. The current methodology controls UET by limiting the value of the moderator temperature coefficient (MTC) inherent in the reactor core design. The proposed license amendment would utilize the Configuration Risk Management Program to administratively control the availability of ATWS risk significant equipment to minimize core UET. By removing the UET MTC constraint, reload cores may be designed with a more positive MTC as allowed by the TS, therefore resulting in significant benefits including reduced fuel cost, reduced outage time, and reduced amount of spent fuel.

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

The change in the methodology of controlling the UET associated with an ATWS event will not increase the probability of any accident previously evaluated, including an ATWS event. All systems, including the existing ATWS Mitigating Systems Actuation Circuitry (AMSAC), will continue to be operated in accordance with current design requirements, and no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the UFSAR.

Currently, the UET for a given fuel cycle must be less than 5% of the operating cycle under a ''base case'' set of plant conditions (i.e., 100% power-operated relief valve (PORV) capacity, 100% AFW system availability, no control rod insertion capability, and AMSAC available). The proposed license amendment would replace the 5% fuel cycle limit on UET with the requirement to administratively control ATWS risk significant equipment when core conditions are ''unfavorable'' over the entire operating cycle. The goal of the administrative control program is to minimize the UET at all times. The methodology used to determine the UET will remain the same as the currently approved methodology. The Configuration Risk Management Program (CRMP), currently described in the Byron Station and Braidwood Station Technical Requirements Manual (TRM), Appendix T, will be used to manage the availability of ATWS risk significant equipment. The CRMP will provide a proceduralized process to perform a configuration risk assessment of the plant equipment configuration and availability prior to planned on-line maintenance of the ATWS risk significant equipment and/or functions. The CRMP is currently used as a tool to manage maintenance activities to minimize any increase in the consequences of an abnormal event or accident. Development of the Byron Station and Braidwood Station CRMP is consistent with 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' paragraph (a)(4), and is governed by Work Control Procedure, WC–AA–101, ''On-Line Work Control Process.''

The ATWS risk significant equipment which will be monitored by the CRMP includes the:

• Rod control system;
• AFW system;
• Pressurizer PORVs; and
• ATWS Mitigating Systems Actuation Circuitry (AMSAC)

This change in methodology will also have no effect on the consequences of any accident previously evaluated including an ATWS event. Should an ATWS occur during an ''unfavorable'' fuel cycle period, the consequences of this event will remain unchanged under the new methodology which only administratively controls plant equipment availability associated with the UET. Also, the consequences of an ATWS event with the core designed with a more positive MTC remain acceptable. Although the time to RCS [reactor coolant system] overpressure and resultant loss-of-coolant accident (LOCA) may decrease, the consequences of the LOCA remain unchanged.

Based on this evaluation, it is concluded that the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes do 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 Byron Station and the Braidwood Station systems, structures or components are unchanged by the proposed TS change which would utilize an alternate method of controlling the UET of a fuel cycle. No transient event would result in a new sequence of events that could lead to a new accident scenario.

No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of utilizing the CRMP to monitor ATWS risk significant equipment. In addition, this methodology does not create any new failure modes that could lead to a different kind of accident. Software changes to the existing CRMP will be made to monitor the above mentioned ATWS risk significant equipment.

Based on this analysis, it is concluded that 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 effect on any safety-related system. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The newly proposed methodology of monitoring and controlling the UET during an operating cycle is more conservative than the currently approved method and; therefore, will increase the margin of safety.

Currently, the UET for a given fuel cycle is limited to less than 5% of the operating cycle and is only evaluated for a ''base case'' set of plant conditions (i.e., 100% PORV capacity, 100% AFW system availability, no control rod insertion capability, and AMSAC available). The UET is currently limited by constraining the value of the MTC inherent in the reload reactor core design.

The proposed methodology will utilize the CRMP as a tool to monitor the availability of ATWS risk significant equipment during the entire operating cycle. By effectively managing the planned on-line maintenance of ATWS risk significant equipment, the cycle UET will be minimized at all times. This methodology also analyzes different combinations of ATWS risk significant equipment availability in addition to the ''base case'' conditions. The proposed license amendment would replace the 5% fuel cycle limit on UET with the requirement to administratively control ATWS risk significant equipment when core conditions are ''unfavorable'' over the entire operating cycle. The goal of the administrative program is to minimize the UET at all times. The methodology used to determine the UET will remain the same as the currently approved methodology. The Configuration Risk Management Program (CRMP) currently described in the Byron Station and Braidwood Station Technical Requirements Manual (TRM), Appendix T, will be used to manage the availability of ATWS risk significant equipment. The CRMP will provide a proceduralized process to perform a configuration risk assessment of the plant equipment configuration and availability prior to planned on-line maintenance of the ATWS risk significant equipment and/or functions. The CRMP is currently used as a tool to manage maintenance activities to minimize any increase in the consequences of an abnormal event or accident. Development of the Byron Station and Braidwood Station CRMP is consistent with 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' paragraph (a)(4), and is governed by Work Control Procedure, WC–AA–101, ''On-Line Work Control Process.''

Based on this evaluation, the proposed TS changes do not involve a significant reduction in a margin of safety.

Based upon the above analyses and evaluations, we have concluded that the proposed change to the TS involve no 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: June 5, 2002.

Description of amendment request: The amendment would revise the Improved Technical Specifications to increase the maximum allowed rated thermal power for Crystal River Unit 3 from 2544 MegaWatts-thermal (MWt) to 2568 MWt.

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

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

The proposed change will increase the maximum core power level from 2544 MWt to 2568 MWt. This increase will only require adjustments and calibrations of existing plant instrumentation and control systems. No hardware upgrades or equipment replacements are needed to implement the proposed change.

Nuclear steam supply systems (NSSS) and balance-of-plant (BOP) systems and components that could be affected by the proposed change have been evaluated using revised NSSS design parameters based on a core power level of 2568 MWt. The results of these evaluations, which used well-defined analysis input assumptions/ parameter values and currently approved analytical techniques, indicate that CR–3 systems and components will continue to function within their design parameters and remain capable of performing their required safety functions at 2568 MWt. Since the revised NSSS parameters remain within the design conditions of the reactor coolant system (RCS) functional specification, the proposed change will not result in any new design transients or adversely affect the current CR–3 design transient analyses.

The accidents analyzed in Chapter 14 of the CR–3 Final Safety Analysis Report (FSAR) have been reviewed for the impact of the uprate. Based on the power levels assumed in the current safety analyses, it has been determined that all FSAR and supporting analyses bound the uprate. This includes the dose calculations for the design basis radiological accidents, which assume a power level of 2619 MWt (2568 MWt plus an assumed 2 percent measurement uncertainty).

Based on the above, the change will not increase the probability or consequences of an accident previously evaluated.

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

As discussed above, no hardware upgrades or equipment replacements are required to implement the proposed change. All CR–3 systems and components will continue to function within their design parameters and remain capable of performing their required safety functions. The proposed change does not impact current CR–3 design transients or introduce any new transients. The design, physical configuration and operation of the plant will not be changed; as a result, no new equipment failure modes will be introduced. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Challenges to the fuel, reactor coolant system (RCS) pressure boundary and containment were evaluated for uprate conditions. Core analyses show that the implementation of the power uprate will continue to meet the current nuclear design basis. Impacts to components associated with RCS pressure boundary structural integrity, and factors such as pressure/temperature limits, vessel fluence, and pressurized thermal shock (PTS) were determined to be bounded by current analyses. Mass and energy release to the containment from a loss-of-coolant accident (LOCA) or main steam line break are also bounded by current analyses, which assume an initial power level of 2619 MWt.

As discussed above, all systems will continue to operate within their design parameters and remain capable of performing their intended safety functions following implementation of the proposed change. Finally, the current CR–3 safety analyses, including the design basis radiological accident dose calculations, bound the uprate.

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

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

Attorney for licensee: R. Alexander Glenn, Associate General Counsel (MAC–BT15A), Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Acting Section Chief: Kahtan N. Jabbour.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: May 22, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification 6.9.1.11.b to add two NRC-approved topical reports to the Core Operating Limits Report (COLR) methodology list, and delete superseded reports. Also, the method of listing topical reports would be revised to be consistent with Technical Specifications Task Force (TSTF) 363, which has been approved by the NRC.

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment updates the list of COLR methodologies and would allow the use of two new NRC approved methodologies, EMF–2310(P)(A), ''SRP [Standard Review Plan] Chapter 15 Non-LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors [(PWR)],'' and EMF–2328 (P)(A), ''PWR Small Break LOCA Evaluation Model, S–RELAP5 Based,'' for the St. Lucie Unit 1 safety analyses. The proposed changes have no adverse impact on the operation of the plant and have no relevance to the accident initiators. There are no changes to the plant configuration, and thus the frequency of occurrence of previously analyzed accidents is not affected by the proposed changes.

With the updated methodologies, the safety analysis would continue to meet the analysis acceptance criteria consistent with the design basis requirements. The proposed changes have no adverse effect on the safety analysis and thus would not involve a significant increase in the consequences of design basis accidents. Changes to the COLR limits would continue to be controlled per Generic Letter 88–16 under the provisions of 10 CFR 50.59 and the requirements of TS 6.9.1.11.c.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(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 previously evaluated.

The proposed amendment updates the list of approved methodologies in TS 6.9.1.11.b. These changes would not create the possibility of a new kind of accident since there is no change to the plant configuration, systems or components, which would create new failure modes. The modes of operation of the plant remain unchanged.

Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(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 have no adverse impact on the safety analysis. The changes proposed would continue to provide margin to the acceptance criteria for Specified Acceptable Fuel Design Limits (SAFDL), 10 CFR 50.46(b) requirements, primary and secondary overpressurization, peak containment pressure, potential radioactive releases, and existing limiting conditions for operation. The future use of updated approved methodologies would follow all design basis requirements to ensure that a safety margin to the acceptance criteria would continue to remain available at all power levels for operation of St. Lucie Unit 1.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) associated with refueling operations to remove the requirement for operability of certain systems (containment penetrations, spent fuel pool and shield building ventilation, and containment isolation) when handling fuel assemblies that have decayed a sufficient period of time such that dose consequences of the postulated fuel handling accident (FHA) remain below the limits of 10 CFR Part 100 and the NRC Standard Review Plan with these systems unavailable. The proposed changes are consistent with the Standard TS for Combustion Engineering plants and a portion of Nuclear Energy Institute TS Task Force change traveler TSTF–51, Revision 2.

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the St. Lucie Units 1 and 2 TSs incorporate line item improvements that are based on assumptions in the postulated fuel handling accident analyses. These proposed changes remove the applicability of TSs regarding operability of certain systems (containment penetrations, spent fuel pool and shield building ventilation, and containment isolation) when handling fuel assemblies that have decayed a sufficient period of time. The results of the FHA analyses demonstrate that sufficient radioactive decay has occurred after 72 hours such that the resulting dose consequences are well within the limits given in 10 CFR 100 and within the limits given in the Standard Review Plan, NUREG–0800. The systems that have been included in these proposed changes will have administrative controls in place to assure that systems are available and can be promptly returned to operation to further reduce dose consequences. These administrative controls will include a single normal or contingency method to promptly close the primary or secondary containment penetrations. These prompt methods need not completely block the penetrations nor be capable of resisting pressure, but are to enable the ventilation systems to draw the release from the postulated FHA such that it can be treated and monitored. This will result in lower doses than those calculated for the FHA.

The equipment or systems involved are not initiators of an accident. Operability of these systems or equipment during fuel movement and/or core alterations has no affect on the probability of any accident previously evaluated.

The proposed changes do not significantly increase the consequences of a fuel handling accident as previously evaluated. The calculated doses are well within the limits given in 10 CFR Part 100 and within the limits given in the Standard Review Plan, NUREG–0800. In addition, the calculated doses are larger than the expected doses because the calculations do not credit any filtration or containment of the source term that will occur by the administrative controls that will be in place.

The changes being proposed do not affect assumptions contained in other plant safety analyses or the physical design of the plant, nor do they affect other TSs that preserve safety analysis assumptions. Therefore, 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 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 previously evaluated.

The proposed changes to the TSs do not affect or create a different type of fuel handling accident. The fuel handling accident analyses assume that all of the iodine and noble gases that become airborne, escape, and reach the exclusion area boundary and low population zone with no credit taken for filtration, containment of the source term, or for decay or deposition. The proposed changes do not involve the addition or modification of equipment nor do they alter the design of plant systems. The revised operations are consistent with the fuel handling accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The calculated doses are well within the limits given in 10 CFR Part 100 and within the limits given in the Standard Review Plan, NUREG–0800. The proposed changes do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any TS that is related to the establishment of or maintenance of a safety margin.

The systems that have been included in the proposed change will have administrative controls in place to assure that the systems are available and can be promptly returned to operation to further reduce dose consequences. These administrative controls will include a single normal or contingency method to promptly close the primary or secondary containment penetrations. These prompt methods need not completely block the penetrations nor be capable of resisting pressure, but are to enable the ventilation systems to draw the release from the postulated FHA such that it can be treated and monitored.

Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

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

Date of amendment request: July 30, 2001.

Description of amendment request: Proposed amendment revises the Cooper Nuclear Station licensing basis with respect to containment overpressure contribution to emergency core cooling system pump net positive suction head.

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 requested amendment:

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

The requested license amendment does not result in any new accident initiators, nor are there changes being proposed to other plant systems or equipment postulated to initiate an accident previously evaluated. Thus, the proposed change does not involve a significant increase in the probability of an accident previously evaluated in the USAR [Updated Safety Analysis Report].

The containment overpressure evaluation conservatively demonstrates that adequate margin between the available containment overpressure and the overpressure required to assure adequate low pressure ECCS [emergency core cooling system] pump NPSH [net positive suction head] are such that ECCS pump operation, as credited in the CNS [Cooper Nuclear Station] accident analysis, remains unchanged. Thus, the ECCS pumps continue to be available to perform the safety functions previously evaluated, and the proposed change does not involve a significant increase in the consequences of an accident previously evaluated in the USAR.

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

The proposed license amendment does not introduce any new equipment or hardware changes. The only equipment affected by this license amendment are the low pressure ECCS pumps. These pumps retain their ability to function following any accident previously evaluated and no new accidents are created as a result of increased reliance on overpressure or methodology changes. Thus, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the USAR.

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

Although there is an increased reliance on containment overpressure, adequate low pressure ECCS pump NPSH is assured, and sufficient margin is conservatively determined to be maintained between the available overpressure and the required overpressure to provide confidence that the ECCS pumps will operate as required. The calculations are revised to show an increased absolute containment overpressure consideration from ~5 psi (original license application) to ~9.5 psi at the time of the peak suppression pool temperatures following a design basis LOCA [loss-of-coolant accident]. At this containment overpressure, the CS [core spray] and RHR [residual heat removal] pumps will utilize only ~4.45 psi and ~6.47 psi, respectively, of the available overpressure. This provides a margin of ~5 psi and ~3 psi, respectively, for the CS and RHR pumps at the peak suppression pool temperature. The calculations also address both short-term and long-term reliance on containment overpressure.

In the short-term (<600 seconds), the RHR pumps do not depend on containment overpressure for adequate NPSH. However, during this short-term period following initiation of the event, the CS pump is conservatively calculated to require as much as ~4.94 psi of containment overpressure to assure adequate NPSH. At the time this overpressure is needed, ~6.85 psi of containment overpressure is available, providing a margin of ~1.9 psi. For the time periods following the peak suppression pool temperature, the required overpressure reliance reduces with time and suppression pool temperature.

During the accident, beyond the time period of the peak suppression pool temperature, a minimum margin of ~0.6 psi is provided for ECCS pump NPSH. However, this minimum margin occurs just prior to 100 hours into the event at a point when no containment overpressure is required for ECCS pump NPSH. During times when containment overpressure is credited, there is a minimum of ~1 psi containment overpressure available.

The analysis also utilizes three new methods for evaluation of the previously evaluated accidents. These are the SHEX code for the containment pressure and temperature response analysis, the ANS 5.1–1979 model for determination of core decay heat, and the use of spatial evaluation of the suppression pool safety relief valve discharge quenchers relative to the ECCS pump intake strainers for prevention of steam bubble ingestion. A benchmark evaluation of the SHEX code is provided which indicates that the results are comparable to previous analysis. The ANS 5.1–1979 model is less conservative than the previously used May-Witt model. However, this change in conservatism is offset by the use of other input parameter changes such as reduced RHR heat exchanger heat removal assumptions and increased service water and suppression pool temperature assumptions. Additionally, both the SHEX code and the ANS 5.1 decay heat model have been previously accepted by NRC as sufficiently conservative analysis methods. The spatial evaluation of the suppression pool safety relief valve discharge quenchers relative to the ECCS pump intake strainers shows steam bubble ingestion is not predicted. This supports the elimination of a local suppression pool temperature limit.

Therefore, sufficient margin and adequate NPSH are demonstrated with the conservatism of a two sigma (two standard deviations) uncertainty in the decay heat model, increased suction strainer debris loading, decreased RHR heat exchanger minimum performance criteria, and increases in SW [service water] and suppression pool temperatures. Thus, the proposed activity does no 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. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: May 7, 2002.

Description of amendment request: The proposed amendment would modify technical specification (TS) requirements for meeting surveillances in TS 4.0.a, TS requirements for missed surveillances in TS 4.0.c, and TS requirements for a Bases control program consistent with TS Bases Control Program described in Section 5.5 of NUREG–1431, Standard Technical Specifications for Westinghouse Plants, Revision 2. The delay period would be extended from the current limit of ''* * * up to 24 hours or up to the limit of the time interval, whichever is less'' to ''* * * up to 24 hours or up to the limit of the time interval, whichever is greater.'' In addition, the following requirement would be added to surveillance requirement 4.0.E: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated May 7, 2002.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: May 8, 2002.

Description of amendment request: The proposed change will exclude the control room normal and emergency air handling system from the requirement to apply Technical Specification (TS) 3.0.4 to actions required by Limiting Condition for Operation 3.7.6 in Modes 5 and 6.

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:

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the significant Hazards Criteria of 10 CFR 50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

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

Response: No.

The proposed change to Technical Specification 3.7.6 does not contribute to the initiation of any accident previously evaluated. The actions within the VCSNS TS associated with the control room normal and emergency air handling system during shutdown (i.e., Modes 5, 6, and defueled) and during the handling of irradiated fuel does not require any physical modification to plant components or systems. Implementing the proposed action has no impact on the probability of an accident.

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

Response: No.

The proposed change to Technical Specification 3.7.6 does not contribute to the initiation of any accident previously evaluated. The actions within the VCSNS TS associated with the control room normal and emergency air handling system during shutdown (i.e., Modes 5, 6, and defueled) and during the handling of irradiated fuel do not introduce any new accident initiator mechanisms. The exclusion of the provisions of Specification 3.0.4 requirements from Specification 3.7.6 Mode 5 and 6, action requirements does not cause the initiation of any accident nor create any new credible limiting single failure nor result in any event previously deemed incredible being made credible. As such, it does not create the possibility of an accident different than any evaluated in the FSAR [Final Safety Analysis Report].

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

Response: No.

When invoked, the proposed change will allow operational transitions involving Modes 5 and 6 within the remedial measures currently defined in the specification, including the following when one train is inoperable:

- A 7-day AOT [allowed outage time] to restore an inoperable train to OPERABLE status.
- Operation of the OPERABLE control room emergency air cleanup system in the recirculation mode.

Although the overall reliability of the system is reduced because a single failure in the OPERABLE train could result in a loss of function, the 7-day AOT provides adequate margins of safety because of the low probability of a design basis accident (DBA) occurring during this time period and the ability of the remaining train to provide the required capability. Adequate margins of safety are also provided by the alternative action that places the unit in a protected condition because this ensures the remaining train is operating, that no failure preventing automatic actuation will occur, and that any active failure can be readily detected.

With two trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

Given the degree of protection provided by the current specification, exclusion * * * of the provisions of Specification 3.0.4 is judged to not result in a significant reduction in the margin of safety as described in the bases of any Technical Specification.

Pursuant to 10 CFR 50.91, the preceding analyses provides a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: John A. Nakoski

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

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment revises the Shutdown Margin limits to Core Operating Limits Report and does not change any requirements that are currently in place.

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

Response: No.

The proposed change to relocate the Shutdown Margin limits to the Core Operating Limits Report [COLR] does not change any requirements that are currently in place. No actual plant equipment or accident analyses will be affected by the proposed change. The Shutdown Margin limits in the COLR will continue to be controlled by the STP [South Texas Project] programs and procedures. The safety analysis addressed in the UFSAR [updated final safety analysis report] will be examined with respect to changes in these limits, which are obtained using NRC-[Nuclear Regulatory Commission] approved methodologies. Changes to the COLR will be conducted per the requirements of 10 CFR 50.59.

The proposed changes to modify the Specification action requirements changing the structure of the specifications to be more consistent with NUREG 1431, Westinghouse Improved Standard Technical Specifications have no technical impact. The changes clarify time requirements and remove details that remain consistent with the UFSAR safety analysis. The changes have no effect on the reactivity control systems to perform their design functions and involve no change to the accident analyses.

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

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

Response: No.

The proposed changes have no influence or impact on, nor do they contribute in any way to the probability or consequences of an accident. No safety-related equipment or safety function will be altered as a result of these proposed changes. The SDM [shutdown margin] will continue to be calculated using the NRC-approved methods that will be submitted to the NRC. The Technical Specifications will continue to require operation within these reactivity limits.

The proposed change modifies the Specification action requirements but does not change the way the system is operated. When the limiting condition for operation is exceeded, the boration control system will continue to be operated in a manner consistent with the safety analyses. The details concerning boron flow rate and concentration that are removed from the Specifications will be added to the TS [technical specification] Bases for the purposes of providing an example.

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

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

Response: No.

The relocation of the Shutdown Margin limits to the COLR will not change any requirements. The values for SDM will remain consistent with the UFSAR and will continue to provide their safety function through the Shutdown Margin Specification. Actions required to be taken to restore SDM will remain in the TS. Therefore, the proposed change will not affect the limits on reactivity control, and will not permit operations that could result in exceeding these limits.

The proposed change modifies action requirements for restoring shutdown margin or refueling boron concentration. The combination of parameters currently in the Specification that are being removed discuss one means, where as several system lineups and boration sources have been evaluated in the safety analysis as acceptable to restore Shutdown Margin. Also, the time requirements for the action were modified to be consistent with the safety analysis assumptions. No actual accident analyses will be affected by these proposed changes. The proposed change will not affect reactivity control limits and will not permit operations that could result in exceeding these limits.

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

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

10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Attorney for licensee: A.H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 23, 2002

Description of amendment request: The proposed amendment revises the Unit 2 Operating License and several sections of Technical Specifications to delete information differentiating between Unit 1 and Unit 2 specific to Model E steam generators.

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

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

Response: No.

The Operating Licenses currently reflect plant operation with both Delta 94 and Model E SGs [steam generators], but all Model E SGs will be replaced with Delta 94 SGs by the end of 2002. The proposed administrative change deletes information associated with the Model E SGs and deletes references to Delta 94 SGs. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The Operating Licenses currently reflect plant operation with both Delta 94 and Model E SGs, but all Model E SGs will be replaced with Delta 94 SGs by the end of 2002. The proposed administrative change deletes information associated with the Model E SGs and deletes references to Delta 94 SGs. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The Operating Licenses currently reflect plant operation with both Delta 94 and Model E SGs, but all Model E SGs will be replaced with Delta 94 SGs by the end of 2002. The proposed administrative change deletes information associated with the Model E SGs and deletes references to Delta 94 SGs. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: A.H. Gutterman, Esq., Morgan, Lewis, & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment revises Technical Specifications Limiting Conditions for Operation 3.7.1.5, Main Steam Isolation Valves, and 3.7.1.7, Main Feedwater Isolation Valves.

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

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

Response: No.

The proposed change extends the action completion time for one MSIV [main steam isolation valve] in Mode 1, and one or more in Mode 2 and 3, from 4 hours to 8 hours. Extending the completion time is not an accident initiator and thus does not change the probability that an accident will occur. However, it could potentially affect the consequences of an accident if an accident occurred during the extended unavailability of the inoperable MSIV. The increase in time that the MSIV is unavailable is small and the probability of an event occurring during this time period, which would require isolation of the main steam flow paths, is low.

The proposed change extends the action completion time for one or more MFIVs [main feedwater isolation valves] from 4 hours to 72 hours. Extending the completion time is not an accident initiator and thus does not change the probability that an accident will occur. However, it could potentially affect the consequences of an accident if an accident occurred during the extended unavailability of the inoperable MFIV. The increase in time that the MFIV is unavailable is small and the probability of an event occurring during this time period, which would require isolation of the main feedwater flow paths, is low.

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

2. Will operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Closure of the MSIVs is required to mitigate the consequences of large Steam Line Break inside containment. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Closure of the MFIVs is required to mitigate the consequences of the Main Steam Line Break and Main Feedwater Line Break accidents. The proposed 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 the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not change any Technical Specification Limit or accident analysis assumption. Therefore it does not involve a reduction in a margin of safety.

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

Attorney for licensee: A.H. Gutterman, Esq., Morgan, Lewis, & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 14, 2002.

Description of amendment request: The proposed changes would revise the Technical Specifications and associated Bases to revise the surveillance frequency of the containment spray and recirculation spray system spray header nozzles from a periodic surveillance to a performance-based surveillance.

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 revision to Technical Specifications changes the frequencies of the surveillance requirements for the Containment Spray and Recirculation Spray nozzles. The frequency is being changed from every 10 years to ''following maintenance which could result in nozzle blockage.'' In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

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

The proposed change revises the surveillance frequencies from every 10 years to ''following maintenance which could result in nozzle blockage.'' Analyzed events are initiated by the failure of plant structures, systems, or components. The Containment Spray and Recirculation Spray Systems are not considered to be initiators of any analyzed event. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. As a result, the probability of any accident previously evaluated is not significantly increased.

The proposed change revises the surveillance frequencies. Reduced testing is justified where operating experience has shown that routinely passing a surveillance test performed at a specified interval has no apparent connection to overall component reliability. In this case, routine surveillance testing at the specified frequency is not connected to any activity, which may initiate reduced component reliability, and therefore has been of limited value in ensuring component reliability. Thus, the proposed frequency change is not significant from a reliability standpoint. The proposed containment spray and recirculation spray nozzle surveillance frequencies have been established based on achieving acceptable levels of equipment reliability.

This change does not affect the plant design. Due to the plant design, the spray ring headers are maintained dry. Formation of significant corrosion products is unlikely. Due to their location at the top of the containment, introduction of foreign material from exterior to the headers is unlikely. Since maintenance that could introduce foreign material is the most likely cause for obstruction, testing or inspection following such maintenance would verify the nozzle(s) remain unobstructed and the systems' continued capability to perform their safety function(s). As a result, the consequences of any accident previously evaluated are not significantly affected by the proposed change in surveillance frequencies.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety for this system is based on the capacity of the spray headers. The system is not susceptible to corrosion induced obstruction or obstruction from external sources to the system. Performance of maintenance on a spray ring header would now require evaluation of the potential for nozzle blockage and the need for a test or inspection. Consequently, the spray header nozzles should remain unblocked and available in the event that the safety function is required. Hence, the change in surveillance frequencies 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: Ms. Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

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/reading-rm/adams.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.*

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

Date of application for amendment: July 9, 2001.

Brief description of amendment: The amendment revises the Technical Specifications to be consistent with changes made to 10 CFR 50.59, ''Changes, tests, and experiments.''

Date of issuance: June 4, 2002.

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

Amendment No.: 151.

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

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44162). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 4, 2002.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: September 11, 2001, as supplemented on April 8, 2002.

Brief description of amendment: The amendment revised the Technical Specifications, deleting the cycle-specific footnote regarding the safety limit minimum critical power ratio in Section 2.1.A, and making associated administrative changes.

Date of Issuance: May 31, 2002.

Effective date: May 31, 2002, and shall be implemented within 30 days of issuance.

Amendment No.: 228.

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

Date of initial notice in ***Federal Register:*** November 28, 2001 (66 FR 59501). The April 8, 2002, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 31, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 4, 2001, as supplemented July 20, 2001.

Brief Description of amendments: The proposed license amendments change the Technical Specifications Surveillance Frequency and Action Requirements for the suppression chamber-to-drywell vacuum breakers at the Brunswick Steam Electric Plant, Units 1 and 2.

Date of issuance: June 3, 2002.

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

Amendment Nos.: 223 and 248.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 27, 2001 (66 FR 34280). The July 20, 2001, supplement 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 3, 2002.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: August 27, 2001.

Brief description of amendment: This amendment revised Technical Specification (TS) 3/4.6.1.3, ''Containment Systems—Containment Air Locks'' and the associated TS Bases section.

Date of issuance: June 7, 2002.

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

Amendment No.: 267.

Facility Operating License No. DPR–65: This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** October 31, 2001 (66 FR 55010). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 7, 2002.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: August 6, 2001.

Brief description of amendments: The amendments revise the Technical Specification 3.3.1 allowable values for the reactor trip system instrumentation overtemperature delta temperature and overpower delta temperature set points.

Date of issuance: May 23, 2002.

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

Amendment Nos.: 202 and 183.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2920). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 23, 2002.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: January 8, 2002, as supplemented on April 15, 2002.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.4.1, ''Main Steam Safety Valves,'' to reduce the maximum allowable power range neutron flux high setpoint when one or more main steam line safety valves are inoperable. The amendment also revises the associated TS Basis to incorporate a more conservative equation to calculate this setpoint.

Date of issuance: June 4, 2002.

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

Amendment No.: 228.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10012). The April 15, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 4, 2002.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: January 8, 2002.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.8, ''Refueling, Fuel Storage and Operations with the Reactor Vessel Head Bolts Less Than Fully Tensioned,'' and TS 4.5.F, ''Fuel Storage Building Air Filtration System,'' by deleting the requirements for the Fuel Storage Building Air Filtration System. The amendment also revised the associated Basis sections.

Date of issuance: June 5, 2002.

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

Amendment No.: 229.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10013). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 5, 2002.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: March 13, 2002, as supplemented by letter dated May 23, 2002.

Brief description of amendment: The amendment corrects several errors that were found subsequent to Nuclear Regulatory Commission issuance of Amendment No. 215, which converted the Plant Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 to Improved TSs.

Date of issuance: June 10, 2002.

Effective date: As of the date of issuance and shall be implemented in conjunction with the implementation of Amendment No. 215.

Amendment No.: 218.

Renewed Facility Operating License No. DPR–51: Amendment revised the Technical Specifications/license.

Date of initial notice in ***Federal Register:*** April 30, 2002 (67 FR 21287).

The supplemental letter dated May 23, 2002, provided additional information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 10, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 2, 2001, as supplemented by letter dated March 20, 2002.

Brief description of amendment: The amendment relocated the requirements for the containment recirculation system from the Technical Specifications to the Technical Requirements Manual.

Date of issuance: May 31, 2002.

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

Amendment No.: 245.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 30, 2001 (66 FR 29352). The March 20, 2002, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 2002.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–237, Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois

Date of application for amendment: September 5, 2001.

Brief description of amendment: The amendment revises the battery terminal voltage on float charge for the alternate battery.

Date of issuance: June 6, 2002.

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

Amendment No.: 193.

Facility Operating License No. DPR–19: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10013). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 6, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 9, 2001.

Brief description of amendments: Replace the phrase ''involves an unreviewed safety question as defined in'' with ''requires NRC approval pursuant to,'' maintaining reference to 10 CFR 50.59, ''Changes, tests, and experiments,'' in order to provide consistency with changes to 10 CFR 50.59 as published in the **Federal Register** (64 FR 53582) dated October 4, 1999.

Date of issuance: June 4, 2002.

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

Amendment Nos.: 182 and 169.

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44170). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 4, 2002.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: February 20, 2002.

Brief description of amendments: These amendments revised TS 3/4.6.5, ''Vacuum Relief Valves,'' to make the Limiting Condition for Operation applicable to vacuum relief ''lines'' and extend the allowed outage time for the containment vacuum relief lines from 4 hours to 72 hours. Also, some specific requirements for surveillance testing and valve actuation setpoints are relocated to the TS Bases documents.

Date of Issuance: May 30, 2002.

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

Amendment Nos.: 182 and 125.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 19, 2002 (67 FR 12602). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 30, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 12, 2002

Brief description of amendment: The amendment revises Surveillance Requirement (SR) 4.0.E to extend the delay period, prior to having to declare the subject equipment inoperable, following a missed surveillance. The delay period is extended from the current limit of '' * * * up to 24 hours or up to the limit of the time interval, whichever is less'' to '' * * * up to 24 hours or up to the limit of the time interval, whichever is greater.'' In addition, the following requirement is added to SR 4.0.E: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: May 31, 2002.

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

Amendment No.: 127.

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 2, 2002 (67 FR 15625). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 2002.

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: March 27, 2002, as supplemented by letter dated May 9, 2002.

Brief description of amendment: The amendment revises the maximum allowable value of the reactor protective system (RPS) variable high power trip (VHPT) setpoint from 107.0% to 109.0%. Specifically, TS Table 1–1, ''RPS Limiting Safety System Settings,'' in the Trip Setpoints column for Trip Number 1 [High Power Level (A) 4-Pump Operation] has been revised from 107.0% to 109.0%. In addition, TS Section 1.3(1), ''Basis,'' describing the high power trip initiation, has been revised from 107.0% to 109.0%.

Date of issuance: May 29, 2002.

Effective date: May 29, 2002, to be implemented within 30 days from the date of issuance.

Amendment No.: 210.

Facility Operating License No. DPR–40. Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (67 FR 34478 dated May 14, 2002). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by June 13, 2002, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Nebraska and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 29, 2002.

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

NRC Section Chief: Stephen Dembek.

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

Date of application for amendment: March 18, 2002.

Brief description of amendment: The amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: June 12, 2002.

Effective date: June 12, 2002.

Amendment No.: 82.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 30, 2002 (67 FR 21293). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 12, 2002.

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: January 24, 2002.

Brief Description of amendments: The amendments delete Technical Specifications Section 5.5.3, ''Post Accident Sampling,'' for Farley Nuclear Plant, Units 1 and 2, and thereby eliminated the requirements to have and maintain the post-accident sampling systems.

Date of issuance: May 22, 2002.

Effective date: As of the date of issuance and shall be implemented by December 31, 2002.

Amendment Nos.: 156 and 148.

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

*Date of initial notice in **Federal Register:*** April 30, 2002 (67 FR 21293). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 22, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 8, 2001, as amended by your letter dated April 8, 2002.

Brief description of amendments: The amendments deleted various reporting requirements from the Sequoyah Technical Specifications (TSs) because they are duplicative to the requirements of 10 CFR 50.72 and 10 CFR 50.73. One exception was reporting of steam generator tube inspection results, TS 4.4.5.5.c, which is more stringent than 10 CFR 50.72 and 10 CFR 50.73. Therefore, the request to delete this TS was denied.

Date of issuance: May 24, 2002.

Effective date: Date of issuance, to be implemented within 45 days of issuance.

Amendment Nos.: 276 and 267.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the TSs.

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5339). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 24, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 24, 2001, as supplemented by letter dated April 15, 2002.

Brief description of amendments: The amendments extend the surveillance test interval from ''92 days'' to ''18 months'' for Westinghouse Electric Company Type AR relays with alternating current coils used as Solid State Protection System slave relays, in Surveillance Requirement (SR) 3.3.2.6 and auxiliary (i.e., interposing) relays in the containment ventilation isolation system in SR 3.3.6.5.

Date of issuance: May 31, 2002.

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

Amendment Nos.: 96 and 96.

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

*Date of initial notice in **Federal Register:*** October 17, 2001 (66 FR 52804). The April 15, 2002, supplement provided clarifying information and did not change the original no significant hazards determination consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 31, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 31, 2001, as supplemented by letters dated October 17, 2001, and March 5, 2002.

Brief Description of amendments: These amendments revise the Technical Specifications to add a 14-day allowed outage time for the power-operated relief valve backup air supply, and additional surveillance, functional testing, and calibration requirements.

Date of issuance: May 31, 2002.

Effective date: May 31, 2002.

Amendment Nos.: 231 and 231.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64310). The supplements dated October 17, 2001, and March 5, 2002, provided clarifying information that did not change the scope of the May 31, 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 May 31, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 14th day of June 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–15683 Filed 6–24–02; 8:45 am]

BILLING CODE 7590–01–P

COMMISSION ON OCEAN POLICY

Public Meeting

AGENCY: U.S. Commission on Ocean Policy.

ACTION: Notice.

SUMMARY: The U.S. Commission on Ocean Policy will hold its seventh regional meeting, the Commission's ninth public meeting, to hear and discuss coastal and ocean issues of concern to the Northeast region of the United States, covering the area from New Jersey to Maine.

DATES: Public meetings will be held Tuesday, July 23, 2002 from 12:30 p.m. to 6 p.m. and Wednesday, July 24, 2002 from 8:30 a.m. to 6 p.m.

ADDRESSES: The meeting location is Historic Faneuil Hall, 0 Faneuil Hall Square, Boston, Massachusetts 02109.

FOR FURTHER INFORMATION CONTACT: Terry Schaff, U.S. Commission on Ocean Policy, 1120 20th Street, NW., Washington, DC, 20036, 202–418–3442, *schaff@oceancommission.gov.*

SUPPLEMENTARY INFORMATION: This meeting is being held pursuant to requirements under the Oceans Act of 2000 (Pub. L. 106–256, Section 3(e)(1)(E)). The agenda will include presentations by invited speakers representing local and regional government agencies and non-governmental organizations, comments from the public and any required administrative discussions and executive sessions. Invited speakers and members of the public are requested to submit their statements for the record electronically by Monday, July 15, 2002 to the meeting Point of Contact. A public comment period is scheduled for Wednesday, July 24, 2002. The meeting agenda, including the specific time for the public comment period, and guidelines for making public comments will be posted on the Commission's Web site at *http://www.oceancommission.gov* prior to the meeting.

Dated: June 19, 2002.

James D. Watkins,

Admiral, USN (Ret.), Chairman, U.S. Commission on Ocean Policy.

[FR Doc. 02–15948 Filed 6–24–02; 8:45 am]

BILLING CODE 6820–WM–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–46085; File No. SR–Amex–2002–42]

Self-Regulatory Organizations; Notice of Filing and Immediate Effectiveness of Proposed Rule Change by the American Stock Exchange LLC Relating to a Six-Month Extension of the Exchange's Pilot Program for Automatic Execution of Orders for Exchange Traded Funds

June 17, 2002.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] notice is hereby given that on May 23, 2002, the American Stock Exchange LLC (''Amex'' or ''Exchange'') filed with the Securities and Exchange Commission (''Commission'') the proposed rule change as described in Items I, II, and III below, which Items have been prepared by the Exchange. The proposed rule change has been filed by the Amex as a ''non-controversial'' rule change under Rule 19b–4(f)(6) under the Act.[3] The Commission is publishing this notice to solicit comments on the proposed rule change from interested persons.

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

The Amex seeks a six-month extension of Amex Rule 128A to continue its pilot program for the automatic execution of orders for Exchange Traded Funds (''ETFs''). The text of the proposed rule change is available at the Office of the Secretary, the Amex, and at the Commission.

II. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

In its filing with the Commission, the Exchange included statements concerning the purpose of and basis for the proposed rule change and discussed any comments it received on the proposed rule change. The text of these statements may be examined at the places specified in Item IV below. The Exchange has prepared summaries, set forth in sections A, B, and C below, of the most significant parts of such statements.

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

[2] 17 CFR 240.19b–4.

[3] 17 CFR 240.19b–4(f)(6).

A. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

1. Purpose

On June 19, 2001, the Commission approved the Exchange's proposal, adopted as Amex Rule 128A, to permit the automatic execution of orders for ETFs on a six-month pilot program basis.[4] On December 20, 2001, the Exchange extended the pilot program for six months.[5] The Exchange now seeks to extend the pilot program for another six months.

Since 1986, the Exchange has had an automatic order execution feature (''Auto-Ex'') for eligible orders in listed options. The Chicago Board Options Exchange, Philadelphia Stock Exchange, and Pacific Exchange established similar automatic option order execution features at about the same time as the Amex, and the newest options exchange, the International Securities Exchange, also features automatic order execution. Auto-Ex, accordingly, has been a standard feature of the options markets for a number of years.

In 1993, the Amex commenced trading Standard and Poor's Depositary Receipts® (''SPDRs®''), the first ETF to be listed and traded on the Exchange. ETFs are individual securities that represent a fractional, undivided interest in a portfolio of securities. Currently, more than 100 ETFs are listed on the Amex. Like an option, an ETF is a derivative security, and, according to the Amex, its price is a function of the value of the portfolio of securities underlying the ETF. Thus, as is the case with options, the Exchange asserts that it is not the price discovery market for ETFs, and that the price discovery market is the market or markets where the underlying securities trade.

The Exchange is now proposing to extend its current Auto-Ex technology for an additional six months to ETFs listed under Amex Rules 1002, 1002A, and 1202. The Amex represents that this will provide investors that send eligible orders to the Exchange with faster executions than they otherwise would receive. The Exchange believes that many investors desire rapid executions in trading securities that are priced derivatively since the value of the underlying instruments may fluctuate

[4] *See* Securities Exchange Act Release No. 44449 (June 19, 2001), 66 FR 33724 (June 25, 2001) (''June Release'') (approving File No. SR–Amex–2001–29).

[5] *See* Securities Exchange Act Release No. 45176 (December 20, 2001), 66 FR 67582 (December 31, 2001) (notice of filing and immediate effectiveness of File No. SR–Amex–2001–105).