Reduction Act of 1995, Pub. L. 104–13. This is the second notice for public comment; the first was published in the **Federal Register** at 66 FR 46292, and two comments, showing a positive response to NSF's implementation of a web-based job recruitment system, were received. NSF is forwarding the proposed renewal submission to the Office of Management and Budget (OMB) for clearance simultaneously with the publication of this second notice. Comments regarding (a) whether the collection of information is necessary for the proper performance of the functions of the agency, including whether the information will have practical utility; (b) the accuracy of the agency's estimate of burden including the validity of the methodology and assumptions used; (c) ways to enhance the quality, utility and clarity of the information to be collected; (d) ways to minimize the burden of the collection of information on those who are to respond, including through the use of appropriate automated, electronic, mechanical, or other technology should be addressed to: Office of Information and Regulatory Affairs of OMB, Attention: Desk Officer for National Science Foundation, 725—17th Street, NW., Room 10235, Washington, DC 20503, and to Suzanne H. Plimpton, Reports Clearance Officer, National Science Foundation, 4201 Wilson Boulevard, Suite 295, Arlington, Virginia 22230 or send E-mail to *splimpto@nsf.gov.* Comments regarding these information collections are best assured of having their full effect if received within 30 days of this notification. Copies of the submission(s) may be obtained by calling 703–292–7556.

NSF may not conduct or sponsor a collection of information unless the collection of information displays a currently valid OMB control number and the agency informs potential persons who are to respond to the collection of information that such persons are not required to respond to the collection of information unless it displays a currently valid OMB control number.

Title: ''eRecruitment'' System.

OMB Control Number: 3145–NEW.

Summary of Collection:

Use of the Information: The information will be used by NSF to provide applicants with the ability to apply electronically for NSF positions and receive notification as to their qualifications, application dispensation and to request to be notified of future vacancies for which they may qualify.

In order to apply for vacancies, applicants will be required to submit certain data in order to receive consideration. Users only need access to the Internet for this system to work. This information will be used to determine which applicants are best qualified for a position, based on applicant responses to a series of job related ''yes/no'' or ''multiple choice'' questions. The resume portion requires applicants to provide the same information they would provide were they submitting a paper OF–612. The obvious benefit being that the applicant may do so on-line, 24 hours a day/seven days a week and receive electronic notification about the status of their application or information on other vacancies for which they may qualify. Staff members of the Human Resource Division and the selecting official(s) for specific positions for which applicants apply are the only ones privy to the applicant data. The most significant data is not the applicant personal data such as address or phone number but rather their description of their work experience and their corresponding responses to those questions, which determine their overall rating, ranking, and referral to the selecting official.

Estimate of Burden: Public reporting burden for this collection of information is estimated to average less than 30 to 45 minutes to create the on line resume and potentially less than 10 to 15 minutes to apply for jobs on-line.

Respondents: Individuals. Approximately 4800 applicants apply for NSF vacancies a year. This number could potentially double based on evidence from other agencies that use electronic recruitment systems; the estimated number of responses is 6500.

Estimated Number of Responses: Approximately 25 responses per job opening.

Estimated Total Annual Burden on Respondents: Approximately 45 minutes per respondent total time is all that will be needed to complete the on-line application, for a total of 4,875 hours annually.

Frequency of Responses: Applicants need only complete the resume one time, and they may use that resume to apply as often as they wish for any NSF job opening.

Dated: December 6, 2001.

Suzanne H. Plimpton,

Reports Clearance Officer.

[FR Doc. 01–30659 Filed 12–11–01; 8:45 am]

BILLING CODE 7555–01–M

NEIGHBORHOOD REINVESTMENT CORPORATION

Sunshine Act Meeting

TIME & DATE: 2 PM, Monday, December 17, 2001.

PLACE: Neighborhood Reinvestment Corporation, 1325 G Street, NW, Suite 800, Washington, DC 20005.

STATUS: Open/Closed.

CONTACT PERSON FOR MORE INFORMATION: Jeffrey T. Bryson, General Counsel/Secretary, 202–220–2372.

AGENDA:

I. Call to Order
II. Approval of Minutes: September 21, 2001 Regular Meeting
III. Treasurer's Report
IV. Strategic Plan Adoption
V. Executive Directors Quarterly Management Report
VI. Executive Session (Closed)
 (A) Personnel Committee Report—11/14/01
 (B) Personnel Committee Report—12/04/01
VII. Adjournment

Jeffrey T. Bryson,

General Counsel/Secretary.

[FR Doc. 01–30853 Filed 12–10–01; 3:48 pm]

BILLING CODE 7570–01–M

NUCLEAR REGULATORY COMMISSION

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

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

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 November 19, 2001 through November 30, 2001. The last biweekly notice was published on November 28, 2001 (66 FR 59498).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 11, 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, which is available at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by email to pdr@nrc.gov.

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

Date of amendment request: July 5, 2001.

Description of amendment request: The proposed amendment would relax Technical Specification (TS) operability requirements for primary containment systems, secondary containment systems, and the standby gas treatment system during the movement of irradiated fuel and during core alterations.

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

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

The equipment affected by the proposed changes are mitigative in nature, and relied upon after an accident has been initiated. Application of the Alternative Source Term (AST) does not involve a change to the plant design. While the operation of the primary and secondary containment systems do change as a result of these proposed changes, these systems are not accident initiators. Application of the AST does not initiate a design basis accident. Similarly, application of the AST does not affect the design or operation for any equipment or systems involved in the mitigation of accidents. The proposed changes to the Technical Specifications (TS), while they revise certain performance requirements, do not involve any physical modifications to the plant. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. As such, removal of operability requirements during the specified conditions will not significantly increase the probability of occurrence for an accident previously analyzed.

The AST changes do not affect the design and operation of the facility. Rather, once the accident has been postulated the new source term is an input to the evaluation of the consequences. The implementation of the AST has been evaluated in revisions to the analyses of the worst case Fuel Handling Accident (FHA) at Clinton Power Station (CPS). Based on the results of the analyses, it has been demonstrated that, with the proposed changes, the dose consequences of the worst case FHA remain a small fraction of the regulatory guidance provided by the NRC for the AST in RG [regulatory guide] 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' dated July 2000. Since the primary containment systems, secondary containment systems and the Standby Gas Treatment (SGT) are not assumed to be operable in the FHA, the consequences of eliminating the requirements that these systems be operable during the handling of irradiated fuel in both primary and secondary containment or during core alterations will not increase significantly.

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

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

No new equipment is introduced, and no installed equipment is operated in a new or different manner. There is no change to the predicted accident response of any plant structure, system or component. The proposed change in availability of mitigative equipment has been evaluated in accordance with the guidance in RG 1.183 and does not produce different or more limiting accident progression or results. As such, no new accident modes or equipment failure modes are created by these proposed changes.

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

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

The proposed changes involve a selective application of the AST for the FHA consistent with the guidance provided in RG 1.183. The existing analyses demonstrated that the dose consequences associated with the FHA were within the applicable NRC specified limits. For offsite dose, the margin to safety for the FHA using the 10 CFR 100, ''Reactor Site Criteria,'' limits was maintained by the existing analysis. For the Control Room dose, the margin of safety using the 10 CFR 50, ''Domestic Licensing of Production and Utilization Facilities,'' Appendix A, ''General Design Criteria for Nuclear Power Plants,'' General Design Criteria 19, ''Control room,'' dose limits was conservatively maintained by the existing analyses. The results of the FHA analysis revised in support of this submittal however, are subject to revised acceptance criteria. The revised dose consequences of the limiting design basis FHA are within the acceptance criteria found in RG 1.183 and 10 CFR 50.67, ''Domestic Licensing of Production and Utilization Facilities, Accident Source Term.'' The proposed changes ensure that the doses at the exclusion area boundary (EAB), low population zone (LPZ), and control room remain a small fraction of the new regulatory limits in RG 1.183 and 10 CFR 50.67.

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

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

Attorney for licensee: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Anthony J. Mendiola.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: October 31, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the technical specifications (TS) for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 31, 2001.

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 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 radionuclides 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: October 31, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the technical specifications (TS) for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR

49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 31, 2001.

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 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 radionuclides 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: May 24, 2001.

Description of amendment request: The proposed amendment would delete License Condition 2.C.(11), which is no longer applicable to the facility. License Condition 2.C.(11) requires inspection of the low-pressure turbine discs during the second refueling outage, including volumetric examination of the disc base using ultrasonic techniques, and specifies that the frequency of subsequent inspections shall be in accordance with the turbine manufacturer's recommendations. The amendment request states that the license condition is no longer applicable for the following reasons: (1) the initial inspection was completed during the second refueling outage as required; and (2) during fifth refueling outage, the low-pressure turbine rotors were replaced with monoblock designed rotors that do not utilize shrunk-on discs, and therefore the subsequent inspections specified in License Condition 2.C.(11) for shrunk-on discs would be meaningless with the new rotor design. The licensee's inspection and maintenance program for the new low-pressure turbine is based on the current turbine manufacturer's recommendations for the monoblock design.

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 removes Fermi 2 Operating License Condition 2.C.(11) which details the inspection frequency of the low-pressure (LP) turbine discs. The inspection frequency was recommended because the original turbine rotor design involved a shrunk-on disc configuration. The inspection attributes applied specifically to this disc design and were intended to enhance design reliability. In 1996, however, the LP turbine steam path consisting of rotors, buckets (blades), diaphragms and steam flow guides, all manufactured by English Electric Co., were replaced with General Electric (GE) components. In particular, the GE design does not utilize shrunk-on discs; it includes rotors of monoblock construction, thus negating the applicability of License Condition 2.C.(11). There are no relevant aspects of the previously recommended inspections that apply to the new monoblock construction.

Section 3.5.1.2.1 of the Fermi 2 UFSAR [Updated Final Safety Analysis Report] addresses the potential for missiles generated from rotating equipment including those generated from a low-pressure turbine rotor segment. Section 10.2.3 of the UFSAR states that following the low-pressure turbine rotor replacement during RFO05, ''there will no longer be a design basis turbine missile at Fermi 2.'' Section 3.5.1.2.2 further states, ''The new low-pressure rotors are of monoblock construction. The monoblock rotors have higher speed capability than the maximum attainable speed of the turbine generator units. Per General Electric, the supplier of the new rotors, the probability of missiles being generated is well below 10 to the –8 power.'' There are no other postulated accidents that were directly attributable to the English Electric Company shrunk-on disc design; therefore, the removal of License Condition 2.C.(11) does not increase the probability of occurrence or the 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 change removes License Condition 2.C.(11) because it is no longer applicable to the design of the low-pressure turbine currently installed at the facility. Therefore, removal of the license condition affects neither the design nor the operation of the plant. It cannot create a new failure mode, nor can its removal 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.

License Condition 2.C.(11) is not applicable to the facility because the low-pressure turbine rotor was replaced with a design which does not include shrunk-on turbine discs. This rotor replacement eliminated the potential for a design basis accident resulting from the turbine missiles at Fermi 2, which was the accident scenario that the inspections referenced in License Condition 2.C.(11) were intended to prevent. Since the license condition no longer applies to the current facility design, and the potential design basis accident associated with the license condition no longer exists, the removal of the license condition will not reduce 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Acting Section Chief: William D. Reckley, Acting.

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

Date of amendment request: November 11, 2001.

Description of amendment request: A change is proposed to Technical Specification 3.0.3 to allow a longer period of time to perform a missed surveillance. The time 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 would be added to the specification: ''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 November 11, 2001.

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 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.

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: William D. Reckley, Acting.

Dominion Nuclear Connecticut, Inc., et al., Docket Nos. 50–245, 50–336, and 50–423, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, New London County, Connecticut

Date of amendment request: November 8, 2001.

Description of amendment request: The proposed amendments would incorporate administrative and editorial changes into the Millstone Unit No. 1 Permanently Defueled Technical Specifications (PDTS) and into the Millstone Unit Nos. 2 and 3 Technical Specifications (TSs). Specifically, the proposed changes would: (1) Relocate redundant design features information already included in other licensing basis (LB) documents (e.g., the Final Safety Analysis Report (FSAR)), from Section 5.0, ''Design Features,'' of the Unit Nos. 2 and 3 TS, to other LB documents, consistent with the improved Standard Technical Specifications (STSs) for the respective unit design; (2) revise TS 5.6.2, ''Technical Specifications Bases Control Program,'' in the Unit No. 1 PDTS to incorporate the 10 CFR 50.59 rule change; (3) add a new TS (TS 6.22 for Unit No. 2 and TS 6.17 for Unit No. 3), to incorporate a TS bases control program within the Unit Nos. 2 and 3 TS; (4) add a new TS (TS 6.18, ''Component Cyclic or Transient Limits''), to the Unit No. 3 TS to define the program for tracking cyclic (or transient) limits. These limits are proposed to be relocated from where they are listed in TS 5.7, ''Component Cyclic or Transient Limit,'' in the Unit No. 3 TS, to the FSAR; (5) revise the Unit No. 1 PDTS and the Unit Nos. 2 and 3 TS related to Radiological Environmental Monitoring Program (REMP) procedure processing to: (a) remove reference to an organization affiliated with Northeast Utilities (NU), the Production Operations Services Laboratory, which is no longer applicable following the change in ownership from NU to Dominion Nuclear Connecticut (DNC); (b) replace the reference to the Radiological Assessment Branch (a Millstone DNC organization) with the ''organization responsible for the REMP'' for review/approval of changes to the REMP to avoid future TS changes due to a change in organizational titles; (c) correct an inconsistency within the Unit No. 1 PDTS which implies that REMP procedures are processed under the general procedure processing specification (*i.e.,* TS 5.5.1), in addition to the specific specifications for processing REMP procedure changes (*i.e.,* Specifications 5.5.6 and 5.5.7); and (6) correct miscellaneous editorial issues and achieve consistency between the TSs for each unit. These changes include: (a) Changes to and corrections in titles; (b) correct references to the quality assurance program, and (c) change titles to utilize the term radiation protection rather than health physics.

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

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

The proposed changes related to Section 5, ''Design Features,'' of the Unit Nos. 2 or 3 TS either relocates or deletes certain details from the Technical Specifications and relocates them to the respective unit's updated FSAR or other plant controlled documents. The FSAR and other plant controlled documents will be maintained in accordance with 10 CFR 50.59. The proposed changes to Section 6, ''Administrative Controls,'' adds new administrative specifications consistent with the guidance of the improved STS, corrects inconsistencies, or represents changes in nomenclature, and will correct editorial issues and achieve consistency within the individual TS and between individual TS. The changes are purely administrative or editorial and do not alter any regulatory requirements or have an impact on the acceptance criteria for any design basis accident described in the respective Unit Nos. 2 or 3 FSAR or the Unit No. 1 Defueled Safety Analysis Report (DSAR).

These changes have no impact on plant equipment operation. Since the changes are solely an administrative or editorial change to the TS, they cannot affect the likelihood or consequences of accidents. Therefore, these changes will 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 have no impact on plant operation. Since the proposed changes are solely an administrative or editorial change to the TS, they do not affect plant operation in any way. The proposed changes do not involve a physical alteration of the plant or change the plant configuration (no new or different type of equipment will be installed). The proposed changes do not require any new or unusual operator actions. The changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The changes do not introduce any new failure modes. Therefore, the proposed changes will 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.

Since the proposed changes are solely administrative or editorial changes to the TS, they do not affect plant operation in any way. The proposed changes to the respective unit's technical specifications will standardize terminology, remove extraneous information and make minor format changes that will not result in any technical changes to current requirements.

The proposed changes do not impact any acceptance criteria for the design basis accidents described in the respective Unit Nos. 2 or 3 FSAR or the Unit No. 1 DSAR and do not impact the consequences of accidents previously evaluated. Therefore, the proposed changes will not result in a reduction in a margin of safety.

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

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, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: May 25, 2001.

Description of amendment request: The amendments would revise Technical Specifications (TS) Definitions for ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME and REACTOR TRIP SYSTEM (RTS) RESPONSE TIME to provide for verification of response time for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the Nuclear Regulatory Commission. The associated Bases will also be revised. The licensee has referenced previously approved WCAP–13632–P–A, Revision 2, ''Elimination of Pressure Sensor Response Time Testing Requirements,'' and WCAP–14036–P–A Revision 1, ''Elimination of Periodic Protection Channel Response Time Tests'' as the justifications for proposing 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, which is presented below:

Conformance of the proposed amendments to the standards for a determination of no significant hazards as defined in 10 CFR 50.92 is shown in the following:

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

This change to the TS does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS and ESFAS instrumentation is being used; the time response allocations/modeling assumptions in the UFSAR Chapter 15 analyses are still the same; only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity 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 amendments do not result in any increase in the probability or consequences of an accident previously evaluated.

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

This change does not alter the performance of the reactor protection system (RPS) or the engineered safety features actuation system (ESFAS). All RPS and ESFAS channels will still have response time verified by test before 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 ESFAS 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 amendments does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will detect any degradation which might significantly affect channel response time. Based on the above, it is concluded that the proposed license amendment request does not result in a reduction in a margin with respect to plant safety.

Based on the preceding analysis, it is concluded that elimination of periodic [response time testing] RTT is acceptable and the proposed license amendments do not involve a significant hazards consideration finding as defined in 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: Ms. Lisa F. Vaughn , Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: August 6, 2001.

Description of amendment request: The amendments would revise Technical Specifications (TS) 3.3.2 for engineered safety feature actuation system instrumentation, TS 3.3.6 for containment purge and exhaust isolation instrumentation. The amendments would also revise the appropriate bases, and the bases for Containment Isolation Valves (TS 3.6.3). Specifically, the proposed amendments would modify the TS requirements so that they exclude the Containment Purge Ventilation System and the Hydrogen Purge System, thereby applying the requirements to only the Containment Air Release and Addition System. At Catawba, the containment isolation valves for the Containment Purge Ventilation System and the Hydrogen Purge System are sealed closed in the modes of applicability (Modes 1, 2, 3, and 4) according to TS requirements.

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

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

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

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

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

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Neither the Containment Purge Ventilation System, the Hydrogen Purge System, nor the Containment Air Release and Addition System is capable of by itself initiating any accident. The safety related portions of these systems, which are responsible for maintaining containment isolation during accident conditions, will continue to function as designed, and in accordance with all applicable TS. The design and operation of the systems are not being modified by this proposed amendment. Therefore, there will be no impact on any accident probabilities or consequences.

Second Standard

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators and does not impact any safety analyses.

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed amendment. It has already been shown that the performance of all containment isolation functions in response to accident conditions will not be impacted by this proposed amendment. There is no risk significance to this proposed amendment, as no reduction in system or component availability will be incurred. No safety margins will be impacted.

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

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

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

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: September 24, 2001.

Description of amendment request: The amendment request proposes to extend the allowed outage time for a Division I or Division II Emergency Diesel Generator (EDG) from 72 hours to 14 days. The proposed changes are intended to provide flexibility in scheduling EDG maintenance activities, reduce refueling outage duration, and improve EDG availability during plant shutdowns.

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

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

The proposed Technical Specification (TS) changes do not affect the design, operational characteristics, function, or reliability of the EDGs. The EDGs are not the initiators of previously evaluated accidents. The EDGs are designed to mitigate the consequences of previously evaluated accidents including a loss of offsite power. Extending the allowed outage time (AOT) for a single EDG would not significantly affect the previously evaluated accidents since the remaining EDGs supporting the redundant ESF [Engineered Safety Feature] systems would continue to perform the accident mitigating functions as designed.

The duration of a TS AOT is determined considering that there is a minimal possibility that an accident will occur while a component is removed from service. A risk-informed assessment was performed which concluded that the increase in plant risk is small and consistent with the USNRC [United States Nuclear Regulatory Commission (NRC)] ''Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,'' **Federal Register**, Vol. 51, p.30028 (51 FR 30028), August 4, 1986, as further described by NRC Regulatory Guide 1.177.

The current TS requirements establish controls to ensure that redundant systems relying on the remaining EDGs are Operable. In addition to these requirements, administrative controls will be established to provide assurance that the AOT extension is not applied during adverse weather conditions that could potentially affect offsite power availability.

Both the RBS [River Bend Station, Unit 1] risk-based analysis and the deterministic evaluation support the increased AOT. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed TS changes do not involve a change in the design, configuration, or method of operation of the plant that could create the possibility of a new or different kind of accident. The proposed change extends the AOT currently allowed by the TS.

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

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

The proposed extended AOT is not in conflict with any of the approved codes and standards applicable to the onsite AC [Alternating Current] power sources. The proposed changes do deviate from the recommendations of Regulatory Guide (RG) 1.93. An extension of the 72 hour AOT recommended in the RG to 14 days is demonstrated herein to be acceptable and has been approved for several other licensees. Assuming there are no additional failures of redundant equipment during the time that the EDG is removed from service, the intended safety functions would still be met.

The proposed AOT change does not affect any of the assumptions or inputs to the safety analyses of the FSAR [Final Safety Assessment Report] and does not erode the decrease in severe accident risk achieved with the issuance of the Station Blackout (SBO) Rule, 10 CFR 50.63 ''Loss of All Alternating Current Power''. RBS is classified as a four-hour coping plant with 0.95 EDG reliability (see U[pdated] FSAR Appendix 15C). The assumptions used in the SBO [Station Blackout] analysis regarding reliability of the EDGs are unaffected by the proposed TS changes since preventive maintenance and testing will continue to be performed to maintain reliability assumptions.

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

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit Number 3, Westchester County, New York

Date of amendment request: October 23, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.10, ''Ventilation Filter Testing Program,'' to adopt the requirements of the American Society for Testing and Materials Standard (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.'' The proposed TS revisions are in response to Nuclear Regulatory Commission (NRC) Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The proposed amendment would also revise the differential pressure criteria for the test of the filter system for the Control Room Ventilation System.

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

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

The proposed license amendment adopts the new test method and acceptance criteria of ASTM D3803–1989, with the exceptions identified, for activated charcoal filters and changes the allowable pressure differential for Control Room ventilation. The changes require laboratory performance testing of adsorber carbon that yields a more accurate result than the testing currently required by the TS and requires a more stringent limit on the Control Room ventilation pressure differential. The proposed change to delete non-conservative TS requirements for testing of adsorber carbon and limiting the Control Room ventilation differential pressure are not plant accident initiators as described in the Final Safety Analysis Report (FSAR). The proposed amendment does not change the function of any structure, system or component (SSC). The function of the ventilation systems is filtration of radiological releases during postulated accidents. The proposed changes will provide greater assurance that this function is provided. The revised TS requirements are for laboratory tests and pressure differential measurements that are currently in place and change only the TS testing requirements. They will not result in any changes to the efficiency assumed in accident analysis. The changes do not alter, degrade or prevent actions described or assumed in an accident described in the FSAR. Therefore, the proposed amendment does not change the possibility of an accident previously evaluated or significantly increase the consequences of an accident previously evaluated.

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

The proposed license amendment adopts the new test method and acceptance criteria of ASTM D3803–1989, with the exceptions identified, for activated charcoal filters and changes the allowable pressure differential for Control Room ventilation. The change does not involve any modifications to the plant, will not require changes to how the plant is operated nor will it affect the operation of the plant. The changes require laboratory performance testing of adsorber carbon that yields a more accurate result than the testing currently required by the TS and requires a more stringent limit on the Control Room ventilation pressure differential. The proposed changes to delete non-conservative TS requirements for testing of adsorber carbon and limiting the Control Room ventilation differential pressure are not plant accident initiators as described in the Final Safety Analysis Report (FSAR). The proposed amendment does not change the function of any structure, system or component (SSC). The function of the ventilation systems is filtration of radiological releases during postulated accidents. The proposed changes will provide greater assurance that this function is provided. The revised TS requirements are for laboratory tests and pressure differential measurements that are currently in place and change only the TS testing requirements. They will not result in any changes to the efficiency assumed in accident analysis. The changes do not alter, degrade or prevent actions described or assumed in an accident described in the FSAR. Therefore, 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 license amendment involve a significant reduction in a margin of safety?*

The proposed license amendment adopts the new test method and acceptance criteria of ASTM D3803–1989, with the exceptions identified, for activated charcoal filters and changes the allowable pressure differential for Control Room ventilation. The proposed license amendment does not reduce the margin of safety but enhances by requiring more accurate testing and a more conservative pressure differential. The proposed test change will require the use of a current and improved ASTM standard to ensure that the carbon ability to adsorb radioactive material will remain at or above the capability credited in our accident analysis. The proposed differential pressure limit will assure that the system provides sufficient flow though the charcoal to meet accident analyses.

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: L. Raghavan (Acting).

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: October 23, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 23, 2001.

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 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 radionuclides 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 [a] 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 [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.

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: L. Raghavan (Acting).

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

Date of amendment request: September 28, 2001.

Description of amendment request: The licensee proposes to revise a single Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip Reactor Pressure High setpoint to replace the current conditional setpoints which are based upon the number of Safety Relief Valves out of service.

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

1. Involve a significant increase in the probability or consequences of an accident previously evaluated because: a change in the ATWS high RPV [reactor pressure vessel] pressure RWR [ATWS Reactor Pressure High Recirculation Pump] pump trip setpoint does not affect initiation of any accident. Operation in accordance with the revised setpoint ensures the consequences of previously analyzed accidents are not changed.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because: RPV pressure following an ATWS with PRFO [Pressure Regulating Valve Open] event (worst case transient for RPV pressurization) remains within acceptable limits with the revised setpoint. Therefore, changing the setpoint will not lead to a new or different kind of accident.

3. Involve a significant reduction in a margin of safety because: the analyses performed to determine the revised ATWS high pressure RWR pump trip setpoint assure maintenance of the same margin of safety as presently exists for limiting RPV pressure following an ATWS with PRFO (limiting transient). The current analyses actually shows an improved margin over the results of the previous analyses (References 2 and 3), which were performed using an earlier computer code.

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

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

NRC Section Chief: L. Raghavan (Acting).

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

Date of amendment request: October 30, 2001.

Description of amendment request: The license amendment request proposes changes to Arkansas Nuclear One, Unit 2 (ANO–2) Technical Specification (TS) 3.4.9, ''Pressure/Temperature Limits,'' and TS 3.4.12, ''Low Temperature Overpressure Protection (LTOP) System.'' The primary changes are to update the existing pressure/temperature (P/T) limits from 21 to 32 effective full power years (EFPYs) and to include additional restrictions in the LTOP TSs.

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—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The probability of occurrence of an accident previously evaluated for ANO–2 is not altered by the proposed amendment to the technical specifications (TSs). The accidents remain the same as currently analyzed in the ANO–2 Safety Analysis Report (SAR) as a result of changes to the P/T limits as well as those for LTOP. The new P/T and LTOP limits were based on NRC [Nuclear Regulatory Commission] accepted methodologies along with ASME [American Society of Mechanical Engineers] Code [Boiler and Pressure Vessel Code] alternatives. The proposed changes do not impact the integrity of the reactor coolant pressure boundary (RCPB) (i.e. there is no change to the operating pressure, materials, loadings, etc.) as a result of this change. In addition, there is no increase in the potential for the occurrence of a loss of coolant accident. The probability of any design basis accident is not affected by this change, nor are the consequences of any design basis accident (DBA) affected by this proposed change. The proposed P/T limit curves and the LTOP limits are not considered to be an initiator or contributor to any accident currently evaluated in the ANO–2 SAR. These new limits ensure the long term integrity of the RCPB.

Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life. A new reactor vessel specimen capsule was withdrawn at the most recent refueling outage and was analyzed to predict the fracture toughness requirements using projected neutron fluence calculations. For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline, inlet nozzle, outlet nozzle, and closure head.

The predicted radiation induced ΔRT_{NDT} [shift in reference temperature for nil-ductility transition] was calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 32 EFPY including an estimated increase in flux due to a proposed power uprate. The ΔRT_{NDT} [reference temperature for nil-ductility transition] and, in turn, the operating limits for ANO–2 were adjusted to account for the effects of irradiation on the fracture toughness of the reactor vessel materials. Therefore, new operating limits are established which are represented in the revised operating curves for heatup/criticality, cooldown and inservice hydrostatic testing contained in the technical specifications.

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

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

The proposed changes to the P/T and LTOP limits will not create a new accident scenario. The requirements to have P/T and LTOP protection are part of the licensing basis of ANO–2. The proposed changes reflect the change in vessel material properties acknowledged and managed by regulation and the best data available in response to NRC Generic Letter 92–01, Revision 1. The approach used meets NRC and ASME regulations and guidelines. The calculational methodology for fluence is based on an NRC approved Framatome ANP approach. Therefore, the adjusted reference temperatures for fracture toughness are consistent with that previously provided to the NRC. The data analysis for the vessel specimen removed at 2R14 (approximately 15.7 EFPY of exposure) confirms that the vessel materials are responding as predicted.

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

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

The existing P/T curves and LTOP limits in the technical specifications are reaching their expiration period for the number of years at effective full power operation. The revision of the P/T limits and curves will ensure that ANO–2 continues to operate within the operating margins allowed by 10 CFR 50.60 and the ASME Code. The material properties used in the analysis are based on results established through CE [Combustion Engineering] material reports for copper and nickel content. The application of ASME Code Case N–641 presents alternative procedures for calculating P/T and LTOP temperatures and pressures in lieu of that established for ASME Section XI, Appendix G–2215. This Code alternative allows certain assumptions to be conservatively reduced. However, the procedures allowed by Code Case N–641 still provide significant conservatism and ensure an adequate margin of safety in the development of P/T operating and pressure test limits to prevent non-ductile fractures.

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

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

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

NRC Section Chief: Robert A. Gramm.

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of amendment request: November 19, 2001.

Description of amendment request: The proposed amendment would to eliminate restrictions imposed by technical specification (TS) 3.0.4 for the Remote Shutdown Instrumentation.

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 of occurrence or consequences of an accident previously evaluated?

Response: No.

Probability of Occurrence of an Accident Previously Evaluated

The Remote Shutdown Instrumentation system ensures that sufficient capability is available to permit shutdown and maintenance of Hot Standby of the plant from locations outside of the control room. The proposed change allows Unit 1 to ascend in mode without meeting the LCO [limiting condition for operation] for TS 3.3.3.5. The proposed change does not impact the ability to comply with the allowed outage time (AOT) described in TS 3.3.3.5. As such, the proposed change does not affect any accident initiators or precursors, since the AOT for TS 3.3.3.5 will continue to be met. The proposed change is also consistent with the Unit 2 TS. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased.

The format changes do not impact any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated

The proposed change to allow Unit 1 to ascend in mode without meeting the LCO for TS 3.3.3.5, while continuing to meet the action statement, will not significantly impact the Remote Shutdown Instrumentation systems' capability of performing its design function. The Remote Shutdown Instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of Hot Standby of the plant from locations outside of the control room. The proposed change does not impact the ability to comply with AOT described in TS 3.3.3.5. The proposed change is also consistent with the Unit 2 TS. Thus, there will be no increase in offsite doses, and the consequences of an accident previously analyzed are not increased.

The format changes do not impact the function of the Remote Shutdown Instrumentation. Thus, there will be no increase in offsite doses, and the consequences of an accident previously analyzed are not significantly increased.

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

Response: No.

The Remote Shutdown Instrumentation system ensures that sufficient capability is available to permit shutdown and maintenance of Hot Standby of the plant from locations outside of the control room. Allowing Unit 1 to ascend in mode without meeting the LCO for TS 3.3.3.5, while continuing to meet the action statement, does not change the function of the Remote Shutdown Instrumentation system or create the possibility of a new or different type of accident. The proposed change does not impact the ability to comply with the AOT described in TS 3.3.3.5. The proposed change is also consistent with the Unit 2 TS. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

Response: No.

The proposed change does not impact the Remote Shutdown Instrumentation system's capability of performing its design function, nor does the proposed change impact the operational characteristics of the Remote Shutdown Instrumentation system. The Remote Shutdown Instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of Hot Standby of the plant from locations outside of the control room. Allowing Unit 1 to ascend in mode without meeting the LCO for TS 3.3.3.5, while continuing to meet the action statement, does not impact CNP's accident analysis. The proposed change is also consistent with the Unit 2 TS. 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 requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: William D. Reckley, Acting.

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

Date of amendment requests: October 12, 2001.

Description of amendment requests: The proposed amendments would delete requirements from the technical specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 12, 2001.

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

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 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 radionuclides 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: William D. Reckley, Acting.

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

Date of amendment requests: November 1, 2001.

Description of amendment requests: The proposed amendments would revise technical specification (TS) surveillance requirements (SR) 4.8.2.3.2.c.2 and 4.8.2.5.2.c.2 and associated TS bases concerning the safety-related batteries to make them more consistent with the Westinghouse Standard TSs.

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

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

Probability of Occurrence of an Accident Previously Evaluated

The proposed change to SRs 4.8.2.3.2.c.2 and 4.8.2.5.2.c.2 to add a requirement to remove visible corrosion and to delete the requirement that the battery be free of corrosion does not affect any accident initiators or precursors. The batteries perform a mitigating function following a loss of AC power, and the presence of corrosion will not adversely impact components whose failure would initiate an accident. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

The proposed change to the TS 3/4.8 bases provides clarification and does not affect any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

The proposed change to SRs 4.8.2.3.2.c.3 and 4.8.2.5.2.c.3 increases the battery charger current required during surveillance testing. The required value is within the capability of the battery charger. Thus, the battery charger is not degraded by this change, and the change does not affect any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

The proposed changes to SR 4.8.2.3.2.d delete the requirement that the battery terminal voltage be maintained greater than or equal to 210 volts during the battery service test, and delete the description of the composite load profile. The removal of the requirement and the description from the SR do not affect any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

The deletion of Tables 4.8–2 and 4.8–3, the incorporation of the words ''this page intentionally left blank,'' and the deletion of the SR 4.8.2.3.2.d and SR 4.8.2.5.2.d references to the tables do not impact battery operation as the tables summarize information used as calculation inputs. These changes do not affect any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

The proposed changes to SR 4.8.2.5.2.d to delete the requirement that the battery terminal voltage be maintained greater than or equal to 210 volts during the battery service test, and to add the term ''design duty cycle'' does not affect any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

The editorial change does not impact any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated

The batteries and their associated chargers provide power to emergency equipment that is used in the mitigation of accidents. The batteries provide power to this equipment following a loss of AC power until the battery chargers are powered by the emergency diesel generators.

The proposed change to SRs 4.8.2.3.2.c.2 and 4.8.2.5.2.c.2 to add a requirement to remove visible corrosion and to delete the requirement that the battery connections be free of corrosion does not impact a battery's capability to power its safety-related loads as the presence of corrosion at the terminal connections does not indicate that the battery is unable to perform its function. Rather, it is the impact of the corrosion on the connections that is of concern. This concern will be addressed by performing a resistance check to verify that battery performance is acceptable. Therefore, this change does not result in an increase in offsite doses. Thus, the consequences of an accident previously analyzed are not increased.

The proposed change to the TS 3/4.8 bases provides clarification and does not impact the battery's capability to power its safety-related loads. Thus, the consequences of an accident previously analyzed are not increased.

The proposed change to SRs 4.8.2.3.2.c.3 and 4.8.2.5.2.c.3 to increase the required battery charger current ensures that the battery charger has sufficient capacity to provide power to emergency equipment while simultaneously recharging batteries that were discharged following a loss of AC power. This ensures that emergency equipment connected to the battery will continue to operate as designed, and offsite doses will not be increased. Thus, the consequences of an accident previously analyzed are not increased.

The proposed changes to SR 4.8.2.3.2.d delete the requirement that the battery terminal voltage be maintained greater than or equal to 210 volts during the battery service test, and delete the description of the composite load profile. However, the SR will still require that the service test demonstrate that the battery capacity is adequate to supply emergency loads. The voltage requirements for the batteries are determined by battery-system specific calculations, and the calculation results are incorporated into the test procedures. This assures that the equipment connected to the battery will continue to operate as designed, and offsite doses will not be increased. Thus, the consequences of an accident previously analyzed are not increased.

The deletion of Tables 4.8–2 and 4.8–3, the addition of the words ''this page intentionally left blank,'' and the deletion of the SR 4.8.2.3.2.d and SR 4.8.2.5.2.d references to the tables do not impact battery operation as the tables summarize information used as calculation inputs. The batteries are tested to a load profile that is developed on the basis of the battery loads for a loss of AC power, and the testing assures that the batteries are capable of performing their safety function. Thus, these changes will not impact battery capability, will not result in an increase in offsite doses, and the consequences of an accident previously analyzed are not increased.

The proposed changes to SR 4.8.2.5.2.d to delete the requirement that the battery terminal voltage be maintained greater than or equal to 210 volts during the battery service test, and to add the term ''design duty cycle'' requires that the battery be tested in accordance with a load profile developed on the basis of the battery loads for a loss of AC power. The testing of the battery assures that it is capable of performing its safety function. Thus, the capability of the battery is not impacted, there will be no increase in offsite doses, and the consequences of an accident previously analyzed are not increased.

The editorial change does not impact battery capability. Thus, there will be no increase in offsite doses, and the consequences of an accident previously analyzed are not increased.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not increased.

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

The batteries perform a mitigating function by providing power to emergency equipment following a loss of AC power.

The proposed change to SRs 4.8.2.3.2.c.2 and 4.8.2.5.2.c.2 adds a requirement to remove visible corrosion and deletes the requirement that the battery terminals be free of corrosion. The presence of corrosion on the battery terminals does not introduce a mechanism that would cause a plant transient, and I&M will ensure that the corrosion does not impact the battery's function. Thus, the possibility of a new or different kind of accident is not created.

The proposed change to the TS 3/4.8 bases provides clarification and does not introduce a mechanism that would cause a plant transient. Thus, the possibility of a new or different kind of accident is not created.

The proposed change to SRs 4.8.2.3.2.c.3 and 4.8.2.5.2.c.3 increases the acceptance criterion for battery charger current to reflect the present demand on the battery charger when it is simultaneously supplying power to emergency equipment and charging a discharged battery. The increase in the acceptance criterion is within the capability of the battery charger, and no failure mechanisms are introduced by this change. Thus, the change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes to SR 4.8.2.3.2.d to delete the requirement that the battery terminal voltage be maintained greater than or equal to 210 volts during a battery service test, and to delete the load profile description do not directly impact any emergency equipment as the SR continues to require that the battery service test demonstrate that the battery is capable of supplying power to connected equipment, and this change does not introduce any battery failure mechanisms. Thus, the change does not create the possibility of a new or different kind of accident from any previously evaluated.

The deletion of Tables 4.8–2 and 4.8–3, the incorporation of the words ''this page intentionally left blank,'' and the deletion of the SR 4.8.2.3.2.d and SR 4.8.2.5.2.d references to the tables do not impact battery operation as the tables summarize information used as calculation inputs. Thus, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes to SR 4.8.2.5.2.d to delete the requirement that the battery terminal voltage be maintained greater than 210 volts during a battery service test, and to add the term ''design duty cycle'' do not introduce any battery failure mechanisms as they do not alter the battery's physical characteristics or the battery testing requirements. Additionally, the term ''design duty cycle'' more accurately reflects the use of a simulated load for the battery test. Thus, the change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

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 change involve a significant reduction in a margin of safety?

The proposed changes do not impact the functional requirements of either the batteries or the battery chargers, nor do the changes impact the operational characteristics of the equipment that is connected to the battery. The batteries will continue to be subjected to a system test to verify that the battery capacity is adequate, and the battery chargers will be tested to verify that they are capable of meeting their rated capacity. These tests will demonstrate that the batteries and the battery chargers are capable of performing their mitigation function for analyzed accidents.

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

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: William D. Reckley, Acting.

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

Date of amendment requests: November 16, 2001.

Description of amendment requests: The proposed amendments would revise technical specification (TS) Table 3.3–4, ''Engineered Safety Feature Actuation System Instrumentation Trip Setpoints.'' The proposed changes are part of a planned design change to replace the existing 4kV offsite power transformers, loss of voltage relays, and degraded voltage relays with components of an improved design to increase the reliability of offsite power for safety-related equipment.

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

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

Probability of Occurrence of an Accident Previously Evaluated

The proposed changes to the degraded voltage and loss of voltage setpoints and time delay affect when an emergency bus that is experiencing low or degraded voltage will trip from offsite power and shift to an emergency diesel generator. While the setpoints that initiate this action will be modified, the function remains the same. The setpoints have been analyzed to ensure spurious trips will be avoided. The proposed changes will not significantly affect any accident initiators or precursors. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated

The proposed changes to the degraded voltage and loss of voltage setpoints and time delay affect when an emergency bus that is experiencing low or degraded voltage will trip from offsite power and shift to an emergency diesel generator. While the setpoints that initiate this action will be modified, they are bounded by the current safety analysis. The function of the plant equipment remains the same. The proposed changes improve the reliability of safety-related equipment to operate as designed. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the consequences of an accident previously analyzed are not significantly increased.

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

The proposed changes to the degraded voltage and loss of voltage setpoints and time delay do not affect existing or introduce any new accident precursors or modes of operation. The relays will continue to detect undervoltage conditions and transfer safety loads to the emergency diesel generators at a voltage level adequate to ensure proper safety equipment performance and to prevent equipment damage. The function of the relays remains the same. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will allow all safety-related loads to have sufficient voltage to perform their intended safety function while ensuring spurious trips are avoided. Thus, the results of the accident analyses will not be affected as the input assumptions are protected. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: William D. Reckley, Acting Section Chief.

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

Date of amendment request: August 2, 2001, as supplemented November 2, 2001.

Description of amendment request: The amendment would change the Seabrook Station Technical Specification (TS) 6.15 to permit a one-time exception to the 10-year frequency for the Integrated Leakage Rate Test (ILRT). This exception would permit the existing ILRT frequency to be extended from 10 years to 15 years from the last test completed on October 30, 1992.

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

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

The proposed change to the Seabrook Station Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously analyzed. The proposed revision to TS 6.15 adds a one-time extension to the current interval for the ILRT test. It is proposed that the current test interval be extended from ten-years to fifteen-years from the date of the last ILRT performed on October 30, 1992. The proposed extension cannot increase the probability of an accident previously evaluated since the test interval extension does not involve modification of the plant, nor a operation of the plant that could initiate an accident. The proposed extension of the ILRT does not involve a significant increase in the consequences of an accident. The increase in risk is very small because ILRTs identify only a few potential leakage paths that cannot be identified by local leakage rate [Type B and C] testing, and the leaks that have been found by ILRTs have been only marginally above existing requirements. An analysis of the 144 ILRT results including 23 failures, found that no ILRT failures were due to a containment liner breach. NUREG–1493 [''Performance-Based Containment Leak Test Program''] concluded that reducing the ILRT testing frequency to one per twenty years would lead to an imperceptible increase in risk.

Therefore, it is concluded that the proposed change to TS 6.15 does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed change to Technical Specification 6.15 does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change adds a one-time extension to the current Integrated Leakage Rate Test frequency of ten-years to fifteen-years from the date of the last test. The proposed change cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant. Additionally, there are no changes to the operation of the plant that could introduce a new failure mode creating an accident.

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

The proposed change does not involve a significant reduction in the margin of safety. The proposed revision to TS 6.15 adds a one-time extension to the current interval for the ILRT test. It is proposed that the current test interval be extended from ten-years to fifteen-years from the date of the last ILRT performed on October 30, 1992. A reduction in the ILRT frequency was found to lead to an imperceptible decrease in the margin of safety. The estimated increase in risk is very small because ILRTs identify only a few potential leakage paths that cannot be identified by local leakage rate [Type B and C] testing, and the leaks that have been found by ILRTs have been only marginally above existing requirements. A Seabrook Station specific risk evaluation is consistent with the generic conclusions identified in NUREG–1493.

Based on the above evaluation, North Atlantic concludes that the proposed change to TS 6.15 does not constitute a significant hazard.

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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 22, 2001.

Description of amendment request: The proposed amendment deletes requirements from the technical specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 22, 2001.

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 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 radionuclides 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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: William D. Reckley, Acting.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: October 16, 2001.

Description of amendment request: The proposed amendment would revise the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications (TSs). The licensee proposed to revise selected sections of the administrative controls chapter of the TSs consistent with Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF) generic changes to NUREG–1433, ''Standard Technical Specifications for General Electric Plants (BWR/4),'' Revision 1 (STS). The licensee also proposed editorial and administrative changes to the affected sections.

The licensee categorized the proposed changes as either ''Administrative Changes'' or ''Less Restrictive Changes—Removed Detail.'' The licensee categorized proposed changes consistent with the approved versions of TSTF–273, TSTF–299, TSTF–308, TSTF–348, and TSTF–364 as ''Administrative Changes.'' An administrative change involves editorial restructuring of the current requirements, or modification of wording that does not affect the technical content of the current TSs. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current TSs. The licensee categorized proposed changes consistent with the approved versions of TSTF–279 and TSTF–363 as ''Less Restrictive Changes—Removed Detail.'' The proposed changes involve moving details out of the TSs and into the TS Bases, the updated Final Safety Analysis Report (UFSAR), the Technical Requirements Manual (TRM), or other documents for which changes are subject to regulatory control. The removal of this information is considered to be less restrictive because it is no longer controlled by the TS change process.

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:

Administrative Changes

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

The proposed change involves reformatting, renumbering, and rewording the existing [technical specification] TS. The reformatting, renumbering, and rewording process involves no technical changes to the existing TS. As such, this change is administrative in nature and does not affect the initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this 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?*

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. 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 change involve a significant reduction in a margin of safety?*

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. Therefore, the change does not involve a significant reduction in a margin of safety.

Less Restrictive Changes—Removed Detail

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

The proposed change relocates certain details from the TS to other documents under regulatory control. The TS Bases, [updated final safety analysis report] UFSAR, and [Technical Requirements Manual] TRM will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the TS Bases are subject to the change control provisions in the Administrative Controls Chapter of the TS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e). Other documents are subject to controls imposed by TS or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this 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?*

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements, and adequate control of the information will be maintained. 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 change involve a significant reduction in a margin of safety?*

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. In addition, the details to be moved from the TS to other documents are the same as the existing TS. Since any future changes to these details will be evaluated, no significant reduction in a margin of safety will be allowed. A significant reduction in a margin of safety is not associated with the elimination of the 10 CFR 50.92 requirement for NRC review and approval of future changes to the relocated details. The proposed change is consistent with NUREG 1433, issued by the NRC staff, revising the TS to reflect the approved level of detail, which indicates that there is no significant reduction in a margin of safety.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: October 25, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by an Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 25, 2001.

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 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 radionuclides 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 [a] 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 [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.

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: August 20, 2001.

Description of amendment request: The licensee is proposing to revise Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) to add a footnote to Table 3.3–3 regarding the Steam Line Isolation and Engineered Safety Feature Actuation System (ESFAS) functions. This revision will allow VCSNS to exclude ESFAS steam line isolation instrumentation operability in Mode 3 when the main steam isolation valves, along with associated bypass valves, are closed and disabled, and ease the restriction of Specification 3.0.4 when performing reactor coolant system (RCS) resistance temperature device (RTD) cross calibrations at temperatures below the ESFAS P–12 Interlock for Low-Low T^{avg}. This request is consistent in part with the improved Standard Technical Specifications (ITS).

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?*

[The] proposed changes involve upgrading the VCSNS TS to more closely agree with ITS and does not result in any hardware changes. The proposed change revises the applicability for the initiating functions of the main steam isolation function such that when a main steam line isolation valve is closed and the isolation function is accomplished, the automatic initiation of this function is no longer required to be operable. The ESFAS is not assumed to be an initiator of any analyzed event. The role of the ESFAS is in mitigating and thereby limiting the consequences of accidents. The proposed change continues to adequately ensure the operability of the ESFAS main steam line isolation function when the lines are unisolated and thereby ensures the protection provided by the function remains operable when required. The relaxation of the P–12 Function during RCS RTD cross calibration allows all associated narrow range temperature channels to remain in test, with test circuitry installed, during the transition between Modes 4 and 3. Surveillance performance is administratively controlled by plant procedures which assure testing is conducted below the ESFAS P–12 interlock setpoint of 552 °F and that TS limits for mode operability are not exceeded. Therefore, the results of the analyses described in the FSAR [Final Safety Analysis Report] remain bounding. Additionally, the proposed change does not impose any new safety analyses limits or alter the plant's ability to detect or mitigate events. Therefore, this 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?*

The proposed changes involve upgrading the ESFAS area of the VCSNS TS to more closely agree with ITS and to support RCS RTD cross calibration. The changes do not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters 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 this change involve a significant reduction in margin of safety?*

The proposed change, which upgrades the ESFAS area of the VCSNS TS to be more consistent with ITS and supports RCS RTD cross calibration, does not have an adverse impact on any design basis safety analysis. In combination with administrative controls, required safety functions will continue to be accomplished in accordance with safety analysis assumptions. As such, the results of the analyses described in the FSAR remain bounding [, thus] assuring the proposed change does not involve a significant reduction in 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: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard Laufer, Acting.

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

Date of amendment request: October 31, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TS) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 31, 2001.

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 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 radionuclides 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: November 15, 2001 (TS–01–08).

Description of amendment request: The proposed amendment would increase the full core thermal power rating by 1.3 percent from 3411 MWt to 3455 MWt, based on planned installation of the improved Caldon, Incorporated (Caldon) Leading Edge Flow Meter, LEFM™ (LEFM) feedwater flow measurement instrumentation. This change affects Operating License Condition 2.C.(1) and Definition 1.26 for Rated Thermal Power. In addition, changes are necessary to the reactor power limits of Technical Specification (TS) Table 3.7.1 with an inoperable main steam safety valve for both units and, for Unit 2 only, the interval for which the pressure and temperature curves and the low temperature over pressure protection curves (TS Figures 3.4–2, 3.4–3, and 3.4–4) are valid. A change to the Bases for TS Section 3/4.7.1.1 is also included to address the changes in main steam safety valve capabilities.

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:

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

The comprehensive analytical efforts performed to support the proposed change included a review of the nuclear steam supply systems (NSSSs) and components that could be affected by this change. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanism, loop piping and supports, reactor coolant pump, steam generator and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. The rod control cluster assembly (RCCA) drop time remains within the current limits assumed in the accident analyses. Thus, there is no increase in the consequences of the accidents which credit RCCA drop. Several steam generator tubes may need to be plugged to preclude the potential for U-bend fatigue if the plant operates below certain steam pressure values. As long as these provisions are maintained, there is no increase in the probability of an steam generator tube rupture event. The leak before break analysis conclusions remain valid and thus the limiting break sizes determined in this analysis remain bounding.

All of the NSSS systems will continue to perform their intended design functions during normal and accident conditions. The pressurizer spray flow remains above its design value. Thus, the control system design analyses that credit the spray flow do not need to be modified for changes in this flow. The auxiliary systems and components continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS and/or balance of plant (BOP) interface systems will continue to perform their intended design functions. The steam generator safety valves will provide adequate relief capacity to maintain the steam generators within design limits. The steam dump system will still relieve 40 percent of the maximum full load steam flow. The current loss-of-coolant accident (LOCA) hydraulic forcing functions are still bounding. Thus, there is no significant increase in the probability of an accident previously evaluated.

The fuel has been completely analyzed to determine the effect of the 1.3 percent power uprate. The fuel assembly and fuel rod integrity have been evaluated. The change results in negligible changes to the hydraulic lift forces and the existing holddown margins remain acceptable. The increase in corrosion of the fuel assembly structural Zircaloy-4 components due to a slight increase in temperature is small, thus acceptable structural margin for normal operating, faulted, and handling conditions exist. The fuel assembly and fuel rod flow-induced vibration (FIV) performance remains acceptable. The existing fuel assembly faulted condition loading and analysis remain applicable and acceptable. The fuel rod strain, creep collapse, and corrosion performance were evaluated at the higher power level with acceptable results.

The fuel cycle design was evaluated and there was no significant effect caused by the 1.3 percent power uprate. The operational analysis of the core was evaluated for the change and found to remain applicable with acceptable results.

The thermal-hydraulic analysis was evaluated and found to remain applicable. The safety analysis addressed all Condition II, III, and IV events with the conclusion that current analyses remain applicable or bounding. The radiological consequences were evaluated and determined to be bounded by current analyses.

Additionally, the current licensing basis steamline break and LOCA mass and energy releases that are used to determine the peak containment pressure and temperature limits continue to remain bounding with the increase in power. Thus, there is no significant increase in the consequences of an accident previously evaluated.

The heatup and cooldown curves for Unit 2 are now applicable for 14.5 EFPY [effective full-power year] instead of 16 EFPY. The heatup and cooldown curves define limits that still ensure the prevention of nonductile failure for the SQN Units 1 and 2 reactor coolant system (RCS). The design-basis events that were protected have not changed. This modification does not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequences of an accident described in the Updated Final Safety Analysis Report. Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

The revised requirements for inoperable MSSVs [main steam safety valves] provide limits for the power range high flux trip setpoint that ensure adequate relief capability for postulated accidents. This change does not alter any plant systems, components, or operating methods. Since the plant will continue to operate in the same manner with the same protective features, this change will not increase the possibility of an accident. The revised setpoint is a conservative change that provides additional margin considering the effect of the proposed power uprate. Since the revised setpoint continues to provide an equivalent level of safety function, this change will not significantly increase the consequences of an accident and the offsite dose impact will not be significantly increased.

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

No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Operation at the 3455 MWt core power does not involve a significant reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. The reduction in the EFPY for the Unit 2 heatup and cooldown curves does not reduce the margin of safety since the curves define the limits for ensuring the prevention of nonductile failure for the RCS and these curves remain unchanged.

The pressure and temperature safety limits will be the same as those for the current operating cycle, thus ensuring that the fuel will be maintained within the same range of safety parameters that form the basis for the Final Safety Analysis Report (FSAR) accident evaluations.

The power uprate represents a small increase in the energy production for the fuel cycle and is well within typical variations that occur as a result of increases in cycle length and capacity factor. The burnup of the fuel will increase proportionally with the increase in power, but will not challenge the current licensed burnup limit for Mark-BW fuel.

The slight increase in core average linear heat rate will result in a slight loss of operating margin, but will not affect safety margins. The centerline fuel melt and transient cladding strain limits will not be affected by the power level uprate, but the margin to these limits will decrease slightly. The LOCA FQ [power peaking factor] limits will not be altered since the increase in core power is absorbed by reducing the power uncertainty used in determination of the limits.

The power peaking limits that provide DNB [departure from nucleate boiling] protection are slightly lower resulting in a proportional loss in DNB margins. The mechanical evaluation of the fuel demonstrates that the power level uprate can be successfully accomplished in compliance with all design criteria.

All FSAR Chapter 15 events have been evaluated and found to remain applicable for the power uprate. The radiological consequences analyses include an initial power assumption of 105 percent of 3411 MWt and remain bounding for the 1.3 percent power uprate.

The more restrictive limits for the power range high flux trip setpoint is based on calculations that ensure sufficient relief capacity to meet accident mitigation requirements. This change will appropriately limit reactor power levels, with inoperable MSSVs, such that the margin of safety is maintained at an equivalent level considering the proposed power uprate.

As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the NRC or that are in compliance with all applicable regulatory review guidance and standards. All of the fuel and safety evaluations for the 1.3 percent power uprate were performed with the Framatome-ANP approved methodology listed in TS Section 6.9.1.14 of the SQN TSs. Therefore, it is concluded that the proposed changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request October 31, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TS) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 31, 2001.

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 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 radionuclides 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Richard P. Correia.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: November 13, 2001.

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Unit 1 (WBN) Technical Requirements Manual to add two new sections, TR 3.7.6, ''Shutdown Board Room (SDBR) Air Conditioning System (ACS),'' and TR 3.7.7, ''Elevation 772.0 480 Volt Board Room Air Conditioning (AC) Systems.'' Each section provides specific actions and associated completion times for various out-of-service conditions associated with the safety-related air conditioning systems.

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:

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

The proposed revision to the WBN Technical Requirements Manual (TRM) will provide formalized operational guidance for coping with partial or complete unavailability of shut down board room (SDBR) and 480V board room air conditioning (AC) equipment for limited periods of time. The change does not impact the frequency of an accident because failure of either the SDBR or the 480V board room AC systems is not an initiator of any accident scenario. The change does not modify any plant hardware including the air conditioning systems, and none of their automatic control features or redundant systems currently credited in failure analyses are being deleted, modified, or otherwise replaced by operator actions as a result of the proposed change.

The proposed TRM revision changes current plant operating practice and WBN Final Safety Analysis Assumptions (FSAR) assumptions by allowing continued power operation with both trains of SDBR air conditioning concurrently inoperable and two 480V board room AC systems of the same unit to be concurrently inoperable for a limited duration, up to 12 hours. This condition is acceptable based on the low probability of the occurrence of postulated accidents resulting in core damage concurrent with multiple inoperable systems or trains of cooling equipment during this timeframe, and based on analyses which demonstrate that peak temperatures in each room served by these systems remain below mild environment temperature limits during this time period. Consequently, there is no significant adverse impact on the ability of required safety-related electrical equipment to continue to operate and perform their required functions, during both normal operation and during design basis events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not modify any plant hardware including the subject air conditioning systems. The change provides specific operational guidance for coping with partial or complete unavailability of shut down board room and 480V board room air conditioning equipment. No new accident or event initiators are created by allowing multiple air conditioning systems to be unavailable for the limited time period of 12 hours. The supported electrical equipment remains capable of performing its intended function both during normal operations and post accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TRM revision changes current FSAR assumptions by allowing continued power operation with both trains of SDBR air conditioning concurrently inoperable and allowing two 480V board room air conditioning systems of the same unit to be inoperable for a limited duration, up to 12 hours. This condition does not significantly reduce the margin of safety due to the low probability of the occurrence of a postulated accident resulting in core damage concurrent with multiple inoperable systems or trains of cooling equipment during the limited time period. In addition, transient temperature analyses demonstrate that peak temperatures in each room served by these systems remain below mild environment temperature limits for a period of 24 hours assuming a complete loss of air conditioning to all rooms served by the SDBR and 480V board room AC systems concurrently. The analysis is bounding for normal operational conditions. Consequently, there is no significant adverse impact on the ability of required safety-related electrical equipment to continue to operate and perform their required functions during both normal operation and during design basis events. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: October 2, 2001.

Brief description of amendments: The proposed amendment deletes requirements from the Technical Specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island Nuclear Station] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2 (TMI–2). Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions. The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271), on possible amendments to eliminate PASS, 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 October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 2, 2001.

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 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, the emergency operating procedures, and site survey monitoring that support modification of emergency plan protective action recommendations.

Therefore, the elimination of PASS requirements from the TSs (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 radionuclides 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 25, 2001.

Brief description of amendments: The proposed amendment would revise Technical Specification (TS) 4.2.1, ''Fuel Assemblies,'' for Comanche Peak Steam Electric Station (CPSES) Units 1 and 2 to allow the use of ZIRLO™ test assemblies and to further allow, '' * * * A limited number of lead test assemblies * * * be placed in non-limiting core regions.''

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

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

Response: No.

Changing the technical specifications within limits of the bounding accident analyses cannot change the probability of an accident previously evaluated, nor will it increase radiological consequences predicted by the analyses of record. Controlling the use of lead test assemblies according to limitations approved by the NRC [Nuclear Regulatory Commission] constrains fuel performance within limits bounded by existing design basis accident and transient analyses.

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

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

Response: No.

Inclusion in the reactor core of lead test assemblies according to limitations set by the NRC for lead test assemblies and of a design approved by the NRC ensures that their effect on core performance remains within existing design limits. Use of fuel assemblies whose design has been previously approved by the NRC as lead test assemblies is consistent with current plant design bases, does not adversely affect any fission product barrier, and does not alter the safety function of safety significant systems, structures and components or their roles in accident prevention or mitigation. Currently licensed design basis accident and transient analyses of record remain valid.

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

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

Response: No.

The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Setpoints, or Limiting Conditions for Operation are determined. This proposed clarification of TS 4.2.1 is bounded by existing limits on reactor operation. It leaves current limitations for use of lead test assemblies in place, conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation within analyzed and licensed boundaries.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 8, 2001.

Brief description of amendments: The amendments would add the following to the Technical Specifications (TSs) for Comanche Peak Steam Electric Station (CPSES): (1) the phrase, ''* * * or if open, capable of being closed * * *'' to the TS Limiting Condition for Operation 3.9.4 for the equipment hatch, during core alterations or movement of irradiated fuel assemblies inside containment; and (2) the requirement to verify the capability to install the equipment hatch in a new Surveillance Requirement (SR) 3.9.4.2. Nothing is proposed to be deleted from the TSs. Existing SR 3.9.4.2 would be renumbered SR 3.9.4.3, but would not otherwise be changed. Item (1) will allow the equipment hatch to be open during the conditions stated above.

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

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

Response: No.

The proposed changes will allow the equipment hatch to be open during CORE ALTERATIONS and movement of irradiated fuel assemblies inside containment. The status of the equipment hatch during refueling operations has no affect on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Since the consequences of a fuel handling accident inside containment with an open equipment hatch are bounded by the current analysis described in the FSAR [Final Safety Analysis Report] and the probability of an accident is not affected by the status of the equipment hatch, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No new equipment will be added and no new limiting single failures will be created. The plant will continue to be operated within the envelope of the existing safety analysis.

Therefore, the proposed changes do not create 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 previously determined radiological dose consequences for a fuel handling accident inside containment with the personnel air lock doors open remain bounding for the proposed changes. These previously determined dose consequences were determined to be well within the limits of 10 CFR [Part] 100 and they meet the acceptance criteria of SRP [Standard Review Plan] section 15.7.4 and GDC [General Design Criterion] 19.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of application request: November 7, 2001.

Description of amendment request: A change is proposed to Technical specification 3.0.3 to allow a longer period of time to perform a missed surveillance. The time 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 would be added to the specification: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The 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 November 7, 2001.

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 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.

The NRC staff proposes to determine that the amendment request involves no significant hazards considerations.

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

NRC Section Chief: Stephen Dembek.

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

Date of application request: November 7, 2001 (ULNRC–04557).

Description of amendment request: The proposed amendment would revise Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3 in the Technical Specifications (TSs) on reactor trip system (RTS) instrumentation. The proposed change to SR 3.3.1.2 would replace the reference to the nuclear instrumentation system (NIS) channel output by a reference to the power range channel output, and delete Note 1 to the SR. The change to SR 3.3.1.3 is editorial in nature.

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The RTS instrumentation will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR [Final Safety Analysis Report] are not adversely affected because the change to the NIS power range channel daily surveillance assures the conservative response of the channel even at part-power levels.

The proposed changes modify the NIS power range channel daily surveillance requirement to assure the NIS power range functions are tested in a manner consistent with the safety analysis and licensing basis.

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

The proposed changes will not alter any assumptions or change any mitigation actions in the [accident] radiological consequence evaluations in the FSAR.

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.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating parameters. No performance requirements or response time limits will be affected; however, the proposed TS Bases changes impose explicit NIS power range high trip setpoint adjustment requirements prior to adjusting indicated power in a decreasing power direction. These requirements are consistent with assumptions made in the safety analysis and licensing basis.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

This amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

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

The proposed changes require a revision to the criteria for implementation of NIS power range channel adjustments based on secondary power calorimetric calculations; however, the changes do not eliminate any RTS surveillances or alter the frequency of surveillances required by the Technical Specifications. The revision to the criteria for implementation of the daily surveillance will have a conservative effect on the performance of the NIS power range channels, particularly at part-power conditions. The nominal trip setpoints specified in the Technical Specification Bases and the safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (FΔH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The imposition of appropriate surveillance testing requirements will not reduce any margin of safety since the changes will assure that safety analysis assumptions on equipment operability are verified on a periodic frequency.

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

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

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

NRC Section Chief: Stephen Dembek.

Virginia Electric and Power Company, Docket No. 50–280, Surry Power Station, Unit No. 1, Surry County, Virginia

Date of amendment request: October 15, 2001, as supplemented November 8, 2001.

Description of amendment request: The proposed amendment would revise Technical Specifications Section 4.4. The proposed changes would permit a one-time 5-year extension of the 10-year performance-based Type A test interval established in NEI 94–01, ''Nuclear Energy Institute Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' Revision 0, July 26, 1995.

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

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

The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurrence nor, is an activity or modification by itself that could lead to equipment failure or accident initiation.

The proposed extension to Type A testing does not result in a significant increase in the consequences of an accident as documented in NUREG–1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and C tests. It concludes that reducing the Type A (ILRT) testing frequency to once per twenty years leads to an imperceptible increase in risk.

Surry provides a high degree of assurance through indirect testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last two Type A tests identified containment leakage within acceptance criteria, indicating a very leak-tight containment. Inspections required by the ASME Code are also performed in order to identify indications of containment degradation that could affect leak-tightness. Also, maintaining the containment subatmospheric during operations provides constant monitoring of the leaktightness of the containment structure. Separately, Type B and C testing, required by Technical Specifications, identifies any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the Surry Type A test interval will not represent a significant increase in the consequences of an accident.

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

The proposed revision to Technical Specifications adds a one-time extension to the current interval for Type A testing for Surry Unit 1. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure.

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

The proposed revision to Surry Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test for Surry Unit 1. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal [effect] on this risk since 95% of the Type A detectable leakage paths would already be detected by Type B and C testing. In addition, the risk impact on the total integrated (fifteen year total) Surry Unit 1 plant risk above baseline, for those accident sequences influenced by Type A testing, is only 0.004%. Furthermore, for Surry, maintaining the containment subatmospheric during plant operations further reduces the risk of any containment leakage path going undetected.

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

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: May 31, 2001 as supplemented October 17, 2001.

Description of amendment request: The proposed changes would revise the Technical Specifications and associated Bases to provide a separate allowed outage time for the backup air supply for the pressurizer power-operated relief valves (PORVs).

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:

Dominion has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed change for Surry Units 1 and 2 and determined that a significant hazards consideration is not involved. The following is provided to support this conclusion.

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

The proposed change does not introduce any new mechanisms for the initiation of transients or accidents or for the failure of equipment relied upon in the accident analyses to mitigate the consequences of accidents. The impact of the proposed change on the availability and reliability of the pressurizer PORVs is negligible. Therefore the accident analysis results and conclusions remain bounding.

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

There are no modifications to the plant as a result of the changes. No new accident or event initiators are created by changing the required actions for various conditions of PORV inoperability. The proposed change will not introduce any new equipment failure modes that could initiate accidents or change the analysis results presented in the UFSAR [Updated Final Safety Analysis Report].

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

The proposed change will not alter the limiting results of the safety analyses presented in Chapter 14 of the UFSAR. Provision of an allowed outage time for the pressurizer PORV backup air system and of more condition specific and appropriate actions for various types of PORV inoperability has an insignificant impact on the availability and reliability of the PORVs for performing their safety related functions.

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

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

NRC Section Chief: Richard J. Laufer, Acting.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) The applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: July 26, 2001.

Brief description of amendments: The amendments modify Technical Specifications 5.5.14.b and 5.5.14.b.2, Technical Specification Bases Control Program, such that they are consistent with Title 10 of the Code of Federal Regulations (10 CFR 50.59).

Date of issuance: November 21, 2001.

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

Amendment Nos.: 247 and 222.

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

*Date of initial notice in **Federal Register:*** September 5, 2001 (66 FR 46475) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated November 21, 2001.

No significant hazards consideration comments received: No.

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: April 19, 2001.

Brief description of amendment: The amendment changes the River Bend Station Technical Specifications (TSs) to allow an increase in the number of spent fuel assemblies (SFAs) to be stored in the spent fuel pool from the current TS limit of 2680 SFAs to 3104 SFAs.

Date of issuance: November 19, 2001.

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

Amendment No.: 123.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 18, 2001 (66 FR 52948) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 19, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 23, 2001, as supplemented by letter dated October 25, 2001.

Brief description of amendment: The change deletes Technical Specification

(TS) 3.9.12, ''Fuel Handling Building Ventilation System,'' and TS 3.3.3.1 Surveillance Requirements for the Fuel Storage Pool area radiation monitors.

Date of issuance: November 21, 2001.

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

Amendment No.: 176.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 22, 2001 (66 FR 44169). The October 25, 2001, supplement contained clarifying information that did not change the scope of the July 23, 2001, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 2001.

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: August 22, 2001.

Brief description of amendments: The amendments revise the Technical Specifications for St. Lucie Units 1 and 2 to allow small, controlled, safe insertions of positive reactivity while in shutdown modes.

Date of Issuance: November 19, 2001.

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

Amendment Nos.: 179 and 122.

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

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48287).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 2001.

No significant hazards consideration comments received: No.

GPU Nuclear Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, Dauphin County, Pennsylvania

Date of amendment request: June 21, 2001.

Brief description of amendment request: The amendment revises Three Mile Island Nuclear Station, Unit 2 Technical Specifications Administrative Controls section to provide consistency with the changes to the revised subsection 50.59 of Title 10 of the Code of Federal Regulations, as published in the **Federal Register** on October 4, 1999 (64 FR 53582).

Date of issuance: November 28, 2001

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

Amendment No.: 57.

Facility Operating License No. DPR–73: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 31, 2001 (66 FR 55020).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 28, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 15, 2001.

Brief description of amendments: The amendments change TS 3/4.8.2.2, ''A. C. Distribution Shutdown,'' TS 3/4.8.2.4 ''D. C. Distribution—Shutdown,'' and TS 3/4.9.4, ''Containment Building Penetrations.'' The proposed amendments replaces the current required actions in TSs 3/4.8.2.2. and 3/4.8.2.4, to establish containment integrity within 8 hours if less than the specified minimum complement of A.C. or D.C. busses and equipment is operable in Modes 5 and 6 with new actions which require to immediately suspend operations involving core alterations, positive reactivity changes, and movement of irradiated fuel assemblies, to immediately initiate actions to restore the required busses and return equipment to operable status, and to immediately declare the associated required residual heat removal loop(s) inoperable. The proposed new actions are consistent with NUREG—1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 1.

In addition, the proposed amendments will change TS 3/4.9.4 to add options to use containment penetration closure methods that are equivalent to those that are currently required by the TSs during core alterations or movement of irradiated fuel in containment, and to allow unisolation of some penetrations under administrative control.

Date of issuance: November 21, 2001.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 259 and 242.

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

*Date of initial notice in **Federal Register:*** June 12, 2001 (66 FR 31709).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 21, 2001.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: October 8, 2001.

Brief description of amendments: The amendments revised the Technical Specifications to allow the main control room boundary to be opened intermittently under administrative controls and to allow 24 hours to restore the main control room boundary to Operable status before requiring the plant to perform an orderly shutdown.

Date of issuance: November 26, 2001.

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

Amendment Nos.: 225 and 168.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 26, 2001 (66 FR 54301).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 26, 2001.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: September 7, 2001 (TS 01–09).

Brief description of amendment: The amendment revised Technical Specifications (TS) Section 3.6.11, ''Ice Bed,'' Surveillance Requirement (SR) 3.6.11.2, SR 3.6.11.3, and the associated Bases, to lower the minimum average ice basket weight from 1236 pounds to 1110 pounds.

Date of issuance: November 29, 2001.

Effective date: As of the date of its issuance and shall be implemented no later than Mode 4 during startup from Cycle 4 refueling outage.

Amendment No.: 33.

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

Date of initial notice in **Federal Register:** October 17, 2001 (66 FR 52804).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 29, 2001.

No significant hazards consideration comments received: No.

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

Dated at Rockville, Maryland, this 3rd of December, 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–30455 Filed 12–11–01; 8:45 am]

BILLING CODE 7590–01–P

OVERSEAS PRIVATE INVESTMENT CORPORATION

Submission for OMB Review; Comment Request

AGENCY: Overseas Private Investment Corporation.

ACTION: Request for comments.

SUMMARY: Under the provisions of the Paperwork Reduction Act (44 U.S.C. Chapter 35), agencies are required to publish a Notice in the **Federal Register** notifying the public that the Agency is preparing an information collection request for OMB review and approval and to request public review and comment on the submission. Comments are being solicited on the need for the information, its practical utility, the accuracy of the Agency's burden estimate, and on ways to minimize the reporting burden, including automated collection techniques and uses of other forms of technology. The proposed form under review is summarized below.

DATES: Comments must be received within 60 days of publication of this Notice.

ADDRESSES: Copies of the subject form and the request for review prepared for submission to OMB may be obtained from the Agency Submitting Officer. Comments on the form should be submitted to the Agency Submitting Officer.

FOR FURTHER INFORMATION CONTACT: *OPIC Agency Submitting Officer:* Carol Brock, Records Manager, Overseas Private Investment Corporation, 1100 New York Avenue, NW., Washington, DC 20527; 202/336–8563.

Summary of Form Under Review

Type of Request: Form Amendment.

Title: Application for Political Risk Investment Insurance.

Form Number: OPIC–52.

Frequency of Use: Once per investor, per project.

Type of Respondents: Business or other institutions.

Standard Industrial Classification Codes: All.

Description of Affected Public: U.S. companies investing overseas.

Reporting Hours: 6½ hours per project.

Number of Responses: 150 per year.

Federal Cost: $24,300 per year.

Authority for Information Collection: Sections 231 and 234(a) of the Foreign Assistance Act of 1961, as amended.

Abstract (Needs and Uses): The OPIC 52 form is the principal document used by OPIC to determine the investor's and the project's eligibility, assess the environmental impact and development effects of the project, measure the economic effects for the United States and the host country economy, and collect information for underwriting analysis.

Dated: December 6, 2001.

Rumu Sarkar,

Assistant General Counsel, Administrative Affairs, Department of Legal Affairs.

[FR Doc. 01–30657 Filed 12–11–01; 8:45 am]

BILLING CODE 3210–01–M

RAILROAD RETIREMENT BOARD

2002 Railroad Experience Rating Proclamations, Monthly Compensation Base and Other Determinations

AGENCY: Railroad Retirement Board.

ACTION: Notice.

SUMMARY: Pursuant to section 8(c)(2) and section 12(r)(3) of the Railroad Unemployment Insurance Act (Act) (45 U.S.C. 358(c)(2) and 45 U.S.C. 362(r)(3), respectively), the Board gives notice of the following:

1. The balance to the credit of the Railroad Unemployment Insurance (RUI) Account, as of June 30, 2001, is $53,029,889.30;

2. The September 30, 2001, balance of any new loans to the RUI Account, including accrued interest, is zero;

3. The system compensation base is $3,095,486,497.55 as of June 30, 2001;

4. The cumulative system unallocated charge balance is ($236,829,145.06) as of June 30, 2001;

5. The pooled credit ratio for calendar year 2002 is zero;

6. The pooled charged ratio for calendar year 2002 is zero;

7. The surcharge rate for calendar year 2002 is 2.5 percent;

8. The monthly compensation base under section 1(i) of the Act is $1,100 for months in calendar year 2002;

9. The amount described in section 1(k) of the Act as ''2.5 times the monthly compensation base'' is $2,750 for base year (calendar year) 2002;

10. The amount described in section 2(c) of the Act as ''an amount that bears the same ratio to $775 as the monthly compensation base for that year as computed under section 1(i) of this Act bears to $600'' is $1,421 for months in calendar year 2002;

11. The amount described in section 3 of the Act as ''2.5 times the monthly compensation base'' is $2,750 for base year (calendar year) 2002;

12. The amount described in section 4(a–2)(i)(A) of the Act as ''2.5 times the monthly compensation base'' is $2,750 with respect to disqualifications ending in calendar year 2002;

13. The maximum daily benefit rate under section 2(a)(3) of the Act is $52 with respect to days of unemployment and days of sickness in registration periods beginning after June 30, 2002.

DATES: The balance in notice (1) and the determinations made in notices (3) through (7) are based on data as of June 30, 2001. The balance in notice (2) is based on data as of September 30, 2001. The determinations made in notices (5) through (7) apply to the calculation, under section 8(a)(1)(C) of the Act, of employer contribution rates for 2002. The determinations made in notices (8) through (12) are effective January 1, 2002. The determination made in notice (13) is effective for registration periods beginning after June 30, 2002.

ADDRESSES: Secretary to the Board, Railroad Retirement Board, 844 Rush Street, Chicago, Illinois 60611–2092.

FOR FURTHER INFORMATION CONTACT: Marla L. Huddleston, Bureau of the Actuary, Railroad Retirement Board, 844 Rush Street, Chicago, Illinois 60611–2092, telephone (312) 751–4779.

SUPPLEMENTARY INFORMATION: The RRB is required by section 8(c)(1) of the Railroad Unemployment Insurance Act (Act) (45 U.S.C. 358(c)(1)) as amended by Public Law 100–647, to proclaim by October 15 of each year certain system-wide factors used in calculating experience-based employer contribution rates for the following year. The RRB is further required by section 8(c)(2) of the Act (45 U.S.C. 358(c)(2)) to publish the amounts so determined and proclaimed.