has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Mr. Noel F. Dudley (telephone 301/415–6888) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: October 4, 1996.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 96–25905 Filed 10–8–96; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 16, 1996, through September 27, 1996. The last biweekly notice was published on September 25, 1996 (61 FR 50338).

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 Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By November 8, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

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

Date of amendments request: September 10, 1996

Description of amendments request: The proposed amendment would extend the Engineered Safety Features Actuation System (ESFAS) automatic actuation logic channel functional test surveillance interval from monthly to quarterly. The amendment request is based on analysis documented in Combustion Engineering Owners Group (CEOG) Topical Reports CEN-327 (Reference a), CEN 327, Supplement 1 (Reference b), and CEN-403, Revision 1-A, (Reference c). We have confirmed that the information presented in CEN-327 and CEN-403 is applicable to Calvert Cliffs, and agree with the methodology used to develop the topical reports. In a related matter, the licensee, also requests that the surveillance test interval for the containment sump isolation valves be extended from monthly to quarterly.

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

The Reactor Protective System and the Engineered Safety Features Actuation System (ESFAS) provide the actuation signals to safety equipment necessary to mitigate design basis accidents and transients. The proposed change would increase the surveillance test interval from monthly to quarterly for the ESFAS automatic actuation logic channel functional tests and associated actuation relays. The proposed change will also extend the containment sump isolation valve automatic opening verification surveillance interval from monthly to quarterly. The ESFAS instruments and containment sump isolation valves are not initiators in any previously evaluated accidents. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

The ESFAS automatic actuation logic circuitry and actuation relays are essentially digital devices, which are not subject to time-related instrument drift. Therefore, a plant-specific instrument drift analysis for these components is not required. However, in support of Calvert Cliffs License Amendment Request, dated May 27, 1994, a plant-specific setpoint drift analysis for each sensor loop demonstrated that the observed changes in instrument uncertainties for the extended surveillance test interval did not exceed the 30-day setpoint assumptions. This provides confidence that the 90-day test interval will not impact the ability to detect and monitor system degradation. A review of previous containment sump isolation valve surveillance test procedures revealed no valve or valve operator failures. Additionally, single failure criteria continues to be satisfied by two redundant and independent valves on each unit. Therefore, the proposed changes will not change the ability of the ESFAS instrumentation or associated engineered safety features equipment to respond to and mitigate the consequences of any previously evaluated accident.

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

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

The proposed extended surveillance test interval for the ESFAS instruments, actuation relays, and containment sump isolation valve automatic opening verification does not involve any changes in equipment or the function of these instruments. The proposed change does not represent a change in the configuration or operation of the plant. The ESFAS setpoints will not be affected since the ESFAS automatic actuation logic circuitry and actuation relays are not subject to time-related instrument drift. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The proposed change will not affect the functions of the ESFAS instruments or associated equipment. Topical Reports CEN-327, ''RPS/ESFAS Extended Test Interval Evaluation,'' and CEN-327, Supplement 1, quantified the corresponding changes in core melt frequency for the representative fault tree models that were developed for Calvert Cliffs. Additionally, the ESFAS actuation relay failure data presented in CEN-403, Revision 1-A, ''ESFAS Subgroup Relay Test Interval Extension,'' justifies extending the test interval for these relays. The proposed change has two principal effects with opposing impacts on core melt frequency. The first impact is a slight increase in core melt frequency that results from the increased possibility of an undetected instrumentation failure due to the extended surveillance interval. This assumed unavailability results from less frequent testing. The undetected ESFAS failure represents the potential for the failure of the appropriate engineered safety features to actuate when required. The opposing impact on core melt risk is the corresponding reduction in core melt frequency that would result due to the reduced exposure of the plant to test-induced transients. Topical Report CEN-327 determined that the two changes are nearly equal, and the net result is no distinguishable effect on plant safety. The NRC issued a Safety Evaluation Report which found that the above evaluations were acceptable for justifying the extensions in the surveillance test intervals for the ESFAS automatic actuation logic channel functional tests from 30 days to 90 days. In addition to the evaluation of risk given in Topical Report CEN-327, we have evaluated the plant specific risk associated with these proposed changes and concluded that changing the surveillance intervals from monthly to quarterly results in a net decrease in the annual core melt frequency.

The ESFAS setpoints will not be changed since ESFAS automatic actuation logic circuitry and actuation relays are not subject to time-related instrument drift. The conclusions of the accident analyses in the Calvert Cliffs Updated Final Safety Analysis Report remain valid and the safety limits continue to be met.

Extending the containment sump isolation valve automatic opening surveillance interval from monthly to quarterly will not significantly reduce the margin of safety. Both Units 1 and 2 are provided with two containment sump isolation valves, which satisfy single failure criteria. Historical review of surveillance test procedures and Nuclear Plant Reliability Data System data revealed no failures of these valves or associated valve operators. We have also evaluated the plant specific risk associated with this proposed change to the surveillance interval and conclude that the risk is acceptable.

Based on the generic and plant specific risk evaluations and the demonstrated low failure rate of the components, we conclude that these 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

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

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

NRC Project Director: Alexander W. Dromerick, Acting Director

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of amendment request: August 29, 1996, as supplemented on September 20, 1996

Description of amendment request: The proposed amendments would change the Technical Specifications to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program,'' with certain exceptions detailed in the licensee's application.

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 amendment cannot affect the probability of an accident since it involves only changes in the containment leakage rate testing program. There is no credible accident which can be initiated by containment leakage rate testing.

The proposed amendment will not affect the consequences of a[n] accident since the allowable containment leakage rates, which determine the offsite consequences of a[n] accident, are unchanged. Only the frequency of measuring the leakage rates may be changed.

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

The proposed amendment will not create the possibility of a new or different kind of accident since there are no changes to any systems, structures, or components, and no changes in the method of operation of any system, structure, or component.

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

The proposed amendment will not involve a significant reduction in the margin of safety. As documented in the 10 CFR 50, Appendix J, Option B Proposed Rule and Final Rule published in the Federal Register, the additional industry wide risk resulting from the proposed change is marginal and within acceptable limits.

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

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: August 23, 1996

Description of amendment request: The proposed amendment would change the technical specifications to allow fuel enrichments of up to 5.0 weight percent uranium-235.

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

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

The calculated k-effective including uncertainties, demonstrates substantial margin to criticality in the fuel assembly storage locations for both normal and accident conditions; therefore, the probability of a previously evaluated accident is not significantly increased. Since a criticality accident is demonstrated to not be feasible under the specified conditions, the consequences of a previously evaluated accident are not significantly increased. Administrative controls are utilized in order to assure that a fuel assembly is not placed in an unanalyzed configuration. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The increase in fuel enrichment could be considered a change in plant equipment; however, it would only affect reactivity. The reactivity increase has been analyzed and shown that no new or different kinds of accidents from any previously evaluated exist. The proposed change does not involve the addition of any plant equipment, nor does it modify the method of operation of any plant equipment. Also, the proposed change would not alter the design or configuration of the plant beyond the standard functional capabilities of the equipment. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change has been analyzed to maintain a k-effective of less than the criticality acceptance criteria of 0.95 including uncertainties at the 95/95 probability/confidence level for all storage configurations. Additionally, the optimum moderation condition for the new fuel storage racks has been analyzed and determined to meet the acceptance criteria of maintaining k-effective of 0.98. The use of physical restraints for blocking the storage locations where fuel is prohibited to be stored in the spent fuel pools prevents misloading of fuel into these locations. A dropped assembly and/or the misplacement of a fuel assembly for each storage configuration has been analyzed. By crediting 1000 ppm boron (ANO-2 Technical Specifications require 1600 ppm), the 95/95 k-effective is well below 0.95. 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, Entergy Operations proposes that the requested change 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.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

NRC Project Director: William D. Beckner

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

Date of amendment request: August 23, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 5.0, Design Features, and would for the most part, adopt NUREG-1432, Revision 1, improved ''Standard Technical Specifications for Combustion Engineering Plants'' (ISTS), for this section of the 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 Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed amendment revises the Section 5.0, Design Features, and would for the most part, adopt NUREG-1432 Revision 1, ''Standard Technical Specifications Combustion Engineering Plants,'' for this section of the technical specifications. This proposed change will also allow the relocation of portions of the design features section of the technical specifications to other licensee controlled documents that are controlled under 10 CFR 50.59. This approach is consistent with the NRC final policy statement and the staff's Technical Specification line item improvement program. The relocation of information to licensee controlled documents will improve the usability and readability of technical specifications without changing any of the design requirements for the facility.

This amendment request does not remove or modify any of the design requirements for the facility or affect any accident initiators, conditions or assumptions for any accident previously evaluated. Therefore, this change does not involve a significant increase in the probability of any accident previously evaluated.

This amendment request is administrative in nature and does not affect any system or component functional requirements. This change does not affect the operation of the plant or affect any component that is used to mitigate the consequences of any accident. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

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

2. Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The relocation of existing requirements from the technical specifications to other licensee controlled documents and the reformatting of the design features section of the technical specifications to the NUREG-1432 format are changes that are administrative in nature. This change does not modify or remove any plant design requirement. The proposed change will not affect any plant system or structure, nor will it affect any system functional or operability requirements. Consequently, no new failure modes are introduced as a result of this change. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment request represents a relocation of a portion of the information previously located in the technical specification design features section to other licensee controlled documents that are controlled under 10 CFR 50.59. The proposed change is administrative in nature because the design requirements for the facility remain the same. The proposed change does not represent a change in the configuration or operation of the plant. Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change 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.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

NRC Project Director: William D. Beckner

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

Date of amendment request: July 16, 1996

Description of amendment request: The amendment would change the Technical Specifications to permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing in accordance with the implementation guidance in NRC's Regulatory Guide (RG) 1.163 dated September 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:

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

NMP1 [Nine Mile Point, Unit 1] is currently implementing Option A of Appendix J of 10 CFR 50 for Type, A, B, and C testing. The proposed change to the Technical Specifications and the Bases would implement Option B to Appendix J of 10 CFR 50 at NMP1 for Type A, B, and C testing. Option B would allow increased testing intervals after satisfying certain performance based criteria.

Appendix J describes the requirements for leakage of the primary containment and its components penetrating the primary containment. The leakage testing interval of the primary containment and its components is not a precursor or initiator to an accident. The primary containment and its penetrations minimizes the leakage of radioactivity into the environment during an accident which pressurizes the primary containment.

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

The proposed change to the Technical Specifications and the Bases would replace the detailed and prescriptive technical requirements contained in Option A of Appendix J with performance based requirements in accordance with supporting regulatory/industry documents referenced in Option B of Appendix J. This proposed change includes a description of the 10 CFR 50 Appendix J Testing Program Plan in Section 6.16 of the Technical Specifications.

This program plan, with two exceptions, is consistent with RG 1.163. Therefore, this program plan establishes leakage-rate test methods, procedures, acceptance criteria and analyses which comply with Option B of Appendix J to 10 CFR 50.

The implementation of this program continues to provide adequate assurance that during a DBA-LOCA [design-basis accident/loss-of-coolant accident], the primary containment and its components will continue to limit leakage rates to less than the allowable leakage rates described in the Technical Specifications and thereby limit leakage consistent with the assumptions of the accident analyses. Therefore, the increased test intervals permitted by Option B for the primary containment and its penetrations will continue to implement the safety objectives underlying the requirements of Appendix J.

As discussed below relative to the margin of safety, the impact of the proposed change on the consequences of a release is negligible. The slight increase in the risk to the population is compensated by the corresponding risk reduction benefits associated with the reduction in component cycling, stress, and wear associated with increased test intervals.

Accordingly, operation with the proposed change to the Technical Specifications and the Bases will not significantly increase the consequences of an accident previously evaluated.

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

The proposed change would implement Option B of Appendix J to 10 CFR 50 for Type A, B, and C testing. Option B would allow increased testing intervals after satisfying certain performance based criteria.

No new plant operating modes, system operating configurations nor failure modes are introduced by the proposed change. The primary containment and its penetrations will continue to perform their accident mitigating function.

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

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

A regulatory impact analysis of implementing performance-based requirements indicates that relaxing the frequency of Type A, B, and C testing leads to an increase in overall risk of approximately two percent. As indicated in the Staff's Regulatory Impact Analysis, this increase is considered to be marginal to safety.

As indicated above, increasing test intervals can slightly increase the risk to the population associated with the consequences of a release; however, this is compensated by the corresponding risk reduction benefits associated with the reduction in component cycling, stress, and wear associated with increased test intervals. Therefore, when considering the total integrated risk, the risk associated with increased test intervals is negligible.

The proposed change is consistent with current plant safety analyses. In addition, the proposed change does not require revisions to the design of NMP1. As such, the proposed TS changes will maintain the same level of reliability of the equipment associated with containment integrity, assumed to operate in the plant safety analysis, or provide continued assurance that specified parameters affecting leak rate integrity, will remain within their acceptance limits.

The as-left leakage after performing a required leakage test continues to be less than 0.60 La for combined Type B and C leakage and less than or equal to 0.75 La for Type A leakage. These as-left acceptance criteria and the testing frequency as established by the 10 CFR 50 Appendix J Testing Program Plan provide assurance that the measured leakage rate will not exceed the maximum allowable leakage of La during plant operation.

Visual examination of accessible interior and exterior surfaces of the primary containment continues to be performed prior to initiating a Type A test. The total number of visual examinations performed will continue to be three times during a 10-year period. Therefore, visual examinations of the primary containment will continue to allow for the timely uncovering of evidence of structural deterioration and satisfy the requirements of RG 1.163.

The leakage limits of LCO 3.3.3 will continue to be met prior to reactor coolant system temperature exceeding 215*F and anytime Primary Containment is required. Satisfying these leakage limits provides assurance that the measured leakage rate will not exceed the maximum allowable leakage rate of La during plant operation. Therefore, operation with the proposed change will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Guy S. Vissing, Acting Director

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment requests: May 31, 1996

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant (DCPP), Unit Nos. 1 and 2 to revise 23 TS surveillance frequencies from at least once every 18 months to at least once per refueling interval (nominally 24 months) and to make administrative changes for 6 other TS to maintain consistency for TS that are not proposed for surveillance extension. The specific TS changes proposed include those for 2 response time tests, 3 containment spray system tests, and 24 ventilation system tests.

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 six administrative changes regarding laboratory carbon testing are administrative changes only and do not affect the probability or consequences of accidents.

The 23 proposed TS surveillance interval increases from 18 to 24 months do not alter the intent or method by which the inspections, tests, or verifications are conducted, do not alter the way any structure, system, or component functions, and do not change the manner in which the plant is operated. The surveillance, maintenance, and operating histories indicate that the equipment will continue to perform satisfactorily with longer surveillance intervals. No recurring surveillance or maintenance problems were identified for response time, containment spray system, or control room ventilation system testing.

Recurring maintenance issues on the fuel handling building and auxiliary building ventilation systems regarding the system control panels and certain dampers have been addressed. These ventilation systems are in service during all modes of operation and experience normal wear. None of the problems are related to refueling frequency testing. The monthly surveillance tests provide assurance of system operability for the control panels. The preventative maintenance program for the dampers is independent of refueling shutdowns and provides assurance that degradation mechanisms such as corrosion and wear are adequately addressed.

There are no known mechanisms that would significantly degrade the performance of the evaluated equipment during normal plant operation. All potential time-related degradation mechanisms have insignificant effects in the timeframe of interest (maximum of 30 months). Based on the past performance of the equipment, the probability or consequences of accidents would not be significantly affected by the proposed surveillance interval increases.

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

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

The six administrative changes regarding laboratory carbon testing are administrative changes only and do not affect the type of accidents possible.

The containment spray system and control room, auxiliary building, and fuel handling building ventilation systems are not associated with the initiation of any accident. The reactor trip and engineered safety feature actuation system response times are assumed in the accident analysis. However, the proposed surveillance interval increases would not affect the type of accidents possible.

For the 23 proposed TS changes involving surveillance interval increases from 18- to 24-months, the surveillance and maintenance histories indicate that the equipment will continue to effectively perform their respective design functions over the longer operating cycles. Additionally, the increased surveillance intervals do not result in any physical modifications, affect safety function performance or the manner in which the plant is operated, or alter the intent or method by which surveillance tests are performed. Only a few problems have been identified and generally have not recurred. All potential time-related degradations have insignificant effects in the timeframe of interest. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The six administrative changes regarding laboratory carbon testing are administrative changes only and do not affect the margin of safety.

For the 23 proposed TS changes involving 18- to 24-month surveillance interval increases, evaluation of historical surveillance and maintenance data indicates there have been only a few problems experienced with the evaluated equipment. There are no indications that potential problems would be cycle-length dependent or that potential degradation would be significant for the timeframe of interest; therefore, increasing the surveillance interval will have little, if any, impact on any margin of safety. There is no safety analysis impact since these changes will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria. Safety margins would not be significantly affected by the proposed surveillance interval increases.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120

NRC Project Director: William H. Bateman

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

Date of amendment request: September 13, 1996

Description of amendment request: The proposed amendment changes the Administrative Controls Section 5.6.6 of the Ginna Station Technical Specifications which would allow referencing of Revision of the Ginna Station Pressure and Temperature Limits Report (PTLR) for the Reactor Coolant System (RCS) pressure and temperature (P/T) limits and low temperature overpressure protection (LTOP) limits.

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

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes only revise the reference to the PTLR in the Administrative Controls section of technical specifications and correct a typographical error. The changes complete implementation of Generic Letter 96-03 by referencing NRC approved methodology within the Administrative Controls. As such, these changes are administrative in nature and do not impact initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because the changes have been shown to ensure that the P/T and LTOP limits in the revised PTLR continue to meet all necessary requirements for reactor vessel integrity. These changes are administrative in nature. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610

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

NRC Project Director: Alexander W. Dromerick, Acting Director

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

Date of amendment request: August 21, 1996 (TS 96-03)

Description of amendment request: The proposed change would result in an amendment to Licenses DPR-77 and DPR-79 to change the Technical Specifications (TS) for the Sequoyah Nuclear Plant, Units 1 and 2. The proposed change would revise TS 3.7.1.3, ''Condensate Storage Tank,'' to include a new mode of applicability that reads: ''Mode 4 when steam generator is relied upon for heat removal.'' In addition, other proposed changes to TS 3.7.1.3 and a Bases change to Bases Sections 3/4.3.1 and 3/4.3.2, ''Protective and Engineered Safety Features (ESF) Instrumentation,'' are included to provide improvements and establish requirements that are consistent with Westinghouse Standard Technical

Specifications (NUREG-1431, Revision 1).

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:

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed TS change revises SQN's condensate storage tank (CST) Specification 3.7.1.3 to incorporate requirements from the Westinghouse Standard Technical Specification (STS) contained in NUREG-1431, Revision 1. The proposed change is consistent with the STS for ensuring that SQN's CST remains operable in Modes 1, 2, 3, and Mode 4 when steam generator (SG) is relied upon for heat removal. In addition, the proposed change provides a general TS improvement by incorporating STS phraseology within the action requirements. Included with this change is an increase in the completion time for achieving hot shutdown. The current completion time of 6 hours is increased to 12 hours. This change allows sufficient time, while in Mode 4, to transition from SGs to residual heat removal entry conditions. The 12-hour completion time is reasonable based on operating experience to reach the required plant condition in an orderly manner without challenging plant systems.

[The Tennessee Valley Authority's] TVA's proposed change also includes deletion of Surveillance Requirement (SR) 4.7.1.3.2. This SR demonstrates operability of SQN's essential raw cooling water (ERCW) system every 12 hours whenever the ERCW system is used as a supply source for the auxiliary feedwater (AFW) system. Deletion of this SR is consistent with STS requirements and is justified based on: (1) current SQN TS 3.7.4 requirements ensure operability of SQN's ERCW in Modes 1, 2, 3, and 4, and (2) newly proposed Action (b) requirements ensure that SQN's ERCW system is verified operable every 12 hours whenever the CST is inoperable.

The proposed changes provide TS requirements for SQN's CST that are conservative with respect to assumptions used in SQN's accident analysis as contained in the Final Safety Analysis Report (FSAR). This change does not involve a physical modification to the plant or affect any instrumentation setpoints. Accordingly, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes provide TS requirements for SQN's CST that are conservative with respect to assumptions used in SQN's accident analysis as contained in the FSAR. No new event initiator has been created, nor has any hardware been changed. This change does not involve a physical change to SQN's CST or any other system. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously analyzed.

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

TVA's proposed change replaces SQN's CST TS requirements with TS requirements from the Westinghouse STS (NUREG-1431, Revision 1). The proposed change to SQN's CST TS to add ''Mode 4 when steam generator is relied upon for heat removal,'' provides consistency with the mode requirements of SQN's AFW TS and resolves a disparity that currently exists between these TSs. The allowed outage time for an inoperable CST remains unchanged and is consistent with the allowed outage time in STS. The proposed change to delete a SR for verifying operability of the ERCW system is considered acceptable based on other existing TSs that verify operability of SQN's ERCW system. Overall, similarity exists between SQN's current CST specification and the STS version. Consequently, with the exception of format, the TS requirements remain essentially unchanged.

The proposed changes provide a line-item improvement for SQN's CST TS that are conservative with respect to the assumptions used in SQN's accident analysis as contained in the FSAR. This changes does not involve a setpoint change or physical modification to the plant. Accordingly, the margin of safety has not been reduced.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

Date of amendment request: August 21, 1996 (TS 96-06)

Description of amendment request: The proposed change would result in an amendment to Licenses DPR-77 and DPR-79 to change the Technical Specifications for the Sequoyah Nuclear Plant (SQN), Units 1 and 2. The proposed change would remove Surveillance Requirement 4.8.1.1.1.b that verifies the ability to transfer the unit power supply from the unit generator supported circuit to the preferred power circuit. The current SQN design and operating configurations do not require the use of this transfer feature making this surveillance unnecessary.

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:

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed change will delete a surveillance associated with a feature that does not provide a safety function based on the current SQN design and operating procedures. The transfer feature that is currently verified will either be achieved by the system alignment or covered by the applicable TS action requirements. This transfer feature provided automatic system alignment to preferred offsite power circuits for accident mitigation purposes. This feature, or the lack of, is not considered a source of any accident and the proposed change to delete the associated surveillance will not increase the possibility of an accident previously evaluated. Safety functions are maintained by the current offsite circuit alignment without the transfer feature or associated surveillance. Therefore, the consequences of an accident can not be increased and may be reduced by eliminating the use of active devices to satisfy safety functions.

2. Create the possibility of a new or different kind of accident from any previously analyzed. The offsite circuit transfer feature is not considered to be a source of an accident and the deletion of a surveillance to verify the operability of this transfer will not impact this potential. Therefore, the deletion of Surveillance 4.8.1.1.1.b will not create the possibility of a new or different kind of accident previously analyzed.

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

The current SQN alignment satisfies all required offsite circuit alignments necessary to support accident mitigation functions without the use of active devices. In addition, any time delays associated with the transfer actuation to realign the offsite circuits, that is tested by the surveillance proposed to be deleted, are eliminated by the current alignment. Therefore, the margin of safety associated with the affected safety function is not reduced and may be increased by the elimination of active devices and their associated time delays.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: August 22, 1996 (TS 96-08)

Description of amendment request: The proposed change would result in an amendment to Licenses DPR-77 and DPR-79 to change the Technical Specifications (TSs) for the Sequoyah Nuclear Plant, Units 1 and 2. The proposed changes would delete TS Table 4.8-1, ''Diesel Generator Reliability,'' and revise TS Section 3.8.1 to allow a once per 18 month, 7-day allowed outage time (AOT) for the emergency diesel generators (EDGs). The first change would remove the accelerated testing requirements for the EDGs in accordance with Generic Letter 96-01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators from Technical Specifications.'' The second change would revise the Units 1 and 2 TS to allow a once per 18 month 7-day AOT for planned maintenance activities, particularly an upcoming major 12-year overhaul of all four EDGs.

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:

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

Deletion of Table 4.8-1, in accordance with Generic Letter (GL) 94-01, is an administrative change that will not impact the plant design or operation. None of the assumptions used in evaluating the radiological consequences of an accident are changed. A new or altered release path is not created. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

The emergency diesel generators (EDGs) supply backup power to the essential safety systems in the event of a loss-of-offsite (normal) power. The EDGs cannot initiate an accident. The requested change will not impact the plant design or operation. The increased out of service time does not invalidate assumptions used in evaluating the radiological consequences of an accident and does not provide a new or altered release path. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

An increase in the allowed outage time (AOT) would not change the conditions, operating configuration, or minimum amount of operable equipment assumed in the plant Final Safety Analysis Report for accident mitigation. The longer AOT would provide a longer time window for maintenance, but would lessen the total EDG unavailability per year. Based on the small increase in plant risk during maintenance, and the decrease in overall plant risk as a result of this change, this change will not result in a significant increase in the consequences of an accident.

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

Deletion of Table 4.8-1, in accordance with GL 94-01, is an administrative change that will not impact plant the plant design or operation. Appropriate testing, in accordance with the Maintenance Rule, will continue. Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change to extend the AOT for the EDGs does not alter the physical design, or configuration of the plant. The EDG operation remains unchanged, therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

Deletion of Table 4.8-1, in accordance with GL 94-01, ensures that the requirements and provisions of 10 CFR 50.65 and the guidance of Regulatory Guide 1.160 are met. The program put in place by these documents will ensure that any degradation of the EDGs is identified and appropriate action is taken. Therefore, this change does not involve a significant reduction in the margin of safety.

A change to the maintenance schedule was performed to conform with vendor recommendations. This change in schedule required an increase in the duration of the 18 month and longer maintenance activities. Due to the number of shared systems, three of the four EDGs are required to meet all of the safety functions for each unit. However, the TSs conservatively assume four EDGs are necessary for unit operation; therefore, loss of any one EDG causes entry into a LCO action statement on both units. Performing the required maintenance with a 72-hour AOT will result in more EDG unavailability per year than would be required if the AOT was 7 days. Therefore, the 7-day AOT would not result in a significant reduction in the margin of safety.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: August 28, 1996 (TS 96-07)

Description of amendment request: The proposed change would result in an amendment to Licenses DPR-77 and DPR-79 to change the Technical Specifications (TS) for the Sequoyah Nuclear Plant, Units 1 and 2. The proposed change would revise the setpoint tolerance for the pressurizer safety valves (PSVs) and main steam safety valves (MSSVs) from plus or minus one percent to plus or minus three percent. An analysis performed by Framatome Technologies Incorporated to support this change is provided in the licensee's submittal. These parameters are contained in TS 3.4.2, TS 3.4.3.1, and Table 3.7-2. Additionally, the sentence ''Following testing, lift settings shall be within plus or minus 1%.'' would be added to Surveillance Requirements (SR) 4.4.2, 4.4.3.1, and 4.7.1.1.

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:

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The evaluation contained in Enclosure 5 [Framatome Report No. 77-1257369-01, ''Safety Evaluation of Safety Valve Setpoint Tolerance Relaxation,'' dated August 1, 1996] discusses the consequences of this change as it pertains to the accidents previously analyzed. The positive increase of the setpoint tolerance, from one percent to three percent, of the PSV[s] and the MSSV[s] does not challenge the design limits of the installed systems. This conclusion is demonstrated by means of the reanalysis of the bounding overpressure events. The negative tolerance for the MSSVs, from minus one percent to minus three percent, will result in an increase in mass discharged through these valves. The increase was evaluated and the analysis indicated that the dose release remained within the limits required by 10 CFR 100. Based on the results of this review, there is no increase in the probability of a previously evaluated accident or a significant increase in the consequences of an accident previously evaluated.

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

The proposed TS change will increase the setpoint tolerance for the PSVs and the MSSVs. This change does not involve an equipment addition or change in the method the plant is operated. Therefore, the possibility of a new or different kind of accident from any previously analyzed is not created.

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

The proposed change is a change in the lift setpoint tolerance of the existing valves. The analysis demonstrates that the design limits are not exceeded nor are the dose release limits exceeded due to this increase in setpoint tolerance. Additionally, the valves will be returned to the original tolerance of plus or minus one percent. This will ensure that the maximum margin is retained; therefore, the margin of safety has not been reduced by this change.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: August 7, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 1.0, ''Definitions,'' by defining a refueling interval to be [less than or equal to] 730 days; and would revise TS 3/4.0, ''Applicability,'' TS 3/4.6.2.1, ''Containment Systems - Depressurization and Cooling Systems - Containment Spray System,'' and TS 3/4.6.3.1, ''Containment Systems - Containment Isolation Valves,'' to reflect performing surveillance tests during a refueling interval rather than every 18 months.

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

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station (DBNPS), Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervals from 18 to 24 months for the containment spray system (Surveillance Requirements 4.6.2.1.b), or the containment isolation valves (Surveillance Requirements 4.6.3.1.2). The proposed change to TS 1.0, adding a definition for ''Refueling Interval,'' and the associated proposed change to TS Bases 3/4.0, are administrative changes associated with the 24 month cycle conversion. Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

These revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of an affected system or component was identified during these reviews.

These proposed revisions are consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' dated April 2, 1991.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy is not being changed by these proposed changes. Existing system and component operation is not being changed by these proposed changes. The assumptions used in evaluating the radiological consequences in the DBNPS Updated Safety Analysis Report are not invalidated.

A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of an affected system or component was identified during these reviews.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated. A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of a system or component was identified during these reviews. No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval.

3. Not involve a significant reduction in a margin of safety because a review of the historical 18 month surveillance data and maintenance records identified no potential for a significant increase in a failure rate of a system or component due to increasing the surveillance test interval to 24 months. Existing system and component redundancy is not being changed by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. Therefore, there are no significant reductions in a margin of safety.

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

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: September 4, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 6.2.3, ''Facility Staff Overtime,'' by removing specific overtime limits and working hours and by adding procedural controls to perform a monthly review of overtime hours.

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

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, in accordance with these changes would:1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed.

The proposed change to TS 6.2.3, ''Facility Staff Overtime,'' to relocate specific overtime limits and working hours to the DBNPS Updated Safety Analysis Report (USAR) is consistent with the NRC Staff's determination previously provided on a generic basis in the Safety Evaluation to License Amendment Number 127 and 116 to the Operating Licenses (Number NPF-10 and NPF-15), for the San Onofre Generating Nuclear Station, Units 2 and 3, dated February 9, 1996. The appropriate relocation of TS requirements, such as portions of TS 6.2.3, to licensee-controlled documents is also addressed generically in the NRC's ''Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors'', dated July 23, 1993.

The relocated overtime limits and working hours will be subject to review and evaluation under Section 50.59, ''Changes, Tests, and Experiments'', of Title 10 of the Code of Federal Regulations (10 CFR) prior to any changes being made. The other changes to TS 6.2.3 are editorial, with an exception being that a new requirement has been added for plant procedures to ensure that an individual's overtime is reviewed monthly by the Plant Manager or his designee(s) to ensure excessive hours have not been assigned.

Overtime will remain controlled by plant administrative procedures and USAR requirements generally following the guidance of the NRC's Policy Statement on working hours contained within Generic Letter 82-12, ''Nuclear Power Plant Staff Working Hours.''

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

The proposed changes to TS 6.2.3 only alter the administrative location of and the regulatory controls applicable to plant staff specific overtime limits and working hours. Therefore, there is no significant increase in the consequences of an accident previously evaluated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated, and no new or different failure modes have been defined for any plant system or component important to safety, nor has any limiting single failure been identified as a result of the proposed changes. No new or different types of failures or accident initiators are introduced by the proposed changes.

The proposed changes to TS 6.2.3 only alter the administrative location of and the regulatory controls applicable to plant staff specific overtime limits and working hours. Therefore, there is no possibility created for a new or different kind of accident.

3. Not involve a significant reduction in a margin of safety because facility staff overtime is not an input in the calculation of any safety margin with regard to TS Safety Limits, Limiting Safety System Settings, other TS Limiting Conditions for Operation or other previously defined margins for any structure, system, or component important to safety.

The proposed changes to TS 6.2.3 only alter the administrative location of and the regulatory controls applicable to plant specific overtime limits and working hours. Therefore, there is no significant reduction in a margin of safety.

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

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: September 12, 1996

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 3/4.1.3.4, ''Reactivity Control Systems - Rod Drop Time,'' and TS 3/4.5.2, ''Emergency Core Cooling Systems - Tavg [greater than or equal to] 280°F,'' to change surveillance test intervals from every 18 months to each refueling interval (nominally 24 months). Additionally, the proposed amendment would remove a footnote for TS 4.5.2.b that is no longer applicable.

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

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station (DBNPS), Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervalsfrom 18 months to 24 months for the reactivity control systems (Surveillance Requirement 4.1.3.4.c), or the emergency core cooling systems (Surveillance Requirement 4.5.2.b), or the proposed administrative change to Surveillance Requirement 4.5.2.b to remove a time-conditional footnote which has expired. Initiating conditions and assumptions remain as previously analyzed for all accidents in the DBNPS Updated Safety Analysis Report.

These revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 months to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of an affected system or component was identified during these reviews.

These proposed revisions are consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle,'' dated April 2, 1991.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy is not being changed by these proposed changes. Existing system and component operation is not being changed by these proposed changes. The assumptions used in evaluating the radiological consequences in the DBNPS Updated Safety Analysis Report are not invalidated.

A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of an affected system or component was identified during these reviews.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated. A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 months to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of a system or component was identified during these reviews. No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval.

3. Not involve a significant reduction in a margin of safety because a review of the historical 18 month surveillance data and maintenance records identified no potential for a significant increase in a failure rate of a system or component due to increasing the surveillance test interval to 24 months. Existing system and component redundancy and operation is not being changed by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. Therefore, there are no significant reductions in a margin of safety.

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

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: September 17, 1996

Description of amendment request: The proposed amendment would revise the surveillance interval from 18 months to less than or equal to 730 days, nominally 24 months, for Technical Specification (TS) 3/4.5.2, ''Emergency Core Cooling Systems - ECCS Subsystems - Tavg greater than or equal to 280 degrees F;'' TS 3/4.6.5.1, ''Containment Systems - Shield Building - Emergency Ventilation System;'' TS 3/4.7.6.1, ''Plant Systems - Control Room Emergency Ventilation System;'' TS 3/4.7.7, ''Plant Systems - Snubbers;'' TS 3/4.9.12, ''Refueling Operations - Storage Pool Ventilation;'' and TS Bases 3/4.7.7 - ''Snubbers.''

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

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station (DBNPS), Unit No. 1 in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervals from 18 months to 24 months for the trisodium phosphate dodecahydrate (TSP) volume, (Surveillance Requirement 4.5.2.d.4), Shield Building and Storage Pool Emergency Ventilation Systems, (Surveillance Requirements 4.6.5.1.b, 4.6.5.1.d, 4.9.12.1, and 4.9.12.2), and the Control Room Emergency Ventilation System, (Surveillance Requirements 4.7.6.1.c and 4.7.6.1.e) and Snubbers (Surveillance Requirement 4.7.7.2.b and associated Bases 3/4.7.7). Initiating conditions and assumptions remain as previously analyzed for all accidents in the DBNPS Updated Safety Analysis Report.

These revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of a system or component was identified during these reviews.

These proposed revisions are consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' dated April 2, 1991.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy is not being changed by these proposed changes. Existing system and component operation is not being changed by these proposed changes. The assumptions used in evaluating the radiological consequences in the DBNPS Updated Safety Analysis Report are not invalidated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated. A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of a system or component was identified during these reviews. No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval.

3. Not involve a significant reduction in a margin of safety because the review results of the historical 18 month surveillance data and maintenance records identified no potential for a significant increase in a failure rate of a system or component due to increasing the surveillance test interval to 24 months. Existing system and component redundancy and operation is not being changed by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. Therefore, there are no significant reductions in a margin of safety.

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

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

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

NRC Project Director: Gail H. Marcus

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Power Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: August 22, 1996

Description of amendment request: The proposed amendment would revise the license for each unit and the bases for Technical Specification (TS) Section 15.3.1, ''Reactor Coolant System.'' The licensed power level would be changed from 1518 to 1518.5 megawatts thermal to agree with other sections of the 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. Operation of this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

There is no physical change to the facilities as a result of the proposed license amendment and all Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications remain unchanged. The proposed change is administrative only and restores consistency within the PBNP license and licensing basis. Therefore, this amendment will not cause a significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment has no effect on the physical configuration of the facilities or the manner in which they operate. The design and design basis of the plants remains the same. The current plant safety analysis therefore remains complete and accurate in addressing the design basis events and in analyzing plant response and consequences for the facilities. The Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications for the facilities are not affected by the proposed license amendment. The plant conditions for which the design basis accident analysis have been performed remain valid. Therefore, the proposed license amendment cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

Plant safety margins are established through the Limiting Conditions for

Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications. Since there is no change to the physical design or operation of the plant, there is no change to any of these margins. Thus, the proposed license amendment does not involve a reduction in any margin of safety.

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

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: Gail H. Marcus

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

Consumers Power Company, Palisades Plant, Van Buren County, Michigan

Date of amendment request: December 11, 1995, supplemented by letters dated January 18, 1996, and September 3, 1996Brief

Description of amendment request: The proposed amendment would revise the Palisades Technical Specifications (TS) Administrative Controls section (Section 6) and other TS associated with the administrative controls section to adopt the format of NUREG-1432, ''Standard Technical Specifications, Combustion Engineering Plants.'' The amendment would also revise certain other surveillance intervals and administrative requirements.Date of individual notice in the Federal Register: September 20, 1996 (61 FR 49493)

Expiration date of individual notice: October 21, 1996

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423

Consumers Power Company, Palisades Plant, Van Buren County, Michigan

Date of amendment request: August 14, 1996 (also refer to related application dated January 18, 1996)

Brief Description of amendment request: The proposed amendment would revise the Palisades Technical Specifications (TS) to extend the surveillance interval frequency for the primary coolant pump (PCP) flywheels by one operating cycle. By letter dated January 18, 1996, the licensee previously submitted a request to amend the TS to delete the requirement to perform PCP flywheel inspections. NRC review of the original request will not be completed in time for the upcoming refueling outage scheduled for November 1996; therefore, the licensee has submitted this separate request to extend the surveillance frequency by one operating cycle.Date of individual notice in the Federal Register: September 24, 1996 (61 FR 50054

Expiration date of individual notice: October 24, 1996

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 9, 1996, as supplemented by letter dated August 27, 1996

Brief description of amendment request: The amendment would revise the Technical Specifications to allow the surveillance of the relief mode of operation of each of the 20 safety/relief valves without physically lifting the disk off the seat at power.Date of individual notice in the Federal Register: September 11, 1996 (61 FR 47971)

Expiration date of individual notice: October 11, 1996

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: August 27, 1996

Description of amendment request: The proposed amendment would clarify the Technical Specifications limiting condition for operation and surveillance requirements for the charging pumps and high pressure safety injection pumps when the unit is shut down (Modes 5 and 6). Date of publication of individual notice in Federal Register: September 20, 1996 (61 FR 49498)

Expiration date of individual notice: October 21, 1996

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut

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

Date of amendment request: August 27, 1996

Brief description of amendment request: This amendment proposes to delete License Condition 2.C.(24)(a) for Unit 2 which required establishment by June 3, 1981, of regularly scheduled 8-hour shifts without reliance on routine use of overtime. The proposed amendment also modifies Technical Specification 6.2.2 for both units to incorporate limits on overtime.

Date of publication of individual notice in Federal Register: September 12, 1996 (61 FR 48175)

Expiration date of individual notice: October 15, 1996

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: August 16, 1996

Brief description of amendments: The amendments would defer the implementation date as stated in Amendment No. 150 for Dresden, Unit 2, and Amendment No. 145 for Dresden, Unit 3, until January 15, 1997.

Date of issuance: September 26, 1996

Effective date: Immediately, to be implemented on or before January 15, 1997.

Amendment Nos.: 151 and 146

Facility Operating License Nos. DPR-19 and DPR-25: The amendments revised the implementation date.

Date of initial notice in Federal Register: August 22, 1996 (61 FR 43391) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 26, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

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

Date of application for amendments: April 9, 1996

Brief description of amendments: The amendments revise the Technical Specifications to eliminate the main steamline radiation monitoring system high radiation trip function for initiating an (1) automatic reactor scram, (2) automatic closure of the main steamline isolation valves, and (3) automatic closure of the reactor recirculation water sample line isolation valves and main steam line drain isolation valves.

Date of issuance: September 20, 1996

Effective date: Immediately, to be implemented within 90 days.

Amendment Nos.: 115 and 100

Facility Operating License Nos. NPF-11 and NPF-18: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25701) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 20, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: February 6, 1996

Brief description of amendment: The amendment deletes the requirement to perform alternate train testing of redundant components when emergency core cooling system and containment cooling system components are found to be inoperable or are to be removed from service for maintenance.

Date of issuance: September 26, 1996

Effective date: September 26, 1996

Amendment No.: 172

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

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28611) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 26, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423

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

Date of application for amendments: November 15, 1995

Brief description of amendments: The amendments revise the Technical Specifications to modify Section 3/4.7.5, ''Standby Nuclear Service Water Pond,'' for the Catawba Nuclear Station, Units 1 and 2, raising the minimum water level by 1 foot (from elevation 570 to 571 feet).

Date of issuance: September 20, 1996

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

Amendment Nos.: 152 and 144

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65676) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 20, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

Date of application for amendment: April 29, 1996, as supplemented September 12, 1996

Brief description of amendment: This amendment revises TS 5.3.1 to allow the use of ZIRLO as an alternate zirconium-based fuel rod material and removes the word ''clad'' to be consistent with the text of the NRC's improved Standard Technical Specifications (NUREG-1431). Limited substitution of fuel rods by ZIRLO filler rods is permitted. The proposed revision to Note 2 on TS Table 3.9-1 to specify that the maximum burnup in the peak fuel rod in a fuel assembly stored in Region 2 spent fuel racks should not exceed the NRC-approved limit for WCAP-12610 was withdrawn by letter dated September 12, 1996.

Date of issuance: September 13, 1996

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

Amendment No: 82

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

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25703) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1996.No significant hazards consideration comments received: No

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

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

Date of amendment request: May 19, 1995, as supplemented by letters dated July 21, 1995, and June 10, September 10 and 13, 1996

Brief description of amendment: The amendment revises the technical specifications to permit the reactor building personnel airlock doors to remain open during fuel handling.

Date of issuance: September 20, 1996

Effective date: September 20, 1996

Amendment No.: 184

Facility Operating License No. DPR-51. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39437) The additional information contained in the supplemental letters dated July 21, 1995, and June 10, September 10 and 13, 1996, were clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 20, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

Date of application for amendment: April 4, 1995, as supplemented April 25 and August 19, 1996

Brief description of amendment: The amendment revised operating criteria and requirements associated with containment personnel air locks.

Date of issuance: September 26, 1996

Effective date: September 26, 1996

Amendment No.: 175

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

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39438) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 26, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

Date of application for amendments: December 18, 1995, as supplemented on May 3, June 11, July 1, July 3, and August 22, 1996.

Brief description of amendments: The amendments increase the authorized rated thermal power from 2200 Megawatt-thermal (MWt) to 2300 MWt. The amendment also approves changes to the Technical Specifications to implement uprated power operation.

Date of issuance: September 26, 1996

Effective date: September 26, 1996, to be implemented within 120 days

Amendment Nos. 191 and 185Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34889) The initial Federal Register notice included information from the licensee's May 3 and May 11, 1996 supplemental letters. The July 1, July 3, and August 22, 1996 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in an Environmental Assessment dated September 12, 1996 and in a Safety Evaluation dated September 26, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

Date of application for amendment: May 15, 1996

Brief description of amendment: The amendment revises Technical Specification 3/4.3.2, ''Isolation Actuation Instrumentation,'' to establish a range of allowable values and trip setpoints for high temperatures in the Main Steam Line Tunnel Lead Enclosure.

Date of issuance: September 17, 1996

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

Amendment No.: 77

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34893) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 17, 1996.No significant hazards consideration comments received: No

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

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: January 16, 1996

Brief description of amendment: The amendment changes the Technical Specification (TS) Limiting Condition for Operation Section 3.6.1 and Surveillance Requirement Section 4.6.1, ''Primary Containment,'' and the corresponding Bases, as well as, adds Administrative Controls Section 6.19, ''Containment Leakage Rate Testing Program.'' These changes will allow the use of the performance-based containment leakage testing requirements described in 10 CFR Part 50, Appendix J, Option B, for Type B, for Type A, B, and C testing.

Date of issuance: September 20, 1996

Effective date: As of the date of issuance.

Amendment No.: 203

Facility Operating License No. DPR-65. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5816) as corrected on February 29, 1996 (61 FR 7825)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 20, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

Northern States Power Company, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: June 22, 1995, as supplemented August 10, 1995, and March 26, 1996

Brief description of amendment: The amendment modifies the Technical Specification requirements for avoidance and protection from thermal-hydraulic instabilities to be consistent with the previously approved Boiling

Water Reactor Owners Group long-term solution Option I-D described in the Licensing Topical Report, ''BWR Owners Group Long-Term Stability Solutions Licensing Methodology (NEDO-31960),'' dated June 1991, and Supplement 1 to NEDO-31960, dated March 1992. The amendment also adds the fuel cycle dependent stability power and flow limits in the Core Operating Limits Report.

Date of issuance: September 17, 1996

Effective date: September 17, 1996, with full implementation within 60 days

Amendment No.: 97

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

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45181) The August 10, 1995, and March 26, 1996, letters provided a nonproprietary version of the topical report GENE-637-043-0295 and clarifying information, respectively. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. Therefore, renoticing was not warranted. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 17, 1996.No significant hazards consideration comments received: No.

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

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

Date of amendment request: May 31, 1995

Brief description of amendment: The amendment revised the technical specifications to require additional restrictions on the component cooling water system heat exchangers.

Date of issuance: September 19, 1996

Effective date: September 19, 1996

Amendment No.: 175

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

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35083) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 19, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Power Authority of The State of New York, Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: July 12, 1996, as supplementedAugust 19, 1996, and August 21, 1996.

Brief description of amendment: The amendment extends the surveillance interval for certain instruments from 18 to 24 months.

Date of issuance: September 24, 1996

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

Amendment No.: 169

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42282) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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

Date of application for amendments: May 3, 1996 (TS 352)

Brief description of amendments: The amendments provide administrative changes to the technical specifications.

Date of issuance: September 18, 1996

Effective Date: September 18, 1996

Amendment Nos.: 231, 246 and 206

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42284) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 18, 1996.No significant hazards consideration comments received: None

Local Public Document Room location: Athens Public library, 405 E. South Street, Athens, Alabama 35611

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

Date of application for amendment: June 28, 1996, as supplementedAugust 30, 1996.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to increase the required shutdown margin. It also revises TSs associated with this shutdown margin increase to allow calculational determination of the highest worth control rod and to relax the action requirements in the event the required shutdown margin is not met. The amendment also makes appropriate editorial changes and minor editorial corrections to the affected TSs.

Date of issuance: September 25, 1996

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

Amendment No.: 148

Facility Operating License No. DPR-28. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40031) The August 30, 1996, letter provided clarifying information that did not change the scope of the application or affect the initial determination. The Commission's related evaluation of the amendment is contained in aSafety Evaluation dated September 25, 1996.No significant hazards consideration comments received: No

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

Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: April 25, 1995

Brief description of amendment: The amendment adds a reactor water cleanup system high blowdown containment isolation trip function and associated limiting condition for operation and surveillance requirements to the Technical Specifications.

Date of issuance: September 19, 1996Effective date: September 19, 1996, to be implemented within 30 days of issuance.

Amendment No.: 147

Facility Operating License No. NPF-21: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 28, 1996 (61 FR 33777) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 19, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352

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

Date of application for amendment: June 10, 1996, as supplemented on August 27, and September 5, 1996.

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant Technical Specification 4.2.b, ''Steam Generator Tubes,'' and its associated basis, by allowing the use of Westinghouse laser-welded sleeves to repair defective steam generator tubes.

Date of issuance: September 24, 1996

Effective date: September 24, 1996, and is to be implemented within 30 days of the date of issuance.

Amendment No.: 127

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34902) The August 27, 1996, submittal increased the TS required sample size for in-service inspection of repaired tubes in both SGs. The September 5, 1996, submittal incorporated the EPRI guidelines for SG inspection scope expansion for repaired SG tubes into the TS. These submittals provided clarifying information and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

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

Date of application for amendment: May 1, 1996 as supplemented on May 31, August 14, August 26, and September 11, 1996.

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant Technical Specification 4.2.b, ''Steam Generator Tubes,'' its associated bases, and Figure TS 4.2-1 by redefining the pressure boundary for Westinghouse mechanical hybrid expansion joint (HEJ) steam generator (SG) tube sleeves.

Date of issuance: September 25, 1996

Effective date: September 25, 1996, and is to be implemented within 30 days of the date of issuance.

Amendment No.: 128

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

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25715) The May 31, August 14, August 26, and September 11, 1996, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 25, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Dated at Rockville, Maryland, this 2nd day of October 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

Director, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation

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Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a proposed revision of a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide is a proposed Revision 3 to Regulatory Guide 1.8, and it is temporarily identified as DG–1012, ''Qualification and Training of Personel for Nuclear Power Plants.'' The guide will be in Division 1, ''Power Reactors.'' This regulatory guide is being revised to provide current guidance acceptable to the NRC staff regarding qualifications and training for nuclear power plant personnel. This regulatory guide would endorse ANSI/ANS–3.1–1993, ''Selection, Qualification and Training of Personnel for Nuclear Power Plants.''

The draft guide has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the guide. Comments should be accompanied by supporting data. Written comments may be submitted to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by November 15, 1996.

Although a time limit is given for comments on this draft guide, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

Comments may be submitted electronically, in either ASCII text or Wordperfect format (version 5.1 or later), by calling the NRC Electronic Bulletin Board on FedWorld. The bulletin board may be accessed using a personal computer, a modem, and one of the commonly available communications software packages, or directly via Internet.

If using a personal computer and modem, the NRC subsystem on FedWorld can be accessed directly by dialing the toll free number: 1–800–303–9672. Communication software parameters should be set as follows: parity to none, data bits to 8, and stop bits to 1 (N,8,1). Using ANSI or VT–100 terminal emulation, the NRC NUREGs and RegGuides for Comment subsystem can then be accessed by selecting the ''Rules Menu'' option from the ''NRC Main Menu.'' For further information about options available for NRC at FedWorld, consult the ''Help/Information Center'' from the ''NRC Main Menu.'' Users will find the ''FedWorld Online User's Guides'' particularly helpful. Many NRC subsystems and data bases also have a ''Help/Information Center'' option that is tailored to the particular subsystem.

The NRC subsystem on FedWorld can also be accessed by a direct dial phone number for the main FedWorld BBS, 703–321–3339, or by using Telnet via Internet, fedworld.gov. If using 703–321–3339 to contact FedWorld, the NRC subsystem will be accessed from the main FedWorld menu by selecting the ''Regulatory, Government Administration and State Systems,'' then selecting ''Regulatory Information Mall.'' At that point, a menu will be displayed that has an option ''U.S. Nuclear Regulatory Commission'' that will take you to the NRC Online main menu. The NRC Online area also can be accessed directly by typing ''/go nrc'' at a FedWorld command line. If you access NRC from FedWorld's main menu, you may return to FedWorld by selecting the ''Return to FedWorld'' option from the NRC Online Main Menu. However, if