the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the DBNPS.

Agencies and Persons Consulted

In accordance with its stated policy, on July 22, 1996, the staff consulted with the Ohio State official, Carol O'Claire of the Ohio Emergency Management Agency, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of no Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensees' letter dated June 28, 1996, 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 located at the University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606.

Dated at Rockville, Maryland, this 7th day of August 1996.

For the Nuclear Regulatory Commission.

Linda L. Gundrum,

Project Manager, Project Directorate III–3, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 96–20679 Filed 8–13–96; 8:45 am]

BILLING CODE 7590–01–P

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of August 12, 19, 26, and September 2, 1996.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of August 12

There are no meetings scheduled for the Week of August 12.

Week of August 19—Tentative

There are no meetings scheduled for the Week of August 19.

Week of August 26—Tentative

Monday, August 26

2:00 p.m. Meeting with Chairman of Nuclear Safety, Research Review Committee (NSRRC) (public meeting), (Contact: Jose Cortez, 301–415–6596)

Tuesday, August 27

10:00 a.m. Briefing on Design Certification Issues (public meeting), (Contact: Jerry Wilson, 301–415–3145)

2:00 p.m. Briefing on Annealing Demonstration Project (public meeting), (Contact: Michael Mayfield, 301–415–6690)

Wednesday, August 28

10:00 a.m. Briefing on Certification of USEC (public meeting), (Contact: John Hickey, 301–415–7192)

11:30 a.m. Affirmation Session (public meeting) (if needed).

Week of September 2—Tentative

Thursday, September 5

10:30 a.m. Briefing by DOE on Status of HLW Program (public meeting)

The schedule for commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Bill Hill (301) 415–1661.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1963).

In addition, distribution of this meeting notice over the internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to alb@nrc.gov or dkw@nrc.gov.

* * * * *

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 96–20828 Filed 8–2–96; 11:03 am]

BILLING CODE 7590–01–M

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 July 20, 1996, through August 2, 1996. The last biweekly notice was published on July 31, 1996 (61 FR 40013).

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 September 13, 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: July 26, 1996

Description of amendments request: The proposed amendment will revise the appropriate Technical Specifications and their Bases to permit the electrosleeving repair technique developed by Framatome Technologies, Inc. to be used at Calvert Cliffs Nuclear Power Plant (CCNPP). Electrosleeving is a steam generator tube repair method where an ultra-fine grained nickel is electrochemically deposited on the inner surface of a tube to form a structural repair of the degraded tube. The electrodeposition of nickel provides a continuous metallurgical bond that eliminates all leak paths and macro-crevices. The electroformed sleeve provides a structural, leak-tight seal, without deforming or changing the microstructure of the parent tube. Thus, unlike the conventional welded sleeves, electrosleeving does not require a post-installation stress relief.

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

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

The implementation of the proposed steam generator tube electrosleeving has been reviewed for impact on the current CCNPP licensing basis.

Since the electrosleeve is designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code as guidance, it meets the objectives of the original steam generator tubing. The applied stresses and fatigue usage for the electrosleeve are bounded by the limits established in the ASME Code. American Society of Mechanical Engineers Code minimum material property values are used for the structural and plugging limit analysis. Mechanical testing has shown that the structural strength of nickel electrosleeves under normal, upset and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Regulatory Guide 1.121. Burst testing of electrosleeved tubes has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

As in the original tube, the electrosleeve Technical Specification depth-based plugging limit is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. A bounding tube wall degradation growth rate per cycle and a nondestructive examination uncertainty has been assumed for determining the electrosleeve plugging limit.

Evaluation of the proposed electrosleeved tubes indicates no detrimental effects on the electrosleeve or electrosleeve-tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Calvert Cliffs. Corrosion testing of electrosleeve-tube assemblies indicates no evidence of electrosleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed electrosleeve has no significant effect on either the configuration of the plant, or the manner in which it is operated. The hypothetical consequences of failure of the electrosleeved tube is bounded by the current steam generator tube rupture analysis described in Section 14.15 of the Calvert Cliffs Updated Final Safety Analysis Report. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system.

Therefore, BGE [Baltimore Gas and Electric] has concluded that 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 kind of accident from any other accident previously evaluated.

As discussed above, the electrosleeve is designed using the applicable ASME Code as guidance; therefore, it meets the objectives of the original steam generator tubing. As a result, the functions of the steam generators will not be significantly affected by the installation of the proposed electrosleeve. Adhesion and ductility tests performed per ASTM [American Society for Testing and Materials] standards verified that the electrosleeve will not fail by de-bonding or cracking. In addition, the proposed electrosleeve does not interact with any other plant systems. Any accident as a result of potential tube or electrosleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. The continued integrity of the installed electrosleeve is periodically verified by the Technical Specification requirements.

The implementation of the proposed electrosleeves has no significant effect on either the configuration of the plant, or the manner in which it is operated. Therefore, BGE concludes that this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The repair of degraded steam generator tubes via the use of the proposed electrosleeve restores the structural integrity of the faulted tube under normal operating and postulated accident conditions. The design safety factors utilized for the electrosleeve are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original steam generator design. The repair limit for the proposed electrosleeve is consistent with that established for the steam generator tubes. The portions of the installed electrosleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of electrosleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. Use of the previously identified design criteria and design verification testing assures that the margin to safety with respect to the implementation of the proposed electrosleeve is not significantly different from the original steam generator tubes.

Therefore, BGE concludes that the proposed changes 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 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: Jocelyn A. Mitchell, Acting Director

Carolina Power & Light Company, et al., Docket No. 50-325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: April 8, 1996, as supplemented on July 30, 1996. This notice supersedes the Federal Register notice published on June 5, 1996 (61 FR 28607).

Description of amendment request: The licensee has proposed to revise the Technical Specifications (TS) to include the following changes: 1. The Minimum Critical Power Ratio (MCPR) Safety Limit specified in TS 2.1.2 from 1.07 to 1.10 for Unit 1 Cycle 11 operation; TS 5.3.1 to reflect the new fuel type (GE13) that will be inserted during Unit 1 Refueling Outage 10; 2. The acceptable range of sodium pentaborate concentration for the standby liquid control system shown in TS Figure

3.1.5-1 to reflect changes to poison material concentration needed to achieve reactor shutdown based on the new GE13 fuel type.

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

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

Proposed Change 1:

The proposed license amendment will allow the loading and use of GE13 fuel assemblies in the Brunswick Unit 1 reactor core. The use of GE13 fuel assemblies requires that the safety limit minimum critical power ratio value also be revised. The safety limit minimum critical power ratio is established to maintain fuel cladding integrity during operational transients. The GE13 fuel assembly design has been analyzed using methods that have been previously approved by the Nuclear Regulatory Commission and documented in General Electric Nuclear Energ's reload licensing methodology Topical Report NEDE-24011, ''General Electric Standard Application for Reactor Fuel (GESTAR II).''Based on a cycle-specific calculation performed by General Electric, a safety limit minimum critical power ratio value of 1.10 has been established for the GE13 fuel type for Brunswick Unit 1 Cycle 11 operation. The cycle-specific calculation has been performed in accordance with the methodology in Revision 12 of NEDE-24011. This cycle-specific calculation has demonstrated that a safety limit minimum critical power ratio value of 1.10 will ensure that 99.9 percent of the fuel rods avoid boiling transition during a transient event when all uncertainties are considered. The safety limit minimum critical power ratio value of 1.10 assures that fuel cladding protection equivalent to that provided with the existing safety limit minimum critical power ratio value is maintained. This ensures that the consequences of previously evaluated accidents are not significantly increased.

The proposed revision of the safety limit minimum critical power ratio does not alter any plant safety-related equipment, safety function, or plant operations that could change the probability of an accident. The change does not affect the design, materials, or construction standards applicable to the fuel bundles in a manner that could change the probability of an accident.

Proposed Change 2:

The standby liquid control system provides a means of reactivity control that is independent of the normal reactivity control system. The standby liquid control system must be capable of assuring that the reactor core can be placed in a subcritical condition at any time during reactor core life. Technical Specification Figure 3.1.5-1 specifies the acceptable range of concentrations and volumes for sodium pentaborate solution used as a neutron absorber (i.e., for reactivity control). The portion of the sodium pentaborate concentration range shown in Technical Specification Figure 3.1.5-1 applicable to the lower range of tank volumes is being revised to increase the required concentration of sodium pentaborate solution. This change is needed to account for the additional shutdown reactivity needed based on the planned use of GE13 fuel assemblies as reload fuel for the Unit 1 reactor core. Since the standby liquid control system is independent from the normal means of controlling reactor core reactivity and not used to control core reactivity during normal plant operations, the proposed revision to the sodium pentaborate concentration curve for the standby liquid control system does not alter any plant safety-related equipment, safety function, or plant operations that could change the probability of an accident.

The current volume-concentration range of sodium pentaborate used in the standby liquid control system will achieve a sufficient concentration of boron in the reactor vessel to ensure reactor shutdown. Based on the increased reactivity of the new GE13 reload fuel assemblies, the required sodium pentaborate volume-concentration range is being revised to ensure sufficient neutron absorbing solution is available to achieve reactor shutdown; therefore, the consequences of an accident previously evaluated are not significantly increased.

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

Proposed Change 1:

The GE13 fuel assembly has been designed and complies with the acceptance criteria contained in General Electric Nuclear Energy's standard application for reactor fuel (GESTAR-II), which provides the latest acceptance criteria for new General Electric fuel designs. The similarity of the GE13 fuel design to the previously accepted GE11 fuel design, in conjunction with the increased critical power capability of the GE13 fuel design, ensure that no new mode or condition of plant operation is being authorized by the loading and use of the GE13 fuel type. The proposed revision of the safety limit minimum critical power ratio from 1.07 to 1.10 does not modify any plant controls or equipment that will change the plant's responses to any accident or transient as given in any current analysis. Therefore, the proposed change to allow the loading and use of the GE13 fuel type and the revision of the safety limit minimum critical power ratio value from 1.07 to 1.10 will not create the possibility for a new or different kind of accident from any accident previously evaluated.

Proposed Change 2:

As discussed above, the standby liquid control system provides a means of reactivity control that is independent of the normal reactivity control system and is capable of assuring that the reactor core can be placed in a subcritical condition at any time during reactor core life. The proposed revision to the sodium pentaborate concentration range does not modify the standby liquid control system or its controls, does not modify other plant systems and equipment, and does not permit a new or different mode of plant operation. As such, the proposed revision to the minimum pentaborate concentration value does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Proposed Change 1:

As previously discussed, the GE13 fuel assembly design has been analyzed using methods that have been previously approved by the Nuclear Regulatory Commission and documented in General Electric Nuclear Energy's reload licensing methodology Topical Report NEDE-24011, ''General Electric Standard Application for Reactor Fuel (GESTAR II).''The safety limit minimum critical power ratio value is selected to maintain the fuel cladding integrity safety limit (i.e., that 99.9 percent of all fuel rods in the core are expected to avoid boiling transition during operational transients). Appropriate operating limit minimum critical power ratio values are established, based on the safety limit minimum critical power ratio value, to ensure that the fuel cladding integrity safety limit is maintained. The operating limit minimum critical power ratio values are incorporated in the Core Operating limits Report as required by Technical Specification 6.9.3.1.

Based on the cycle-specific calculation performed by General Electric, a safety limit minimum critical power ratio value of 1.10 has been established for the GE13 fuel type for Unit 1 Cycle 11 operation. This cycle-specific calculation has been performed based on the methodology contained in Revision 12 of NEDE-24011-P-A. The new GE13 safety limit minimum critical power ratio value of 1.10 for Unit 1 Cycle 11 operation is based on the same fuel cladding integrity safety limit criteria as that for the GE11 safety limit minimum critical power ratio (i.e., that 99.9 percent of all fuel rods in the core are expected to avoid boiling transition during operational transients); therefore, the proposed change does not result in a significant reduction in the margin of safety.

Proposed Change 2:

As previously stated, the purpose of the standby liquid control is to inject a neutron absorbing solution into the reactor in the event that a sufficient number of control rods cannot be inserted to maintain subcriticality. Sufficient solution is to be injected such that the reactor will be brought from maximum rated power conditions to subcritical over the entire reactor temperature range from maximum operating to cold shutdown conditions. General Electric methodology establishes a fuel type dependent standby liquid control system shutdown margin to account for calculational uncertainties. General Electric calculations show that an in-vessel concentration of 660 ppm will provide a standby liquid control system minimum shutdown margin in excess of the 3.2% delta k value required for the GE13 fuel. To achieve an in-vessel concentration of 660 ppm, the acceptable range of standby liquid control system tank concentrations is being revised for the lower range of tank volumes. Thus, the proposed revision of the standby liquid control system sodium pentaborate volume-concentration range ensures that there will not be a significant reduction in the amount of available shutdown margin and, therefore, not a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Eugene V. Imbro

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

Date of amendment request: June 21, 1996

Description of amendment request: The proposed amendments would extend the surveillance interval for TS 4.7.2.b and 4.7.2.d related to testing of the Control Room Emergency Filtration System from 18 months to 24 months. The amendments would also include a one-time extension of the allowed outage time for the Control Room and Auxiliary Electric Equipment Room Emergency Filtration System to allow each subsystem to be inoperable for up to 30 days during modifications to replace the existing deep bed charcoal absorbers with tray-type units.

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

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

This Technical Specification change does not involve accident initiators or initial accident assumptions. The Control Room and Auxiliary Equipment Room Emergency Filtration System (CREFS) trains A and B are post-accident atmospheric cleanup components that are designed to limit the radiation exposure to personnel occupying the Control Room to 5 rem or less whole body during and following all design basis accident conditions. Therefore, this Technical Specification change does not increase the probability of occurrence of an accident previously evaluated.

CREFS trains A and B are utilized to control the onsite dose to personnel in the Control Room. This Technical Specification change extends the [Limiting Condition for Operation] LCO duration for allowing each train to be inoperable one at a time from 7 days to 30 days total for the current surveillance interval. This change is a one time change to allow for the repair/replacement work associated with the corroded filter unit charcoal retaining screens in the high efficiency charcoal adsorber section of each train. The...normal preventative maintenance and testing [will] be performed on the operable CREFS train just prior to taking the [opposite] filter train out of service for the modification. This action will ensure that the remaining subsystem is operable and ensure maximum reliability of the system. The Technical Specification change will not affect onsite dose if a [design-basis accident] DBA occurs and the operating filter unit does not fail. The operable filter unit will be sufficient to maintain the operating areas habitable. The original LCO allowed 7 day operation with only one operable train and is also susceptible to a single failure during the Allowed Outage Time. The probability that a DBA will occur coupled with the single failure of the operable train during the extended allowed outage time per the Technical Specification change is the same order of magnitude as for the current 7 day allowed outage time. Therefore, this change does not increase the consequences of an accident previously evaluated.

The extension of the surveillance interval from 18 months to 24 months extends the maximum interval between TS surveillances of the filter trains from 22.5 months to 30 months. The equipment that is affected are the CREFS filter trains A and B, which are comprised of HEPA filters, heaters, charcoal adsorbers, and fans. This equipment has a history of satisfactory surveillance testing (in-place testing and laboratory analysis of charcoal), and has had little maintenance problems for the past 5 years. Although the SER Section 6.4.1 and the [Regulatory Guide] RG 1.52 state that the units shall be tested every 18 months, a review of the basis documents for the testing (ANSI N510) shows that the 1975 edition recommended annual testing and later editions (1980 and 1989) state that testing be performed ''at least once every operating cycle''. Therefore the extension of the surveillance intervals from 18 months to 24 months will not increase the consequences of an accident previously evaluated.

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

This Technical Specification change will allow each train of CREFS to be inoperable one at a time for up to 30 days to repair/replace charcoal retaining screens and changes surveillance intervals from 18 months to 24 months. Prior to the extended LCO on a given train, the scheduled monthly surveillance and preventive maintenance will be performed. This Technical Specification change does not involve components that are accident initiators and therefore will not create a new or different kind of accident than those previously analyzed.

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

The purpose of CREFS trains A and B are to control the onsite dose to personnel in the Control Room following an accident that involves a potential radiological release. Redundant filter trains are utilized to ensure that a single active failure will not impact the ability of the system to perform its safety function. Since the probability of an accident occurring during the extended Technical Specification LCO for the inoperable train in conjunction with the probability that the operable CREFS train will fail is the same order of magnitude as for the current LCO, then the proposed Technical Specification change has minimal impact on the safe operation of the plant. The CREFS trains were both determined operable following their last surveillance and no events have occurred at the plant to indicate that they may be inoperable. Normal preventative maintenance and testing will be performed on the operable CREFS train just prior to taking the [opposite] filter train out of service for the modification. This action will ensure that the remaining subsystem is operable and ensure maximum reliability of the system. The change in surveillance intervals from 18 months to 24 months will not cause a significant reduction in the margin of safety, because the previous five surveillances have been satisfactory and the equipment/components do not have a tendency to drift over time. Therefore, the proposed amendment will not significantly impact the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra

Dairyland Power Cooperative (DPC), Docket No. 50-409, LaCrosse Boiling Water Reactor (LACBWR), Vernon County, Wisconsin

Date of amendment request: April 10, 1996

Description of amendment request: The proposed amendment would update the facility Possession Only License and Technical Specifications to reflect the permanently shutdown and defueled condition of the plant.

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:

DPC proposes to modify the LACBWR Technical Specifications to more accurately reflect the permanently shutdown, defueled, possession-only status of the facility.

Analysis of no significant hazards consideration:

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

The proposed changes delete system requirements that are no longer necessary to prevent, or mitigate the consequences of, a credible SAFSTOR accident as described in our current SAFSTOR Accident Analysis.

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

The proposed changes are either administrative in nature or were made based on the analysis of previously evaluated accident scenarios. In no other way do they change the design or operation of the facility and therefore do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The changes incorporate into the proposed Technical Specifications the margin of safety associated with the current SAFSTOR accident analysis and thus don't involve a significant reduction in the margin of safety.

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

Local Public Document Room location: LaCrosse Public Library, 800 Main Street, LaCrosse, Wisconsin 54601.

Attorney for licensee: Wheeler, Van Sickle and Anderson, Suite 801, 25 West Main Street, Madison, Wisconsin 53703-3398

NRC Project Director: Seymour H. Weiss

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

Date of amendment request: July 25, 1996 (NRC-96-0064)

Description of amendment request: The proposed amendment would relocate or delete a number of items currently in the Administrative Controls Section (Section 6.0) of the technical specifications (TS). This submittal revises a previous submittal dated December 15, 1994 (NRC-94-0107), to modify the proposed TS change to be consistent with NRC Administrative Letter 95-06, ''Relocation of Technical Specifications Administrative Controls Related to Quality Assurance,'' the Improved Standard TS (ISTS), and pending changes to the ISTS. The previous submittal was noticed in the Federal Register on June 6, 1995 (60 FR 29873).

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes are administrative in nature. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR [Updated Final Safety Analysis Report] transient analyses. No Limiting Condition for Operation, ACTION statement or Surveillance Requirement is affected by any of the proposed changes.

Also, these proposed changes, in themselves, do not reduce the level of qualification or training such that personnel requirements would be decreased. Therefore, this change is administrative in nature and does not involve a significant increase in the probability or consequences of an accident previously evaluated. Further, the proposed changes do not alter the design, function, or operation of any plant component and therefore, do not affect the consequences of any previously evaluated accident.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new mode of plant operation, surveillance requirement or involve a physical modification to the plant. The proposed changes are administrative in nature. The changes propose to revise, delete or relocate the stated administrative control provisions from the TS to the UFSAR, plant procedures or the QA [Quality Assurance] Program whereby, adequate control of information is maintained. Further, as stated above, the proposed changes do not alter the design, function, or operation of any plant components and therefore, no new accident scenarios are created.

3. The proposed changes do not involve a significant reduction in a margin of safety because they are administrative in nature. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR transient analyses. No Limiting Condition for Operation, ACTION statement or Surveillance Requirement is affected. The proposed changes do not involve a significant reduction in a margin of safety. Additionally, the proposed change does not alter the scope of equipment currently required to be OPERABLE or subject to surveillance testing nor does the proposed change affect any instrument setpoints or equipment safety functions. Therefore, the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226

NRC Project Director: Mark Reinhart

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

Date of amendment request: April 29, 1996

Description of amendment request: The proposed amendment revises the permissible values of the maximum and minimum pressurizer water levels and incorporates a graph to display these values for various operating conditions. The amendment also revises the Bases section of the Technical Specification. The Bases changes revise the acceptable value of the as-found tolerance for the settings of the pressurizer safety valves and change the value of flowrate through the pressurizer safety valves. The moderator temperature coefficient as described in the Bases Section is removed.

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 startup accident and the rod withdrawal accident have been reanalyzed to justify the proposed increase in pressurizer coder safety value as-found tolerance. The analyses establish more appropriate boundaries and re-analyze the same initiators as are currently found in the ANO-1 Safety Analysis Report. Changing the as-found setpoint tolerance does not change how the pressurizer code safety valve operates as it will continue to be reset to 2500 psig plus or minus 1% prior to reactor startup.

The acceptance criteria for these analyses are that the reactor coolant system (RCS) pressure shall not exceed the safety limit of 2750 psig (110% of design pressure and that the reactor thermal power remains below 112% Rated Power. The analyses using the proposed setpoint tolerance have shown that the acceptance criteria were met and that the consequences of the events were essentially the same as those in the ANO-1 SAR. Analyses were performed to determine the pressurizer maximum water level that would prevent the RCS from exceeding the safety limit of 2750 psig in the event of either a startup accident or a rod withdrawal accident. More appropriate pressurizer level requirements have been incorporated in accordance with these analyses.

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 proposed changes introduce no new mode of plant operation. The reanalysis of the startup accident and the rod withdrawal accident were performed using methodologies identical to that employed in the ANO-1 SAR and an improved computer code (RELAP5/MOD2). The pressurizer code safety valve setpoint will continue to be reset at 2500 psig plus or minus 1% prior to reactor startup and will continue to function to maintain RCS pressure below the safety limit of 2750 psig. Analyses were performed to determine the pressurizer maximum water level that would prevent the RCS from exceeding the safety limit of 2750 psig in the event of either a startup accident or a rod withdrawal accident. More appropriate pressurizer level requirements have been incorporated in accordance with these analyses.

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 safety function of the pressurizer code safety valves is not altered as a result of the proposed change in setpoint tolerance. The reanalysis of the startup accident and rod withdrawal accident have shown that with a plus or minus 3% setpoint tolerance, the pressurizer code safety valves will function to limit RCS pressure below the safety limit of 2750 psig. The sensitivity studies for the startup accident showed the acceptance criteria would still be met even if one pressurizer code safety valve lifted at 5% above 2500 psig at startup conditions. Additional analyses were performed to determine the pressurizer maximum water level that would prevent the RCS from exceeding the safety limit of 2750 psig in the event of either a startup accident or a rod withdrawal accident.

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

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

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, N.W., Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2), Pope County, Arkansas

Date of amendment request: June 28, 1996

Description of amendment request: The proposed amendments would remove the Unit 1 and Unit 2 Technical Specification requirements to secure the containment equipment hatch during core alterations or fuel handling.

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

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

The proposed change would allow the containment equipment hatch door to remain open during fuel movement and core alterations. This door is normally closed during this time period in order to prevent the escape of radioactive material in the event of a fuel handling accident. This door is not an initiator of any accident. The probability of a fuel handling accident is unaffected by the position of the containment equipment hatch door. The current fuel handling analysis, which has been approved by the Staff for ANO-2 and submitted for ANO-1, calculates maximum offsite doses to be well within the limits of 10 CFR Part 100. The current fuel handling accident analysis results in maximum offsite doses of 63.6 and 41.8 Rem to the Thyroid and 0.902 and 0.598 Rem to the whole body (sum of beta and gamma) for ANO-1 and ANO-2, respectively. This analysis assumes the entire release from the damaged fuel is allowed to migrate to the site boundary unobstructed. Therefore, allowing the equipment hatch doors to remain open results in no change in consequences. Also, the calculated doses during a fuel handling accident would be considerably larger than the actual doses since the calculation does not incorporate the closing of the equipment hatch door following evacuation of containment. The proposed change would significantly reduce the dose to workers in the containment in the event of a fuel handling accident by expediting the containment evacuation process. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not involve the addition or modification of any plant equipment. Also, the proposed change would not alter the design, configuration, or method of operation 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.

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

The proposed change does not have the potential for an increased dose at the site boundary due to a fuel handling accident. The margin of safety as defined by 10 CFR Part 100 has not been significantly reduced. Closing the equipment hatch door following an evacuation of containment further reduces the offsite doses in the event of a fuel handling accident and provides additional margin to the calculated offsite doses. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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, N.W., 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 12, 1996

Description of amendment request: The proposed amendment would change Technical Specification (TS) Sections 6.2.2.h and 6.2.2.i. To provide adequate shift coverage without routine heavy use of overtime, TS Section 6.2.2.h specifies an objective to have operating personnel work ''a normal 8-hour day, 40-hour week'' while the facility is operating. The proposed amendment would change the objective to ''an 8 to 12 hour day, nominal 40-hour week.''

TS Section 6.2.2.i currently states, ''The General Supervisor Operations, Supervisor Operations, Station Shift Supervisor Nuclear, and Assistant Station Shift Supervisor Nuclear shall hold senior reactor operator licenses.'' The proposed amendment would change this section to state, ''The

Manager Operations, Station Shift Supervisor Nuclear and Assistant Station Shift Supervisor Nuclear shall hold senior reactor operator licenses.'' This change is based upon a reorganization that eliminates the positions of General Supervisor Operations and Supervisor Operations from the Unit 1 Operations management structure. The responsibilities of these positions will be assumed by the Manager Operations or delegated to off-shift Senior Reactor Operators. Thus, Senior Reactor Operators will report directly to the Manager Operations.

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 consequence of an accident previously evaluated.

Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40-hour week,'' provides enhanced continuity for normal plant operations. There has been no noticeable increase in safety related problems during the trial period [The facility has been implementing 12-hour operator shifts for over 1 year on a trial basis]. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement of working hours (Generic Letter 82-12). The probability for operating personnel error due to (1) incomplete or insufficient turnover or (2) interruption of in-plant maintenance and testing is reduced. No physical plant modifications are involved, and none of the precursors of previously evaluated accidents are affected. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The assimilation of the responsibilities of the previous positions of General Supervisor Operations and Supervisor Operations into the position of Manager Operations and to off-shift Senior Reactor Operators reflects a restructuring of the operations department, and is essentially a reduction in layers of management. This proposed change does not involve any physical modification to the plant, and does not affect any precursor of a previously evaluated accident. Therefore, this change will not involve a significant increase in the probability or 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.

Establishing operating personnel hours at ''an 8 to 12-hour day, nominal 40-hour week'' provides increased flexibility in scheduling and does not adversely affect their performance. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). No physical modification of the plant is involved. As such, the change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, operation in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The responsibilities of the previous positions of General Supervisor Operations and Supervisor Operations will be assimilated into the positions of the Manager Operations and the off-shift Senior Reactor Operators. There is no physical plant modification. The change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, the change does not in itself 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.

Establishing operating personnel hours at ''an 8 to 12-hour day, nominal 40-hour week,'' provides increased flexibility in scheduling and does not adversely affect their performance. This change also decreases the risk of miscommunication between shifts by reducing the number of turnovers per day and increases operations and maintenance efficiency by promoting continuity in ongoing plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12) and is consistent with the Improved Standard Technical Specifications. The proposed change involves no physical modification of the plant, or alterations to any accident or transient analysis [...], and the changes are administrative in nature. Therefore, the change does not involve any significant reduction in a margin of safety.

The assimilation of the responsibilities of the positions of General Supervisor Operations and Supervisor Operations, into the positions of the Manager Operations and the off-shift Senior Reactor Operators, effectively reduces layers of management. The proposed change is consistent with Standard Review Plan (SRP) 13.1.2-13.1.3. This administrative transformation of the operations department management structure involves no physical modification of the plant or alterations to any accident or transient analysis. Therefore, this change in itself does not involve any 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

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: Jocelyn A. Mitchell, Acting Director

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

Date of amendment request: July 12, 1996

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 6.2.2.i. To provide adequate shift coverage without routine heavy use of overtime, TS Section 6.2.2.i specifies an objective to have operating personnel work ''a normal 8-hour day, 40-hour week'' while the facility is operating. The proposed amendment would change the objective to ''an 8 to 12 hour day, nominal 40-hour week.''

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 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequence of an accident previously evaluated.

Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40-hour week,'' allows normal plant operations to be managed more effectively and with enhanced continuity. There has been no noticeable increase in safety related problems during the trial period [The facility has been implementing 12-hour operator shifts for over 1 year on a trial basis]. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). The probability for operating personnel error due to (1) incomplete or insufficient turnover or (2) interruption of in-plant maintenance and testing is reduced. No physical plant modifications are involved, and none of the precursors of previously evaluated accidents are affected. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Establishing operating personnel hours at, ''an 8 to 12-hour day, nominal 40-hour week,'' improves the quality of life for operating personnel and does not adversely affect their performance. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). No physical modification of the plant is involved. As such, the change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, operation in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

Establishing operating personnel hours at, ''an 8 to 12-hour day, nominal 40-hour week,'' improves the quality of life for operating personnel and does not adversely affect their performance. This change also decreases the risk of miscommunication between shifts and increases operations and maintenance efficiency by promoting continuity in ongoing plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12) and is consistent with the Improved Standard Technical Specifications. The proposed change involves no physical modification of the plant, or alterations to any accident or transient analysis [...], and the changes are administrative in nature. Therefore, the change does not involve any 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: Jocelyn A. Mitchell, Acting Director

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: February 2, 1996

Description of amendment request: This request would change Technical Specification (TS) 3.6.1.2 for each unit to permit primary containment leakage testing of the main steam isolation valves (MSIVs) at either 22.5 psig or 45 psig according to the type of test to be conducted. Currently the TS only specifies 22.5 psig for the MSIVs' test pressure.

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:

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

The proposed change to the allowable test pressure for MSIV leak testing was reviewed from two perspectives. First is the potential for the change in testing pressure, and test methodology, to impact testing results. The second perspective is the potential for a failure of the testing configuration to result in undesirable consequences.

Under the proposed change, an increased test pressure of 45.0 psig (P_a) in the accident direction will be used to perform Technical Specification required MSIV leak testing. However, the acceptance criteria for testing is maintained consistent with current Technical Specifications. Therefore, the proposed change to allow a test pressure of P_a will not affect the validity of leak test results. The existing Technical Specification required leak integrity of the MSIVs will be maintained under the proposed test methodology and thus the ability of the MSIVs to act as a containment isolation valves is not affected.

The proposed test pressure of P_a will be applied in the accident direction, and will result in a back pressure being applied to the Main Steam Line (MSL) Plugs. The potential for MSL Plug ejection has been reviewed and adequate precautions have been taken to ensure that fuel damage would not result from [local leak rate test] LLRT induced MSL Plug ejection. The MSL Plugs are installed using a restraint ring which prevents inadvertent ejection. [Pennsylvania Power and Light Company] PP&L procedures require that the restraint ring be installed as a prerequisite for LLRT testing of the MSIVs at P_a. However, in the unlikely event that the MSL Plug and restraint ring were installed improperly and then subjected to back pressurization at P_a, ejection could occur. If this event did occur, the MSL Plug could hit the fuel which is an accident bounded by the fuel assembly handling accident analysis addressed in [Final Safety Analysis Report] FSAR Section 15.7.4. The MSL Plugs, MSL Plug Restraint Ring, and MSL Plug Insert and Remove Tool meet the requirements of NUREG 0612 and PP&L's Heavy Loads Program.

Therefore, the proposal to allow an alternative test pressure, P_a, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

II. This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

All components within the test volume have been evaluated for structural integrity under the proposed test pressures. In addition, pressurization of the Main Steam Line Plugs during testing will be below the evaluated pressure. The acceptance criteria for the test will be maintained, thus verification of the leak integrity of the MSIVs will not be impacted. Therefore, the proposed change to allow for an alternative test pressure of (P_a) does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. This change does not involve a significant reduction in a margin of safety.

The proposed change does not affect the acceptance criteria for the MSIV LLRT. As a result, testing at P_a in the accident direction will provide an equivalent test to that which is performed at P_a. No change in the leak integrity of the MSIVs is anticipated as a result of performing the testing at the alternative pressure. The potential for MSL Plug ejection during MSIV LLRT at P_a has been evaluated and found to be bounded by existing accident analysis. Therefore the proposed change to allow an alternative test pressure, P_a, 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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701

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

NRC Project Director: John F. Stolz

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 amendment request: July 12, 1996

Description of amendment request: The proposed amendment would revise the Indian Point 3 (IP3) Technical Specifications (TSs) by changing the surveillance frequency requirements in Table 4.1-1, ''Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels'' to accommodate a 24-month operating cycle.

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

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

Response:

The proposed changes do not involve a significant increase in the probability or consequence of any accident previously evaluated. The proposed changes are being made to extend surveillance frequencies from 18 months to 24 months for:

Vapor Containment High Radiation Monitors

Reactor Coolant System Subcooling Margin Monitor (SMM),

Overpressure Protection System (OPS), and

Reactor Vessel Level Indication System (RVLIS).

These proposed changes are being made using the guidance provided by Generic Letter 91-04 to accommodate a 24-month fuel cycle. The containment radiation monitors, SMM, and RVLIS are used to provide operator information during post-accident conditions and have no effect on event initiators associated with previously analyzed accidents. The OPS is used only when the plant is shutdown, with RCS [reactor coolant system] temperature below a low temperature limit, and the RCS is not vented. The function of the OPS is to protect the RCS from Low Temperature Overpressurization (LTOP) transients and has no effect on accident initiators. No credit is taken in the IP3 safety analyses for accident mitigation effects that might result from use of these instrument channels. Updated calculations and evaluations to assess the proposed increase in the surveillance intervals demonstrate that the effectiveness of these instrument channels in fulfilling their respective functions is not reduced. The containment high radiation monitors are used for post accident monitoring purposes to provide operators with an indication of adverse conditions in containment based on releases of radioactivity from the RCS to the containment atmosphere. These monitors provide no signals to plant control systems or automatic safety systems used for accident mitigation and have no role as an accident initiator.

Use of the subcooling margin monitor and core exit thermocouples by plant operators is specified in the Indian Point 3 Emergency Operating Procedures (EOPs) to assess post accident cooling conditions in the RCS. Changes to the EOPs will be made to reflect the results of the updated loop accuracy calculations for this instrumentation. These changes will ensure that safety analysis input assumptions associated with subcooling margin, for small break LOCA [loss-of-coolant accident], steam generator tube rupture, and steamline break, remain valid, and that the response strategies outlined in the Westinghouse Owners Group Emergency Response Guidelines are maintained. Core exit thermocouple readings are not used for input to plant safety analyses.

The OPS provides a protective function to prevent RCS pressure limits from being exceeded while the plant is shutdown and the RCS is being maintained at a low temperature and not vented. Failure of the OPS is not assumed to be an accident initiator in the plant safety analyses.

The change to the RVLIS calibration interval does not affect design or operation of plant systems and will not affect the probability of accidents. Revised loop accuracy calculations have demonstrated that operator actions for responding to postulated accidents using RVLIS in conjunction with the Indian Point 3 EOPs will remain consistent with the accuracy requirements RVLIS. The consequences of a previously evaluated accident will not be affected.

Equipment and system design requirements and safety analysis acceptance criteria continue to be met with the proposed new surveillance intervals. Based on the above information it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

Response:

The proposed changes to extend the surveillance frequencies for the above listed instrument channel do not create the possibility of a new or different kind of accident from any previously evaluated. The increased surveillance frequencies were evaluated based on past equipment performance and do not require any plant hardware changes or changes in system operation. There are no new failure modes introduced as a result of extending these surveillance intervals, which could lead to the creation of new or different kinds of accident.

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

Response:

The proposed changes do not involve a significant reduction in a margin of safety. [A decreased] surveillance frequency for the Containment High Radiation Monitor, SMM, OPS, and RVLIS does not adversely affect the performance of safety-related systems, equipment, or instruments and does not result in increased severity of accidents evaluated. The radiation monitor, SMM, and RVLIS are not used to support margins of safety identified in the Technical Specifications. OPS provides an equipment protection function to prevent inadvertent overpressurization of the RCS at shutdown conditions. The Low Temperature Overpressurization (LTOP) curve in the Technical Specifications represents material stress limits based on fracture toughness requirements for ferritic steel. Analysis of the proposed change to the OPS surveillance frequency verified sufficient margin to the LTOP curve and therefore does not involve a significant reduction in margin to the material stress limits.

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: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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 amendment request: July 12, 1996

Description of amendment request: The proposed amendment would change the Indian Point 3 (IP3) Technical Specifications (TS) relating to minimum reactor coolant system (RCS) flow and maximum RCS average temperature to make these parameters consistent with an assumption of 100% helium release from the boron coating of the integral fuel burnable absorber (IFBA) rods.

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

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

The proposed changes to the RCS minimum flow and maximum T^{avg} requirements will not increase the probability or consequences of an accident previously evaluated. Reference 2 [SECL-96-046, ''IFBA Helium Release Evaluation for Cycle 9 Restart,'' Westinghouse Electric Corporation, dated July 8, 1996] states that, for the remainder of Cycle 9, all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the licensing basis accident analyses for the assumption of a 100% helium release from the IFBA rods. Reference 3 [Westinghouse letter, ''Technical Specification Value for T-Average,'' INT-96-557, dated July 3, 1996] states that a reduction of maximum allowable indicated T_{avg} from 578.3°F to 571.5°F specifications consistent with the more limiting containment integrity analyses. The associated plant and technical specification changes do not affect any of the mechanisms postulated in the FSAR [Final Safety Analysis Report] to cause licensing basis events. Therefore, the probability of an accident previously evaluated has not increased. Because design limitations continue to be met, and the integrity of the RCS pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Therefore, the consequences of an accident previously evaluated will not be increased.

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

The proposed changes to the RCS minimum flow and maximum T_{avg} requirements do not create the possibility of a new or different kind of accident from any previously evaluated. Reference 2 states that, for the remainder of Cycle 9, all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the licensing basis accident analyses for the assumption of a 100% helium release from the IFBA. Reference 3 provides clarifications of the assumptions made in the design basis and restricts DNB temperature limits to be consistent with non-DNB analyses. The associated plant and technical specification changes do not change the plant configuration in a way which introduces a new potential hazard to the plant (i.e., no new failure mode has been created). Therefore, an accident which is different than any previously evaluated will not be created.

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

The proposed changes to the RCS minimum flow and maximum T_{avg} requirements do not involve a significant reduction in a margin of safety. Reference 2 demonstrates that, for the remainder of Cycle 9, all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the Technical Specification Bases is not reduced in any of the licensing basis accident analyses for the assumption of a 100% helium release from the IFBA. Reference 3 maintains the margin of safety by restricting a DNB limit to bound other analyses. Since References 2 and 3 demonstrate that all applicable acceptance criteria continue to be met, the subject operating conditions 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: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: Jocelyn A. Mitchell, Acting

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 amendment request: May 3, 1996 (TS 352)

Description of amendment request: The proposed amendment requests administrative changes to the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 technical specifications. The proposed amendment consists of three parts, designated by the licensee as A, B, and C. Part A deletes technical specification requirements associated with BFN Unit 2 Amendment 219, issued November 12, 1993, to permit modification of reactor vessel water level instrumentation requested by NRC Bulletin 93-03. Part B deletes technical specification requirements associated with Amendment 228, issued on December 7, 1994, which provided a temporary change to permit upgrade of electrical equipment. The modifications associated with Parts A and C are complete. Part C provides other administrative changes to clarify requirements and to implement rule changes.

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

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

Part A: The proposed Technical Specification change to remove the temporary revisions, which were in place to modify the reactor vessel water level instrumentation requested by NRC Bulletin 93-03, is administrative. The temporary limiting condition for the minimum number of trip systems operable will no longer be accurate and the minimum number operable per trip system will be the same as they were prior to November 12, 1993. Therefore, the proposed changes will not significantly increase the consequences of an accident previously evaluated.

Part B: The proposed Technical Specification change to remove the temporary revisions, which were in place to replace the 250 volt shutdown board batteries is administrative. The LCO to extend the allowed outage time (AOT) from a five-day to a 45-day AOT will no longer be accurate and the five day AOT will be the same as it was prior to Unit 2, Cycle 7. Therefore, the proposed changes will not significantly increase the consequences of an accident previously evaluated.

Part C: The proposed Technical Specifications change revises items 1 through 5 above (Section I, Description of the Proposed Change, Part C), and is administrative. TVA has evaluated the proposed technical specification changes and has determined that the proposed changes are administrative in nature. Further, it provides a revision based on an NRC Code of Federal Regulations rule change. Also, the proposed changes provide correction of administrative errors from previous technical specifications. For example, the Main Steamline High Radiation remarks in Table 3.2.A, 1.b., should have been deleted from the TS as part of TS-322. It also clarifies some requirements to ensure consistent application throughout the specifications. These changes do not affect any of the design basis accidents. They do not involve an increase in the probability or consequences of an accident previously evaluated.

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

Part A: The proposed Technical Specification change to remove the temporary revisions, which were in place to modify the reactor vessel water level instrumentation requested by NRC Bulletin 93-03, is administrative. The temporary limiting condition for the minimum number of trip systems operable will no longer be accurate and the minimum number operable per trip system will be the same as they were prior to November 12, 1993. No modifications to any plant equipment are involved. There are no effects on system interactions made by these changes. They do not create the possibility of a new or different kind of accident from an accident previously evaluated.

Part B: The proposed Technical Specification change to remove the temporary revisions, which were in place to replace the 250 volt shutdown board batteries is administrative. The LCO to extend the allowed outage time (AOT) from a five day to a 45-day AOT will no longer be accurate and the five day AOT will be the same as it was prior to Unit 2, Cycle 7. No modifications to any plant equipment are involved. There are no effects on system interactions made by these changes. They do not create the possibility of a new or different kind of accident from an accident previously evaluated.

Part C: The proposed Technical Specifications change revises items 1 through 5 above (Section I, Description of the Proposed Change, Part C), and is administrative. TVA has evaluated the proposed changes and has determined that they are administrative in nature. Further, it provides revisions based on an NRC Code of Federal Regulations rule change. It also provides correction of administrative errors in previous technical specification changes. For example, the Main Steamline High Radiation remarks in Table 3.2.A, 1.b., should have been deleted from the TS as part of TS-322. It also clarifies some requirements to ensure consistent application throughout the specifications. These changes do not affect any of the design basis accidents. No modifications to any plant equipment are involved. There are no effects on system interactions made by these changes. They do not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change is administrative in nature for Parts A, B, and C. The proposed change includes the deletion of temporary changes as a result of modifications to systems and clarification of some requirements to ensure consistent application throughout the specifications. Further, the proposed change corrects errors in previous TS submittals. No safety margins are affected by these changes.

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: Athens Public Library, South Street,Athens, Alabama 35611

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Local Public Document Room location: Athens Public Library, South Street,Athens, Alabama 35611

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

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 amendment request: June 21, 1996 (TS 377)

Description of amendment request: The proposed amendment provides a new minimum critical power ratio safety limit to replace the current non-conservative value. The amendment also updates the technical specification bases to clarify the usage of the residual heat removal supplemental spent fuel pool cooling mode.

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

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

The proposed change in the Safety Limit Minimum Critical Power Ratio (SLMCPR) does not increase the frequency of the precursors to design basis events or operational transients analyzed in the Browns Ferry Final Safety Analysis Report. Therefore, the probability of an accident previously evaluated is not significantly increased.

The proposed change in the SLMCPR ensures that 99.9 percent of the fuel rods in the core are expected to avoid boiling transition during the most limiting anticipated operational occurrence, which is the design and licensing basis for the analysis of accidents and transients described in the Browns Ferry Updated Final Safety Analysis Report (UFSAR). It does not change the nuclear safety characteristics of any safety system or containment system. Therefore, the consequences of an accident, operator error, or malfunction of equipment important to safety previously evaluated in the UFSAR has not been increased.

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

The proposed change to the Technical Specification requirements for the safety limit minimum critical power ratio does not involve a modification to plant equipment. No new failure modes are introduced. There is no effect on the function of any plant system and no new system interactions are introduced by this change. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will ensure that during any anticipated operational transient, at least 99.9% of the fuel rods would be expected to avoid boiling transition which is consistent with the licensing basis. Since the margin [of] safety is being increased with this change, the proposed amendment does not involve a reduction in a margin of safety.

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

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611

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

NRC Project Director: Frederick J. Hebdon

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

Date of application request: July 18, 1996

Description of amendment request: The amendment adopts ASTM D-3803-1989 as the laboratory testing standard for charcoal samples from the charcoal adsorbers in the auxiliary/fuel building emergency exhaust system.

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

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

The requested change to the charcoal sample surveillance acceptance criteria for the fuel building and auxiliary building emergency exhaust system will not affect the method of operation of the system. The testing of the charcoal filter samples will continue to be performed in accordance with NRC-accepted methods and acceptance criteria, and the new test protocol will still ensure filter efficiency is maintained equal to or greater than 90%. There are no changes to the emergency exhaust system and it will continue to function in a manner consistent with the safety analysis assumptions and the plant design basis. There will be no degradation in the performance of or an increase in the number of challenges to equipment assumed to function during an accident. Therefore, the proposed changes will not increase 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 changes to the surveillance requirements are being made to adopt current NRC-accepted methods of testing charcoal samples. These changes will not affect the method of operation of the applicable systems and the laboratory testing will continue to demonstrate the required adsorber performance after a design-basis LOCA [loss-of-coolant accident] or fuel handling accident. No new or different kind of accident from any previously evaluated will be created.

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

The new charcoal adsorber sample laboratory testing protocol is more stringent than the current testing practice and meets current NRC-approved test methods. The new testing criteria will continue to demonstrate the required adsorber performance after a design-basis LOCA or fuel handling accident and will not affect the filter system performance. Therefore, this change will not reduce the margin of safety of the emergency exhaust system filter operation.

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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

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

NRC Project Director: William H. Bateman

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

Date of amendment request: July 18, 1996

Description of amendment request: The proposed amendment would revise

Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 3.8, ''Refueling Operations,'' and its associated Basis, by allowing the containment personnel air lock doors to remain open during refueling operations as long as at least one door is capable of being closed in 30 minutes or less.

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

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to determine that no significant hazards exist. The proposed changes will not:

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

Maintaining the doors of the personnel air lock open during REFUELING OPERATIONS does not adversely affect the probability or consequences of accidents previously evaluated. The only applicable accident is a fuel handling accident described in [Updated Safety Analysis Report] USAR Section 14.2.1. The fuel handling accident evaluated in the USAR Section 14.2.1 assumes the accident to be in the spent fuel pool in the Auxiliary Building. The accident assumes a sudden release of the gaseous fission products held in the voids between the pellets and cladding of all of the rods in the highest rated fuel assembly at 100 hours following reactor shutdown. The accident activity is assumed to discharge from the spent fuel pool directly to the atmosphere at ground level. No credit is taken for existing building structures, ventilation, or filtration systems. A fuel handling accident in containment is bounded by this evaluation. Furthermore, any release from a fuel handling accident in containment can still be terminated by closing one of the personnel air lock doors following containment evacuation.

The containment personnel air lock doors are components integral to the containment structure. They are not accident initiators. Therefore, the proposed amendment does not increase the probability of any previously evaluated accident.

The control room operator immersion and inhalation doses were reviewed as part of the updated Control Habitability Evaluation Report. The report states that thyroid and whole body doses received by control room operators in each of the other design basis accidents discussed in KNPP USAR Section 14.2 are less than the [loss of coolant accident] LOCA dose. This amendment does not change the results of the Control Room Habitability Evaluation Report, since the fuel handling accident evaluated in KNPP USAR Section 14.2.1 assumes a release directly to the atmosphere. This change does not significantly increase the consequences of an accident previously evaluated.

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

The accident evaluated in USAR section 14.2.1 bounds a fuel handling accident in containment with the personnel air lock doors open. The fuel handling accident evaluated in USAR section 14.2.1 assumes activity is discharged directly to the atmosphere at ground level. Since no credit is taken for building structures, ventilation systems or filtration systems, the position of the doors does not affect the analysis of record. Furthermore, one of the air lock doors can still be closed following containment evacuation to terminate the release.

The containment personnel air lock doors are components integral to the containment structure. They are not accident initiators. The proposed amendment does not create the possibility of any new or different kind of accident [from any accident] previously evaluated.

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

Maintaining the containment personnel air lock doors open during REFUELING OPERATIONS does not involve a significant reduction in the margin of safety. A fuel handling accident in containment is bounded by a fuel handling accident in the spent fuel pool. The spent fuel pool fuel handling accident is assumed to have a sudden release of the gaseous fission products held in the voids between the pellets and cladding of all of the rods in the highest rated fuel assembly, 100 hours following reactor shutdown. The accident activity leaving the spent fuel pool is assumed to discharge directly to the atmosphere at ground level. No credit is taken for existing building structures, ventilation, and filtration systems. Therefore, there is no reduction in the current margin of safety. Furthermore, the release caused by a fuel handling accident in containment can be terminated by closing one of the personnel air lock doors following containment evacuation.

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

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497

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.

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: July 12, 1996

Brief description of amendment request: The amendment would change Technical Specification 3.3.2.1, ''Engineered Safety Feature Actuation System Instrumentation,'' to reflect a revised setpoint for the interlock designated P-12.

Date of publication of individual notice in Federal Register: July 23, 1996 (61 FR 38229)

Expiration date of individual notice: August 22, 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.

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

Date of application for amendment: January 29, 1996, as supplemented June 17, 1996.

Brief description of amendment: The amendment revises the technical specifications (TS) table 4.1-3, item 4 to change the frequency of main steam safety valve (MSSV) testing to that specified in NUREG-1431, the improved ''Standard Technical Specifications, Westinghouse Plants'' and adds the MSSV test acceptance requirements.

Date of issuance: August 1, 1996

Effective date: August 1, 1996

Amendment No.: 171

Facility Operating License No. DPR-23. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7545). The June 17, 1996, submittal provided supplemental information that was not outside the scope of the February 28, 1996, notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550

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

Date of application for amendment: March 20, 1996

Brief description of amendment: To relocate Technical Specification 3.3.3.2, Movable Incore Detectors, to plant procedures.

Date of issuance: July 24, 1996

Effective date: July 24, 1996

Amendment No.: 65

Facility Operating License No. NPF-63. Amendment revises the Technical Specifications

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18164) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County and Northeast Nuclear Energy Company, et al., Docket Nos. 50-245, 50-336, and 50-423, Millstone Nuclear Power Station, Units 1, 2, and 3, New London County, Connecticut

Date of application for amendments: November 22, 1995

Brief description of amendments: The amendments replace the title-specific designation of members representing specific functional areas on the Plant Operating Review Committee (PORC) for the Haddam Neck Plant and Millstone Units 1, 2, and 3 with a functional area-specific designation that stipulates membership qualification and experience requirements. The amendments also clarify the composition of the Site Operations Review Committee (SORC) at Millstone.

Date of issuance: July 16, 1996

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

Amendment Nos.: 190, 95, 200, 130

Facility Operating License Nos. DPR-61, DPR-21, DPR-65, AND NPF-49: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7549) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 16, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Russell Library, 123 Broad Street Middletown, Connecticut 06457, for the Haddam Neck Plant, and the Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385, for Millstone 1, 2, and 3.

Duke Power Company, et al., Docket No. 50-413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of application for amendment: January 26, 1996, as supplemented May 6, May 20, and June 5, 1996

Brief description of amendment: The amendment revises the Technical Specifications to permit a one-time operation of the containment purge ventilation system during Mode 3 and 4 after the steam generator replacement outage.

Date of issuance: July 30, 1996

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

Amendment No.: 150

Facility Operating License No. NPF-35: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18165) The supplemental submittals provided clarifying information that did not change the scope of the January 26, 1996, application for amendment nor the initial proposed no significant hazards consideration determination.

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

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: March 4, 1996

Brief description of amendments: The amendments delete Flow Monitoring System from Technical Specification 3.4.6.1 and associated surveillance requirements.

Date of issuance: July 29, 1996

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

Amendment Nos.: 168 and 150

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18166) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 29, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: March 4, 1996

Brief description of amendments: The amendments consist of changes to the Final Safety Analysis Report for McGuire Units 1 and 2 to delete the seismic qualification requirement for the Containment Atmosphere Particulate Radiation Monitors.

Date of issuance: July 30, 1996

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

Amendment Nos.: 169 and 151

Facility Operating License Nos. NPF-9 and NPF-17: Amendments revised the Final Safety Analysis Report.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20845) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 30, 1996, and an Environmental Assessment dated July 22, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: May 20, 1996

Brief description of amendment: The amendment revised the Facility Operating License and Appendix C to the license to reflect the name change from Gulf States Utilities Company to Entergy Gulf States, Inc.

Date of issuance: July 30, 1996

Effective date: July 30, 1996

Amendment No.: 88

Facility Operating License No. NPF-47: The amendment revised the operating license and Appendix C to the license.

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31183) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 1996. No significant hazards consideration comments received. No

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: November 20, 1995, as supplemented by letter dated December 15, 1995

Brief description of amendment: The amendment revised and deleted surveillance requirements, notes, and action statements involved with the requirements for the drywell leak rate testing, and the air lock leakage and interlock testing in Subsections 3.6.5.1 (Drywell), 3.6.5.2 (Drywell Air Lock), and 3.6.5.3 (Drywell Isolation Valves) of the technical specifications.

Date of issuance: August 1, 1996

Effective date: August 1, 1996

Amendment No: 126

Facility Operating License No. NPF-29: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25704) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1996. No significant hazards consideration comments received: No.

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

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: March 21, 1996 as supplemented May 13, 1996.

Brief description of amendments: Relocate requirements for Radiological Effluent Controls from Technical Specifications (TS) to the Offsite Dose Calculation Manual or the Process Control Program. New programmatic controls for radioactive effluent and radiological environmental controls will be incorporated into the TS. Also, requirements for Gas Decay tanks and Explosive Gas Mixture will be placed in a different area of the TS.

Date of issuance: July 31, 1996

Effective date: July 31, 1996

Amendment Nos.: 188 and 182Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 19, 1966 (61 FR 31180) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 31, 1996. No significant hazards consideration comments received: No

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

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: May 28, 1996

Brief description of amendments: Amendment changes Technical Specification 6.2.2.i, ''Administrative Controls,'' regarding Operations Manager qualifications.

Date of issuance: July 22, 1996

Effective date: July 22, 1996

Amendment Nos.: 187 and 181Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31181) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 22, 1996. No significant hazards consideration comments received: No

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

GPU Nuclear Corporation and Saxton Nuclear Experimental (SNEC) Corporation, Docket No. 50-146, Saxton Nuclear Experimental Facility (SNEF)

Date of application for amendment: February 2, 1996, as supplemented on February 28, April 24, and May 24, 1996.

Brief description of amendment: The proposed amendment would (1) increase the scope of work permitted at SNEF to include asbestos removal, removal of defunct plant electrical services, and installation of decommissioning support facilities and systems; (2) eliminate areas within the containment vessel requiring administrative access controls; and (3) revise the facility layout diagram to allow the exclusion area to consist of, at a minimum, the containment vessel and, at a maximum, to extend to the SNEF outer security fence and to include on the diagram the footprint of the proposed decommissioning support facilities.

Date of issuance: July 23, 1996

Effective date: July 23, 1996

Amendment No.: 14

Amended Facility License No. DPR-4: Amendment changed the Technical Specifications.

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31182).

The Commission's related evaluation of the amendment is contained in a safety evaluation dated July 23, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Saxton Community Library, 911 Church Street, Saxton, Pennsylvania 16678

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

Date of amendment request: February 1, 1996

Brief description of amendment: The amendment revised Technical Specifications to allow an increase in the initial nominal Uranium-235 enrichment limit for fuel assemblies which may be stored in the spent fuel pool.

Date of issuance: July 30, 1996

Effective date: July 30, 1996

Amendment No.: 174

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

Date of initial notice in Federal Register: March 13, 1996 (61 FR 10396) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 1996 . No significant hazards consideration comments received: No.

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

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 application for amendments: May 9, 1996

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant (DCPP), Unit Nos. 1 and 2 by revising Technical Specifications (TS) 3/4.3.2, ''Engineered Safety Features Actuation System Instrumentation,'' and 3/4.6.2, ''Containment Spray System.'' The changes clarified the description of the initiation signal required for operation of the containment spray system at DCPP and correctly incorporated changes made in previous license amendments. All of the changes are administrative in nature.

Date of issuance: August 1, 1996

Effective date: August 1, 1996

Amendment Nos.: Unit 1 - 114; Unit 2 - 112

Facility Operating License Nos. DPR-80 and DPR-82: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31184) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 1, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 3, 1996, as superseded by application dated June 25, 1996.

Brief description of amendments: These amendments revise Improved Technical Specification (TS) 3.3.11, ''Post Accident Monitoring Instrumentation (PAMI),'' and Improved TS 5.5.2.13, ''Diesel Fuel Oil Testing Program.'' Specifically, the number of instruments required to measure reactor coolant inlet temperature (T_{Cold}), and reactor coolant outlet temperature (T_{Hot}), will be revised from two per loop to two (with one cold leg indication and one hot leg indication per steam generator). These changes to the Improved TS reinstate provisions of the current San Onofre Nuclear Generating Station (SONGS), Unit Nos. 2 and 3 TS revised as part of NRC Amendment Nos. 127 and 116 for SONGS Units 2 and 3 (referred to as the Improved TS).

Date of issuance: August 1, 1996

Effective date: August 1, 1996, to be implemented by August 9, 1996.

Amendment Nos.: Unit 2 - 130; Unit 3 - 119

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

Date of initial notice in Federal Register: July 2, 1996 (61 FR 34452) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 1, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 26, 1995, as supplemented April 25, 1996. The April 25, 1996, letter provided clarifying information that did not change the scope of the July 26, 1995, application and initial proposed no significant hazards consideration determination.

Brief description of amendments: The amendments clarify the Technical Specifications to allow switching of charging and low-head safety injection pumps during unit shutdown conditions. These amendments also allow additional methods of rendering these same pumps incapable of injecting into the reactor coolant system when required for low-temperature conditions.

Date of issuance: July 24, 1996

Effective date: July 24, 1996

Amendment Nos.: 202 and 183

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

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45190) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 24, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498.

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

Date of application for amendment: May 8, 1996

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant Technical Specification (TS) 5.3, ''Reactor,'' and TS 5.4, ''Fuel Storage,'' by removing the enrichment limit for reload fuel and imposing fuel storage restrictions on the spent fuel storage racks and the new fuel storage racks. The revised TS are structured consistent with the Westinghouse Standard Technical Specifications and the fuel storage restrictions are based on the criticality analyses used to support Amendment No. 92 dated March 7, 1991.

Date of issuance: July 23, 1996

Effective date: July 23, 1996

Amendment No.: 124

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

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31185) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 23, 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

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

Date of amendment request: May 1, 1995

Brief description of amendment: This amendment revises TS Section 6.0, throughout, to reflect an organization change in which the position of Vice President Plant Operations has been eliminated and the positions of Chief Operating Officer and Plant Manager were created. This change assigns certain management responsibilities to the Chief Operating Officer and Plant Manager.

Date of issuance: August 1, 1996

Effective date: August 1, 1996, to be implemented within 30 days of issuance.

Amendment No.: 100

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25716) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1996. No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 7th day of August 1966.

For the Nuclear Regulatory Commission

Steven A. Varga, Director,

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

[Doc. 96–20586 Filed 8-13-96; 8:45 am]

BILLING CODE 7590-01-F

RAILROAD RETIREMENT BOARD

Sunshine Act Meeting

Notice is hereby given that the Railroad Retirement Board will hold a meeting on August 21, 1996, 9:00 a.m., at the Board's meeting room on the 8th floor of its headquarters building, 844 North Rush Street, Chicago, Illinois, 60611. The agenda for this meeting follows:

Portion Open to the Public

(1) Legislative Proposals 105–4 (Greater Access to Tax Return Information) and 105–14 (Conform the Statute of Limitations on the Crediting of Compensation to the Statute of Limitations on the Payment of taxes).

(2) Regulations:

A. Part 211, Pay for Time Lost.

B. Parts 211, 230 and 255 (Proposed Cost Savings Analyses).

(3) Coverage Determination—CSX Transportation Company—Nurse Consultants.

(4) CSX Intermodal, Inc.

(5) Proposed Draft Agreement with the Social Security Administration.

(6) Medicare Part B Service Contract.

(7) Press Release No. 96–8—Direct Deposit Required for New RRB Claims.

(8) Policy for Determining Competitive Areas for a Reduction-in-Force (RIF).

(9) Labor Member Truth in Budgeting Status Report.

Portion Closed to the Public

(A) *Pending Board Appeals*

1. Walter Coleman

2. Grace P. Sansom

The person to contact for more information is Beatrice Ezerski, Secretary to the Board, Phone No. 312–751–4920.

Dated: August 9, 1996.

Beatrice Ezerski,

Secretary to the Board.

[FR Doc. 96–20818 Filed 8–12–96; 9:38 am]

BILLING CODE 7905–01–M

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–22127; No. 812–10204]

American Skandia Life Assurance Corporation, et al.

August 8, 1996.

AGENCY: Securities and Exchange Commission (''SEC'' or ''Commission'').

ACTION: Notice of Application for an Exemption from the Investment Company Act of 1940 (''1940 Act'').

APPLICANTS: American Skandia Life Assurance Corporation (''American Skandia''), American Skandia Assurance Corporation Variable Account B (Class 2 Sub-Accounts) (''Separate Account'') and American Skandia Marketing, Inc. (''Marketing'').

RELEVANT 1940 ACT SECTIONS: Order requested under Section 6(c) of the 1940 Act granting exemptions from the provisions of Sections 26(a)(2)(C) and 27(c)(2) of the 1960 Act.

SUMMARY OF APPLICATION: Applicants seek an order to permit the deduction of a mortality and expense risk charge from the assets of the Separate Account or any other separate account (''Other Account'') established by American Skandia to support certain flexible premium variable annuity contracts (''Contracts'') as well as other variable annuity contracts issued by American Skandia that are substantially similar in all material respects to the Contracts (''Future Contracts''). In addition, Applicants request that the exemptions requested herein apply to any other broker-dealer that may in the future serve as distributor of and/or principal underwriter for Contracts or Future Contracts (''Future Broker-Dealers''). Any Future Broker-Dealer will be a member of the National Association of Securities Dealers, Inc. (''NASD''), and will be controlling, controlled by, or under common control with American Skandia.

FILING DATE: The application was filed on June 17, 1996.

HEARING OR NOTIFICATION OF HEARING: An order granting the Application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Secretary of the SEC and serving Applicants with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on September 3, 1996, and should be accompanied by proof of service on Applicants in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the requestor's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the Secretary of the SEC.

ADDRESSES: Secretary, Securities and Exchange Commission, 450 5th Street, N.W., Washington, D.C. 20549. Applicants, M. Patricia Paez, Corporate Secretary, c/o Jeffrey M. Ulness, Esq., American Skandia Life Assurance Corporation, One Corporate Drive, Shelton, Connecticut 06484–9932.

FOR FURTHER INFORMATION CONTACT: Peter R. Marcin, Law Clerk, or Patrice M. Pitts, Special Counsel, Office of Insurance Products (Division of Investment Management), at (202) 942–0670.

SUPPLEMENTARY INFORMATION: Following is a summary of the application; the complete application is available for a fee from the Public Reference Branch of the SEC.

Applicants' Representations

1. American Skandia, a stock life insurance company, is organized in Connecticut and licensed to do business in the District of Columbia and all of the