license with respect to such 12.58-percent ownership interest and that the transfer, subject to the conditions set forth herein, is otherwise consistent with applicable provisions of law, regulations, and orders issued by the Commission.

III

By August 30, 1996, any person adversely affected by this order may file a request for a hearing with respect to issuance of the order. Any person requesting a hearing shall set forth with particularity how such person's interest is adversely affected by this order and shall address the criteria set forth in 10 CFR 2.714(d).

If a hearing is to be held, the Commission will issue an order designating the time and place of such hearing.

If a hearing is held concerning this order, the issue to be considered at any such hearing will be whether this order should be sustained.

Any request for a hearing 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. Copies should also be sent to the Office of the General Counsel and to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to Gerald Charnoff, Esquire, of Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037.

IV

Accordingly, pursuant to Sections 161b, 161i, and 184 of the Atomic Energy Act of 1954, as amended, 42 U.S.C. §§ 2201(b), 2201(i), and 2234; and 10 CFR 50.80, It is hereby ordered that the Commission consents to the proposed transfer of the license described herein from Ohio Edison to OES, subject to the following: Should the transfer not be completed by September 30, 1996, this order will become null and void, unless upon application and for good cause shown this date is extended.

This order is effective upon issuance. For further details with respect to this action, see the application for transfer dated December 28, 1995, under cover of letter dated December 29, 1995, 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 Perry Public Library, 3753 Main Street, Perry, Ohio.

For the Nuclear Regulatory Commission.

Dated at Rockville, Maryland this 25th day of July 1996.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 96–19436 Filed 7–30–96; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 6, 1996, through July 19, 1996. The last biweekly notice was published on July 17, 1996 (61 FR 37295).

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 August 30, 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.

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

Date of amendment request: March 25, 1996 (NRC-96-0003)

Description of amendment request: The proposed amendment would modify the charcoal testing standards for the Control Room Emergency Filtration System (CREFS) and the Standby Gas Treatment System (SGTS) to the current industry standard. The changes affect Surveillance Requirements (SRs) 4.6.5.3.b.2, 4.6.5.3.c, 4.7.2.1.c.2, and 4.7.2.1.d in Technical Specifications (TS) 3/4.6.5.3 ''Standby Gas Treatment System'' and TS 3/4.7.2 ''Control Room Emergency Filtration 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 TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. By providing an improved protocol for charcoal testing the proposal provides greater assurance that the installed charcoal can perform its design function and, thus, the consequences of evaluated accidents remain valid. The method of laboratory analysis has no effect upon how the plant is operated, including the method of sample removal. Therefore, the probability [or consequences] of any evaluated accident is unchanged.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposal has no effect on the manner of plant operation. The proposal does not involve any change to the plant design. Therefore, the change creates no new accident modes.

3. The proposed TS changes do not involve a significant reduction in a margin of safety. By providing an improved protocol for charcoal testing the proposal acts to maintain existing safety margins. The change to the SGTS charcoal acceptance criteria also acts to ensure that the existing margins, as discussed in Regulatory Guide 1.52, Revision 2 [Design, Testing and Maintenance Criteria for Post-Accident Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants], are maintained.

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, Acting

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: June 18, 1996

Description of amendment request: For Beaver Valley Power Station, Unit No. 1 (BVPS-1) only, the proposed amendment would revise Technical Specification (TS) 3.4.5 and associated Bases; the Bases for TS 3.4.6.2 would also be revised. The proposed changes are editorial in nature and are intended to provide consistency between the TSs and associated Bases. Index page XIX would be revised to reflect the revision of page numbers for TS Tables 4.4-1 and 4.4-2 due to shifting of text.

For Beaver Valley Power Station, Unit No. 2 (BVPS-2) only, the proposed amendment would implement a voltage-based repair criteria for steam generator tubes similar to the changes approved for BVPS-1 by License Amendment No. 198. The proposed changes are intended to reflect the guidance provided in NRC Generic Letter 95-05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.'' The proposed changes would revise TSs 3.4.5 and 3.4.6.2 and associated Bases. TS Table 4.4-2 would be revised to reference TS 6.6 for reporting requirements.

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

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

Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate (TSP). Test data indicates that tube burst cannot occur within the TSP, even for tubes which have 100% throughwall electric discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Since tube-to-TSP proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition, main steamline break (MSLB) pressure differential. The Regulatory Guide (RG) 1.121 criterion requiring maintenance of a safety factor of 1.43 times the MSLB pressure differential on tube burst is satisfied by 7/8'' diameter tubing with bobbin coil indications with signal amplitudes less than 8.6 volts, regardless of the indicated depth measurement.

The upper voltage repair limit (V_{URL}) will be determined prior to each outage using the most recently approved NRC database to determine the tube structural limit (V_{SL}). The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{NDE}) and growth (V_{GR}) to establish V_{URL}. Using the Generic Letter (GL) 95-05 NDE and growth allowances for an example, the NDE uncertainty component of 20% and a voltage growth allowance of 30% per full power year can be utilized to establish a V_{URL} of 5.7 volts. The 20% NDE uncertainty represents a square-root-sum-of-the-squares (SRSS) combination of probe wear uncertainty and analyst variability. The degradation growth allowance should be an average growth rate or 30% per effective full power year, whichever is larger.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated MSLB outside of containment but upstream of the main steam isolation valve (MSIV) represents the most limiting radiological condition relative to the plugging criteria. In support of implementation of the revised plugging limit, analyses will be performed to determine whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary-to-secondary leakage would result in postulated site boundary and control room doses exceeding 10 CFR 100, 10 CFR 50 Appendix A, and GDC-19 [General Design Criterion-19] requirements, respectively. A separate calculation has determined the maximum allowable MSLB leakage limit in a faulted loop. This limit was calculated using the technical specification reactor coolant system (RCS) Iodine-131 activity level of 1.0 microcuries per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800. The projected MSLB leakage rate calculation methodology prescribed in Section 2.b of GL 95-05 will be used to calculate the end-of-cycle (EOC) leakage. Projected EOC voltage distribution will be developed using the most recent EOC eddy current results and considering an appropriate voltage measurement uncertainty. The log-logistic probability of leakage correlation will be used to establish the MSLB leakrate used for comparison with the faulted loop allowable limit. Therefore, as implementation of the voltage-based repair criteria does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

The proposed changes to the BVPS-1 Index, Specifications and associated Bases and the proposed change to BVPS-2 Table 4.4-2 are editorial in nature. Therefore, these changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

Implementation of the proposed steam generator tube voltage-based repair criteria does not introduce any significant changes to the plant design basis. Use of the voltage-based repair criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations as no outside diameter stress corrosion cracking (ODSCC) is occurring outside the thickness of the tube support plates. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging limit has been applied (during all plant conditions).

Duquesne Light Company will implement a maximum primary-to-secondary leakage rate limit of 150 gpd [gallons per day] per steam generator to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit provides for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. RG 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded.

The single through-wall crack lengths that result in tube burst at 1.43 times the MSLB pressure differential and the MSLB pressure differential alone are approximately 0.57 inch and approximately 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of approximately 0.41 inch long cracks at nominal leak rates and approximately 0.62 inch long cracks at the lower 95% confidence level leak rates. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during MSLB conditions, the leakage from the maximum permissible crack must preclude tube burst at MSLB conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for MSLB conditions using the lower 95% leakrate data. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection. Analyses have shown that only a small percentage of the TSPs are deflected greater than the TSP thickness during a postulated MSLB.

As steam generator tube integrity upon implementation of the voltage-based repair criteria continues to be maintained through inservice inspection and primary-to-secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

The proposed change to BVPS-1 Index, Specifications and associated Bases and the proposed change to BVPS-2 Table 4.4-2 are editorial in nature. These changes do not change the performance of plant systems, plant configuration or method of operating the plant.

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

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

The use of the voltage-based repair criteria at BVPS-2 maintains steam generator tube integrity commensurate with the criteria of RG 1.121. This guide describes a method acceptable to the Commission for meeting GDCs 14, 15, 30, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be repaired or removed from service. Upon implementation of the proposed criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the tube support plate elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences remain within the licensing basis.

In addressing the combined effects of loss-of-coolant-accident (LOCA) + safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse. There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to complete through-wall cracks during tube deformation or collapse.

The results of an analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation. Since the leak-before-break methodology is applicable to the reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes the pressurizer spray line break. Analysis results have demonstrated that no tubes were subject to deformation or collapse. No tubes have been excluded from application of the subject voltage-based steam generator tube repair criteria.

Addressing RG 1.83 considerations, implementation of the voltage-based repair criteria is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, the bobbin coil inspection will include 100% of the hot-leg TSP intersections and cold-leg intersections down to the lowest cold-leg TSP with known ODSCC, the determination of the TSPs having ODSCC will be based on the performance of at least 20% random sampling of tubes inspected over their full length, and rotating pancake coil inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

As noted previously, implementation of the tube support plate intersection voltage-based repair criteria will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, implementation of the voltage-based repair criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

The proposed change to the BVPS-1 Index, Specifications and associated Bases and the proposed change to BVPS-2 Table 4.4-2 are editorial in nature. These changes will not reduce the margin of safety because they have no impact on any safety analysis assumptions.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the UFSAR or any BASES of the plant technical specifications.

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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001

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

NRC Project Director: John F. Stolz

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

Date of amendment request: May 9, 1996

Description of amendment request: The proposed amendment changes both technical and administrative requirements associated with station batteries. The proposed changes are modeled after ''Standard Technical Specifications - Babcock and Wilcox Plants,'' NUREG-1430 and Nuclear Energy Institute guidance, ''IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,'' IEEE Std 450-1995.

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

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

The switchyard 125V DC control power source requirements do not meet the criteria for inclusion in Technical Specifications (TSs) as evaluated with respect to the selection criteria of 10 CFR 50-36. These control power sources are not assumed to mitigate accident or transient events. The effects of a loss of these control power sources are enveloped by the Loss of Offsite Power (LOOP) event and relocation is considered to have a non-significant impact on the probability or severity of a LOOP event. These requirements will be relocated from the TSs to an appropriate administratively controlled document and maintained pursuant to 10 CFR 50.59.

Proposed changes incorporating the requirements of TS 3.7.1.D, 3.7.2.E, 3.7.2.F, and 3.7.2.A, as related to the DC electrical power subsystems in the new TS 3.7.3 results in a more stringent requirement for the ANO-1 TSs in that reductions to lower conditions of operation in shorter periods of time are now required. These more stringent requirements are not assumed to be initiators of any analyzed events and will not alter assumptions relative to mitigation of accident or transient events.

Proposed changes incorporating TS 3.7.4. requirements for the station batteries allowing the battery parameters to be outside the limits of the Battery Inspection Program for 31 days do not result in an increase in the frequency of consequences of any analyzed accident, as the actions require more frequent checks of other parameters to ensure battery capability during this 31 day period. The Battery Inspection Program also requires evaluations to determine battery operability in the event these limits are exceeded. If an evaluation shows the battery is incapable of performing its design basis function, that DC electrical subsystem will be declared inoperable, and the appropriate actions taken.

Proposed changes to allow the use of float current in lieu of specific gravity incorporate current industry guidance on operability measures for station batteries, as stated in IEEE-450, ''IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications.'' This Surveillance Requirement is not considered to initiate or mitigate any analyzed accident.

The proposed incorporation of a Battery Inspection Program relocates maintenance requirements from the TSs to a program under 10 CFR 50.59 control and allows the TSs to concentrate on those items required to ensure battery operability. These relocated requirements are not considered to be initiators of any analyzed accident. Battery operability is assured by the combination of TS Surveillance Requirements and Battery Inspection Program maintenance requirements based on IEEE-450 guidance.

Proposed changes in Surveillance Requirements and Frequencies reflect current industry guidance on maintenance and testing of the station batteries. These requirements, in themselves, are not considered to be initiators of any analyzed accident condition. Although some frequencies have been extended, continued performance of maintenance activities in accordance with IEEE-450, in addition to the required Surveillance Requirements, ensures that corrective maintenance can be performed prior to a condition challenging an operability limit.

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

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

The proposed changes do not change the design, configuration, or method of operation of the plant.

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.

Relocation of the switchyard 125V DC control power source requirements has no impact on any safety analysis assumptions. In addition, the requirements associated with these control power sources are relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of 10 CFR 50.59.

Proposed changes incorporating the requirements of TS 3.7.1.D, 3.7.2.E, 3.7.2.F, and 3.7.2.A, as related to the DC electrical power subsystems, in the new TS 3.7.3 impose more stringent requirements than previously specified for ANO-1.

Proposed changes incorporating TS 3.7.4 requirements for the station batteries allowing the battery parameters to be outside the limits of the Battery Inspection Program for 31 days may involve an incremental reduction in the margin of safety since the battery may be in a slightly degraded state. However, this reduction is not considered significant in that the associated actions require more frequent checks of other parameters to ensure battery capability during this 31 day period. The attery Inspection Program also requires evaluations to determine battery operability in the event these limits are exceeded.

If an evaluation shows the battery is incapable of performing its design basis function, that DC electrical subsystem will be declared inoperable, and the appropriate actions taken.

The proposed change to allow the use of float current in lieu of specific gravity as a measure of battery operability is expected to result in a more representative measure of operability. IEEE-450 states that specific gravity may not be an appropriate measure of battery capability following addition of electrolyte or when the battery is on recharge following a discharge.

Proposed incorporation of a Battery Inspection Program relocates maintenance requirements from the TSs to a program under 10 CFR 50.59 controls and allows the TSs to concentrate on those items required to ensure battery operability. The relocation of these requirements is not considered to be a reduction in the margin of safety. Battery operability is assured by the combination of TS Surveillance Requirements and Battery Inspection Program maintenance requirements based on IEEE-450 guidance.

Proposed changes in Surveillance Requirements and Frequencies reflect current industry guidance on maintenance and testing of the station batteries. Although some frequencies have been extended, continued performance of maintenance activities in accordance with IEEE-450, in addition to the required Surveillance Requirements, ensures that corrective maintenance can be performed prior to a condition challenging an operability limit.

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 No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: June 27, 1996

Description of amendment request: The proposed amendment will modify Technical Specification 3/4.3.3.6, ''Accident Monitoring Instrumentation,'' based on the Combustion Engineering improved Standard Technical Specifications (STS) issued by the NRC as NUREG 1432. The amendment will also revise the Technical Specification (TS) to include Accident Monitoring Instrumentation as recommended by Regulatory Guide (RG) 1.97, Revision 3.

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 change deletes all non-Type A and non-Category 1 instruments from the requirements of TS 3/4.3.3.6, ''Accident Monitoring Instrumentation.'' Type A variables provide the primary information required to permit the control room operators to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions during a DBA [Design Basis Accident]. Category 1, non-Type A variables are important in reducing public risk and are retained in TS because they are intended to assist operators in minimizing the consequences of accidents. Category 2 instruments are generally designated for indicating system operating status and are not designated as essential key variables necessary for the safe shutdown of the plant. The proposed change preserves the safety requirements of RG 1.97, Revision 3, and will not adversely affect any material condition of the plant that could directly contribute to causing or mitigating the affects of an accident.

The proposed change also adds two parameters to TS 3/4.3.3.6 which were previously controlled administratively or per another TS. Containment Pressure (Wide Wide Range) is being added because it is a Category 1 parameter required in addition to Containment Pressure (Wide Range), which is currently in the TS. Neutron Flux is being added to distinguish the RG 1.97 channels from the non-RG 1.97 channels and to provide action and surveillance requirements consistent with the other accident monitoring instrumentation. These additions to TS 3/4.3.3.6 contribute to the overall safety of the plant and therefore in no way increase the probability or consequences of an accident previously evaluated.

Additionally, the proposed change also extends the AOTs [Allowed Outage Times] for TS 3/4.3.3.6 and replaces the HOT SHUTDOWN requirement for the number of OPERABLE channels being less than the Required Number of channels with a Special Report requirement. These changes are based on the relatively low probability of an accident occurring which would require these instruments, the passive nature of these instruments, and alternate means of monitoring available. This is consistent with the CE improved STS and associated safety analyses which have been approved and issued by the NRC as NUREG 1432.

The remainder of the proposed change provides enhancements and clarifications to TS 3/4.3.3.6 which have no potential to impact plant operations. No previous accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change will not alter the operation of the plant or the manner in which the plant is operated. No new or different failure modes have been introduced. TS 3/4.3.3.6 ensures the OPERABILITY of essential Post Accident Monitoring Instrumentation. This instrumen-tation provides information to the control room operators during an accident so that appropriate actions can be taken to mitigate the consequences of the accident. These instruments are passive in nature in that no critical automatic action is assumed to occur from these instruments. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change revises TS 3/4.3.3.6 based on the information provided in CE improved STS, NUREG 1432. The deletion and addition of specific components from the TS per this change is commensurate with the safety significance of their associated parameters. The proposed change ensures the operability of the post accident monitoring instrumentation which has been designated, by RG 1.97 and Waterford 3's associated analysis, as essential for availability during and following a DBA. The proposed change preserves the single failure criteria required for this instrumentation and maintains the level of safety currently established in the Technical Specifications. The proposed change will not affect any physical protective boundary. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: July 17, 1996 (TSCR 242, Rev. 2)

Description of amendment request: The proposed change to the Technical Specifications would allow the implementation of 10 CFR Part 50, Appendix J, Option B. This application supersedes the previously submitted application dated February 23, 1996, which was noticed in the Federal Register on March 27, 1996 (61 FR 13526).

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:

GPU Nuclear has determined that this TSCR involves no significant hazards considerations as defined by NRC in 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or occurrence or the consequences of an accident of malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report.

The proposed change implements Option B of 10 CFR 50, Appendix J on performance based containment leakage testing. The proposed change does not involve a change to the plant design or operation. Therefore, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any of the analyzed accidents or malfunctions. The proposed change does not request an allowable extension of containment testing. Therefore, a hypothetical leak could remain undetected for a greater period of time. This slight increase in risk has been determined to be insignificant as:

Type A Testing

NUREG 1493 [Performance-Based Containment Leak Test Program] determined that the effect of containment leakage on overall accident risk is small as risk is dominated by accident sequences that result in the failure or bypass of the containment. Industry wide PCILRTs [primary containment integrated leak rate tests] have demonstrated that only a small fraction of the leaks discovered during testing exceeded acceptance criteria, and that the leak rate has been only marginally above the acceptable limit. Only 3% of all leaks can be detected only by PCILRT, therefore, only 3% of the theoretical leaks are affected by the extension to the Type A test interval. Experience at Oyster Creek agrees with the industry wide data in that the majority of the detected leakage from the primary containment is found through Type B and C testing. NUREG 1493 found that these observations, together with the insensitivity of reactor accident risk to the containment leakage rate, demonstrates that increasing the Type A leakage test intervals would have a minimal impact on public risk.

Type B and C Testing

Penetrations are designed to ensure reliability of the containment isolation function. Type B penetrations use a double passive seal (e.g. o-ring, gasket) and Type C penetrations use a double isolation valve design to ensure reliability of the isolation function. Because valves perform the isolation function actively, they are more likely to fail on demand (e.g. failure to completely close on demand). To address this failure mode, Type C valves are subjected to increased design constraints and testing to ensure both acceptable leak rates and stroke times. The proposed change does not alter the installation, operation, operating environment, or testing method of these valves. Therefore, the proposed change does not introduce any new component failure modes, nor does it affect the probability of occurrence of any existing evaluated failure mode.

The failure of any single penetration barrier (isolation valve or passive seal) does not cause penetration failure. Therefore, a double failure would have to occur to cause a failure of the penetration and affect containment. Additionally, the proposed change does not change the acceptance criteria for acceptable leakage testing.

The proposed change does not alter plant design or operation, nor does it alter the allowable maximum leakage rate limit. Thus, the proposed change does not affect the probability of occurrence nor the consequences of any evaluated accident or malfunction of equipment important to safety.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of any accident or malfunction different from any accident or malfunction previously evaluated.

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. This change only involves the reduction in Type A, B, and C test frequencies, and the Type A test pressure.

Type A Testing

The only changes proposed to the Type A testing are to frequency and test pressure. As the proposed test pressure is greater than the existing test pressure, no new type of accident or malfunction is created, and the increase in pressure provides an additional margin of safety. The increase in surveillance interval cannot introduce any new type of accident or malfunction.

The PCILRT is presently performed at 20 psig. Performance of the PCILRT at PGG5GA(35 psig) will provide a more direct leak rate for analysis. P_a is the design pressure of the torus (the drywell design pressure is 44 psig, but the torus is non isolable from the drywell). Therefore, P_a will not create the possibility of the failure of the torus due to overpressurization. No new accident modes can be created by extending the test intervals. No safety related functions or components are altered as a result of this change. Therefore, no new accident or malfunction different from those evaluated in the Safety Analysis Report can result due to the increase in test pressure or increase in surveillance interval.

Type B and C Testing

The proposed change only deals with the frequency of performing Type B and C testing. It does not change what components are tested or the method of testing. There is no proposed change to the design or operation of the plant. Therefore, no new accident or malfunction different from those evaluated in the Safety Analysis Report can result due to the increase in test pressure or increase in surveillance interval.

3. Operation of the facility in accordance with the proposed amendment would not decrease the margin of safety as defined in the bases of the Technical Specifications.

Type A Testing

Except for the method of defining the test frequency and pressure at which the PCILRT is performed, the methods for performing the actual test are not changed. However, the proposed change can increase the probability that an increase in leakage could go undetected for an extended period of time. NUREG 1493 has determined that under several different accident scenarios, the increased risk of radioactivity release from containment is negligible with the implementation of these proposed changes.

Type B and C Testing

The proposed change only affects the frequency of Type B and C testing. The methods for performing the actual test are not changed. The design or operation of Type B and C components are not changed. The proposed change will result in a longer interval between tests of good performing Type B and C components.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a, which is defined by the Oyster Creek Technical Specifications to be 1.0 percent by weight of the containment air at 35 psig per 24 hours. The limitation on containment leakage rate is designed to ensure the total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a). The margin of safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the 1.0 L_a acceptance criteria. The L_a value is not being modified by this proposed Technical Specification change request.

Therefore, the margin of safety as defined in the bases for the Technical Specification will not be reduced.

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: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

Attorney for licensee: Ernest L. Blake, Jr., Esquire. Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: June 28, 1996

Description of amendment request: This amendment would allow implementation of Option B to 10 CFR Part 50, Appendix J, which permits performance based determination of the frequency of containment leak rate testing.

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 (SHC), which is presented below:

The proposed change has been evaluated against the standards in 10 CFR 50.92 and determined not to involve a significant hazards consideration, in that the editorial changes do not change the meaning or intent of the technical specifications, and operation of the facility in accordance with the proposed amendment.

1. Would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated, because the proposed changes are either purely administrative changes (involving format, wording, or reporting requirements) or changes in containment leakage test requirements (minor scope changes or increased intervals between containment leakage tests). None of these changes are related to conditions which cause accidents. The proposed changes do not involve a change to the plant design or operation.

NUREG-1493, ''Performance-Based Containment Leak-Test Program,'' contributed to the technical bases for Option B of 10 CFR 50 Appendix J. NUREG-1493 contains a detailed evaluation of the expected leakage from containment and the associated consequences. The increased risk due to lengthening of the intervals between leakage tests was also evaluated and found to be acceptable. Using a statistical approach, NUREG-1493 determined the increase in the expected dose to the public from extending the testing frequency to be extremely small.

2. Would not create the possibility of a new or different kind of accident from any accident previously evaluated, because the testing or reporting requirements associated with this change do not involve a physical alteration of the plant design or changes in the methods governing normal plant operation. No safety related equipment or safety related functions are altered as a result of this change. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents.

3. Would not involve a significant reduction in a margin of safety because the proposed changes are either purely administrative (involving format, wording, or reporting requirements) or changes in containment leakage test requirements (minor scope changes or increased intervals between containment leakage tests) such that the allowable containment leakage rates presently specified in the Technical Specifications remain unchanged. The Technical Specifications and the Reactor Building Leakage Rate Testing Program will ensure that containment system testing is performed in full compliance with 10 CFR 50 Appendix J.

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:

Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 1, 1995, as supplemented by letters dated June 22, August 28, November 22, and December 19, 1995, and January 4, 8 (two letters), and 23, June 27, and July 9, 1996.

Description of amendment request: The proposed amendment would allow extension of the standby diesel generator allowed outage time to 14 days, and extension of the essential cooling water loop and the essential chilled water loop allowed outage times to 7 days. The proposed change would also add to Administrative Controls a description of the Configuration Risk Management Program (CRMP) used to assess changes in core damage probability resulting from applicable plant configurations. This application was previously published in the Federal

Register on February 8, 1996, (61 FR 4805).

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 Standby Diesel Generators are not accident initiators, therefore the increase in Allowed Outage Times for this system does not increase the probability of an accident previously evaluated. The three train design of the South Texas Project ensures that even during the seven days the Essential Cooling Water loop or the Essential Chilled Water loop is inoperable there are still two complete trains available to mitigate the consequences of any accident. If the Essential Cooling Water and the Essential Chilled Water loops are operable during the 14 days the Standby Diesel Generator is inoperable, the Engineered Safety Features bus and equipment in the train associated with the inoperable Standby Diesel Generator will be operable. This ensures that all three redundant safety trains of the South Texas Project design are operable. In addition the Emergency Transformer will be available to supply the Engineered Safety Features bus normally supplied by the inoperable Standby Diesel Generator. These actions will ensure that the changes do not involve a significant increase in the consequences of previously evaluated accidents.

The addition of the Configuration Risk Management Program to the Administrative Section of the Technical Specifications does not affect current accident analyses.

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

The proposed changes affect only the magnitude of the Standby Diesel Generator, Essential Cooling Water and the Essential Chilled Water Allowed Outage Times as identified by the marked-up Technical Specification. As indicated above, the proposed change does not involve the alteration of any equipment nor does it allow modes of operation beyond those currently allowed. Therefore, implementation of these proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes result in no significant increase in core damage or large early release frequencies. Three sets of PSA [probabilistic safety assessment] results have been presented to the NRC for the South Texas Project. One submitted in 1989 from the initial Level 1 PSA of internal and external events with a mean annual average CDF [core damage frequency] estimate of 1.7E-4, a second one submitted in 1992 to meet the IPE [individual plant examination] requirements from the Level 2 PSA/IPE with a CDF estimate of 4.4E-5, and an update of the PSA that was reported in the August 1993 Technical Specifications submittal with a variety of CDF estimates for different assumptions regarding the rolling maintenance profile and different combinations of modified Technical Specifications. The South Texas Project PSA was updated in March of 1995 to include the NRC approved Risk-Based AOTs [allowed outage times] and STIs [surveillance test intervals], Plant Specific Data and incorporate the Emergency Transformer into the model. This update resulted in a CDF estimate of 2.07E-5 per reactor year. When the requested changes are modeled, the resulting CDF estimate is 2.18E 10-5 (sic) [2.18E-5] per reactor year. This corresponds to 5.2% decrease in the Core Damage Frequency calculated for the previously submitted 21 Day AOT. The Large, Early Release Frequency is quantified as 4.69E-07 per reactor year which represents a decrease of 7.5% from the value calculated for the previously submitted 21 Day AOT. Therefore, it is concluded that there is no significant reduction in the margin of safety.

Based on the above evaluation, the South Texas Project has concluded that these changes do not involve a significant hazards consideration.

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

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036-5869

NRC Project Director: William D. Beckner

IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: July 5, 1996

Description of amendment request: The proposed Technical Specification (TS) amendment would support implementation of Noble Metal Chemical Addition (NMCA) at the Duane Arnold Energy Center (DAEC) as a method to enhance the effectiveness of Hydrogen Water Chemistry (HWC) in mitigating Intergranular Stress Corrosion Cracking (IGSCC) in Boiling Water Reactor (BWR) vessel internal components. The proposed amendment would raise the reactor water conductivity limit in STARTUP and HOT SHUTDOWN only during the application of NMCA. The reactor water conductivity will be restored after the NMCA.

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 TS amendment will not significantly increase the probability or consequences of any previously evaluated accidents.

It is expected that during the NMCA application period, the reactor water conductivity will increase and exceed the conductivity limit of 2.0 [micro]mhos/cm specified in our current TS. Our current TS requires that whenever the reactor is in STARTUP or HOT SHUTDOWN Mode, the conductivity shall not exceed 2.0 [micro]mhos/cm for more than 48 continuous hours or be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

The expected increase in conductivity is due to the presence of noble metal chemistry in the reactor water and is appropriate during the [NMCA] application period. The deposited layer of noble metals is beneficial for mitigating IGSCC in reactor vessel internal components. Other reactor water chemistry parameters such as chloride and sulfate are not expected to change; pH is expected to change but not out of the acceptable range. The reactor water chemistry parameters will be analyzed to ensure they are within the normal range, on a frequency consistent with the existing TS, Sections 4.6.B.2.c and 4.6.B.2.d when conductivity is elevated during the NMCA application.

During and after the application, the Reactor Water Cleanup (RWCU) system will continue to operate to remove the excess ions from the reactor water and restore the reactor water conductivity to the limit specified in Section 3.6.B. Therefore, this proposed TS amendment will not significantly increase the probability or consequences of any previously evaluated accidents.

2. The proposed TS amendment will not create the possibility of a new or different kind of accident. The proposed TS amendment will only permit a higher value of the reactor water conductivity limit during the application period of NMCA. The application is anticipated to increase the reactor water conductivity.

During and after the application, the RWCU system will continue to operate to remove the excess ions and restore the reactor water conductivity to the limit specified in Section 3.6.B. As is discussed above, the deposited layer of noble metals is beneficial for mitigating IGSCC in reactor vessel internal components. Therefore, this proposed TS amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS amendment will only permit a higher value of the reactor water conductivity limit during the application period of NMCA. The increase in

conductivity is anticipated during the application and is appropriate. The deposited layer of noble metals is beneficial for mitigating IGSCC in reactor vessel internal components. During and after the application, the RWCU system will continue to operate to remove the excess ions and restore the reactor water conductivity to the limit specified in Section 3.6.B. Therefore, no margin of safety is reduced as a result of the anticipated increase in conductivity due to the addition of the known noble metals.

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

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis, & Bockius, 1800 M Street, NW., Washington, DC 20036-5869

NRC Project Director: Gail H. Marcus

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: June 21, 1996

Description of amendment request: The proposed amendment would modify Section 5.7, ''High Radiation Areas,'' of the ''Administrative Controls'' section of the Clinton Power Station technical specifications (TS). The proposed changes include: (1) allowing utilization of a Radiation Work Permit (RWP) ''or equivalent'' to control entry into a high radiation area; (2) clarifying the example given in the TS of individuals who are qualified in radiation protection procedures; (3) clarifying the requirements for when specified access controls and barriers for high radiation areas within large areas like the containment must be established; (4) clarifying that it is acceptable for an RWP to specify a maximum dose, i.e., a specified setpoint on an alarming dosimeter in lieu of a stay time for entry into a high radiation area (where an individual could receive a deep dose equivalent of 3000 mrem in one hour); (5) eliminating the upper dose limit for specifying the applicability of the requirements of Specification 5.7.1; (6) providing additional flexibility regarding who may control the keys to locked doors for preventing unauthorized entry into high radiation areas; (7) reorganizing TS Sections 5.7.1, 5.7.2, and 5.7.3 into four sections (5.7.1, 5.7.2, 5.7.3 and 5.7.4); and (8) making minor edits to enhance readability.

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) None of the proposed changes involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed changes do not change the design or the operation of the plant. The proposed changes are only related to the control of access to high radiation areas for the purpose of controlling dose to plant personnel. Because no change to plant design is proposed, there is no impact to any accident mitigating system. Likewise, because there is no proposed change to plant operating procedures, plant operation is not impacted. This proposed change does not impact any accident scenario or the previously calculated post-accident doses. Therefore, the limits of 10 CFR 100 will continue to be met. No probability or consequence of any accident previously evaluated is impacted by the proposed changes to TS.

(2) None of the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment is administrative in nature and does not impact directly or indirectly the design or the operation of the Clinton Power Station, thus no new accident can be created.

(3) None of the proposed changes involve a significant reduction in a margin of safety.

There is no reduction to the margin of safety because the operating limits and functional capabilities of plant safety systems are unaffected by the proposed changes to administrative requirements. As noted previously, the proposed changes do not impact any accident analyses, including the associated dose calculations. With respect to controls for controlling operational dose to plant personnel, the proposed changes are intended to provide clarity and/or flexibility with respect to the administration and programmatic controls for controlling such dose, and yet maintain an adequate margin of safety for minimizing dose to site personnel consistent with the requirements of 10 CFR 20 and guidance of Regulatory Guide 8.38.

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: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525

NRC Project Director: Gail H. Marcus

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: June 28, 1996

Description of amendment request: The proposed amendment would allow removal of the Inclined Fuel Transfer System (IFTS) primary containment blind flange while primary containment is required to be operable. This will provide flexibility to operate the IFTS for the purpose of testing and exercising the system during such conditions. Primary containment integrity will be provided by an alternate means while the blind flange is removed. The change would be incorporated via a provisional note into Technical Specification (TS) Surveillance Requirement 3.6.1.3.3, associated with TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs).''

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 allows operation of the IFTS while primary containment operability is required. The proposed change does not involve any modifications to plant systems or design parameters or conditions that contribute to the initiation of any accidents previously evaluated. Therefore, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak-tight integrity of the containment structure which is designed to mitigate the consequences of a loss-of-coolant accident (LOCA). The function of the primary containment is to maintain functional integrity during and following the peak transient pressures and temperatures that result from any LOCA. The primary containment is designed to limit fission product leakage following the design basis LOCA. Because the proposed change does not alter the plant design, only the extent of the boundaries that provide primary containment isolation for the IFTS penetration, the proposed change does not result in an increase in primary containment leakage. However, temporarily using the IFTS transfer tube and its attached appurtenances as part of the primary containment boundary (which have not been fabricated or installed to exactly the same requirements as a fully certified primary containment penetration) can increase the probability that a LOCA would challenge the pressure retaining integrity of these components. Since the subject components have been built to withstand pressure, temperature, and seismic conditions similar to those of the existing penetration, they are judged to be an acceptable barrier to prevent the uncontrolled release of post-accident fission products for the purposes of this amendment request.

Further, it has been shown that the largest potential leakage pathway, the IFTS transfer tube itself, would remain sealed by the depth of water required to be maintained in the fuel building fuel transfer pool. The transfer tube drain line constitutes the other possible leakage pathway, and will be required to be capable of being isolated via administrative control of the manual isolation valve in the drain line. Additionally, due to the physical relationships of the buildings and components involved, any leakage from either of these pathways is fully contained within the boundaries of the secondary containment and would be filtered by the Standby Gas Treatment System prior to release to the environment.

Based on the above, Illinois Power has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed change does not involve a change to the plant design or operation (except when the IFTS is operated). As a result, the proposed change does not affect any of the parameters or conditions that could contribute to the initiation of any accidents. No new accident modes are created by this change. Extending the primary containment boundary to include portions of the IFTS has no influence on, nor does it contribute to the possibility of a new or different kind of accident or malfunction from those previously analyzed.

Based on the above, Illinois Power has concluded that the proposed change will not create the possibility of a new or different kind of accident not previously evaluated.

(3) The request does not involve a significant reduction in a margin of safety. The proposed change only affects the extent of a portion of the primary containment boundary. Precautions will be taken to administratively control the IFTS transfer tube drain path so that the proposed change will not increase the probability that an increase in leakage from the primary containment to the secondary containment could occur.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a, which is defined by the Clinton Power Station Technical Specifications to be 0.65% of primary containment air weight per day at the calculated peak constant pressure (P_a). The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a). The margin of safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the 1.0 L_a acceptance criteria. The L_a value is not being modified by this proposed technical specification change. The IFTS will continue to provide an acceptable barrier to prevent containment leakage during a LOCA, and therefore this change will not create a situation causing the containment leakage rate acceptance criteria to be violated.

As a result, Illinois Power has concluded that the proposed change will not result in a reduction in a margin of safety.

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

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525

NRC Project Director: Gail H. Marcus

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

Date of amendment requests: June 11, 1996 (AEP:NRC:80027)

Description of amendment requests: The proposed amendments would remove from the technical specifications (TS) certain requirements for administrative controls, related to quality assurance requirements, in accordance with the guidance of NRC Administrative Letter 95-06,≥Relocation of Technical Specifications Administrative Controls Related to Quality Assurance.''

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:

We have evaluated the proposed T/S changes and have determined that the changes should involve no significant hazards consideration based on the criteria established in 10 CFR 50.92(c). Operation of Cook Nuclear Plant in accordance with the proposed amendment will not satisfy any of the following criteria:

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

The proposed change does not involve any physical alteration of plant configurations, changes to setpoints, or operating parameters. This proposed amendment is to relocate the T/S requirements for administrative controls that are related to quality assurance to the QAPD [Quality Assurance Program Description]. This is in accordance with the guidance provided in AL 95-06. Also, the relocated requirements and future changes are controlled by 10 CFR 50.54(a) which requires prior NRC approval for changes that reduce the commitments in the program description previously accepted by the NRC. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

(b) Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change does not involve any physical alteration of plant configurations, changes to setponts, or operating parameters. This proposed amendment is to relocate the T/S requirements for administrative controls that are related to quality assurance to the QAPD. This is in accordance with the guidance provided in AL 95-06. Also, the relocated requirements and future changes are controlled by 10 CFR 50.54(a) which requires prior NRC approval for changes that reduce the commitments in the program description previously accepted by the NRC. Therefore, this proposed change does not create the possibility of a new of different kind of accident from any previously analyzed.

(c) Involve a significant reduction in a margin of safety.

The proposed change does not involve any physical alteration of plant configurations, changes to setpoints, or operating parameters. This proposed amendment is to relocate the T/S requirements for administrative controls that are related to quality assurance to the QAPD. This is in accordance with the guidance provided in AL 95-06. Also, the relocated requirements and future changes are controlled by 10 CFR 50.54(a), which requires prior NRC approval for changes that reduce the commitments in the program description previously accepted by the NRC. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

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

NRC Project Director: Mark Reinhart, Acting

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

Date of amendment request: June 19, 1996 [AEP:NRC:1166AA]

Description of amendment request: The proposed amendment would modify the technical specifications (T/

S) to allow continued use of the 2-volt steam generator (SG) tube plugging criteria for future operating cycles as discussed in NRC Generic Letter 95-05, ''Voltage-Based Repair Criteria for the Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.''

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:

In accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed license amendment is analyzed using the following standards and found not to: 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in margin of safety. Conformance of the proposed amendment to the standards for a determination of no significant hazards as defined in 10 CFR 50.92 (three factor test) is shown in the following paragraphs:

1) Operation of Cook Nuclear Plant Unit 1, in accordance with the proposed license amendment, does not involve a significant increase in the probability or consequences of an accident previously evaluated. Testing of model boiler specimens for free span tubing

(no TSP [tube support plate] restraint) at room temperature conditions show burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking [ODSCC] with voltage measurements as high as 19 volts. Burst testing performed on pulled tubes from Cook Nuclear Plant Unit 1 with up to a 2.02 volt indication shows measured burst pressure in excess of 10,000 psi at room temperature. Burst testing performed on pulled tubes from other plants show burst pressures in excess of 5,300 psi at room temperatures. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst resistance significantly exceeds the safety factor requirements of RG [Regulatory Guide] 1.121 [Bases for Plugging Degraded PWR Steam Generatory Tubes]. As stated earlier, tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the TSP. Test data indicates that tube burst cannot occur within the TSP, even for tubes which have 100% throughwall electric-discharge machined notches 0.75 inch long, provided the TSP is adjacent to the notched area. Since tube-to-tube support plate proximity precludes tube burst during normal operating conditions, it follows that use of the proposed plugging criteria must, therefore, retain tube integrity characteristics which maintain the RG 1.121 margin of safety of 1.43 times the bounding faulted condition (steam line break) pressure differential.

During a postulated main SLB [steamline break], the TSP has the potential to deflect during blowdown, thereby uncovering the intersection. Based on the existing data base, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the SLB pressure differential on tube burst is satisfied by 7/8 inch diameter tubing with bobbin coil indications with signal amplitudes less than V_{SL}, regardless of the indicated depth measurement. A 2 volt plugging criteria compares favorably with the current V_{SL} (8.8 volt) structural limit, considering the previously calculated growth rates for ODSCC within Cook Nuclear Plant Unit 1 SGs. Considering a voltage growth component of 0.8 volts (40% voltage growth based on 2 volts BOC [beginning of cycle] and a nondestructive examination uncertainty of 0.40 volts (20% voltage uncertainty based on 2 volts BOC), when added to the BOC plugging criteria of 2 volts, results in a bounding EOC [end of cycle] voltage of approximately 3.2 volts for a cycle operation. A 5.6 volt safety margin exists (8.8 - 3.2 volt EOC = 5.6 volt margin).

For the voltage/burst correlation, the EOC structural limit is supported by a voltage of 8.8 volts. Using this V_{SL} of 8.8 volts, a BOC maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit a significant number of EOC indications to exceed the V_{SL} and should assure that acceptable tube burst probabilities are attained. By adding NDE [nondestructive examination] uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. The previous plugging criteria submittal established the conservative NDE uncertainty limit (V_{NDE}) of 20% of the BOC repair limit. For consistency, a 40% voltage growth allowance (V_{GR}) to the BOC repair limit is also included. This allowance is extremely conservative for Cook Nuclear Plant Unit 1. Therefore, the maximum allowable upper voltage repair limit V_{URL} for BOC, based on the V_{SL} of 8.8 volts, can be represented by the expression:

$V_{URL} + (V_{NDE} \text{ x } V_{URL}) + (V_{GR} \text{ x } V_{URL}) = 8.8$ volts, or,

the maximum allowable BOC repair limit can be expressed as, $V_{URL} = 8.8$ volt structural limit/1.6 = 5.5 volts.

This structural repair limit supports this application for plugging criteria implementation to repair bobbin indications greater than 2 volts based on RPC [rotating pancake coil] confirmation of the indication. Conservatively, an upper limit of 5.5 volts will be used to repair bobbin coil indications which are above 2 volts but do not have confirming RPC calls.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main SLB outside of containment, but upstream of the main steam isolation valve, represents the most limiting radiological condition relative to the plugging criteria. In support of implementation of the plugging criteria, it will be determined whether the distribution of crack indications at the TSP intersections at the EOC are projected to be such that primary-to-secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines. A separate calculation has determined this allowable SLB leakage limit to be 8.4 gpm. Although not required by the Cook Nuclear Plant design basis, this calculation uses the recommended Iodine-131 transient spiking values consistent with NUREG-0800 [Standard Review Plan], and the T/S reactor coolant system activity limit of 1 micro curie per gram dose equivalent Iodine-131. Control room dose calculations were also performed and found to be less limiting than the offsite dose leakrate. Therefore, the more conservative offsite dose leakrate is used. The projected SLB leakage rate calculation methodology prescribed in GL 95-05 and WCAP 14277 [Steam Line Break Leak Rate and Tube Burst Probability Analysis Methods for Outside Diameter Stress Corrosion Cracking at Tube Support Plate Intersections] will be used to calculate EOC leakage, based on actual EOC distributions and EOC projected distributions. Due to the relatively low voltage growth rates at Cook Nuclear Plant Unit 1 and the relatively small number of indications affected by the plugging criteria, SLB leakage prediction per GL 95-05 is expected to be significantly less than the permissible level of 8.4 gpm in the faulted loop.

The inclusion of all intersections in the leakage model, along with application of a probability of detection of 0.6, will result in extremely conservative leakage estimations. Close examination of the available data shows that indications of less than 2.8 volts will not be expected to leak during SLB conditions.

The proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated within the cook Nuclear Plant Unit 1 Final Safety Analysis Report (FSAR).

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

Implementation of the proposed SG tube plugging criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the TSP elevations. Neither a single nor a multiple tube rupture event would, under any plant conditions, be expected in a SG in which the plugging criteria has been applied. Specifically, we will continue to implement a maximum leakage rate limit of 150 gpd (0.1 gpm) per SG to help preclude the potential for excessive leakage during all plant conditions. The T/S limits imposed on primary-to-secondary leakage at operating conditions are a maximum of 0.4 gpm (600 gpd) for all SGs with a maximum of 150 gpd allowed for any one SG.

The RG 1.121 criteria for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break (LBB) considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. Regulatory Guide 1.121 acceptance criteria for establishing operating leakage limits are based on LBB considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at faulted conditions maximum pressure differential. A voltage amplitude of 8.8 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95% prediction limit on the burst correlation coupled with 95/95 lower tolerance limit material properties. Alternate crack morphologies can correspond to 8.8 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations were used to define the ‘‘longest permissible crack’’ for evaluating operating leakage limits. Consistent with the cycle 13, 14 and 15 license amendment requests for plugging criteria, and Section 5 of Enclosure 1 of the GL, operational leakage limits will remain at 150 gpd per SG. Axial cracks leaking at this level are expected to provide LBB protection at both the SLB pressure differential of 2560 psi and, while not part of any established LBB methodology, LBB protection will also be provided at a value of 1.43 times the SLB pressure differential. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions. Additionally, this LBB evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection.

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

The use of the voltage-based bobbin probe interim TSP elevation plugging criteria at Cook Nuclear Plant Unit 1 is demonstrated to maintain SG tube integrity commensurate with the criteria of RG 1.121. Regulatory Guide 1.121 describes a method acceptable to the NRC staff for meeting GDC [General Design Criteria] 14, 15, 31, and 32 by reducing the probability or the consequences of SG tube rupture. This is accomplished by determining the limiting conditions of degradation of SG tubing, as established by in-service inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a SG tube rupture event during normal or faulted plant conditions. It will be confirmed by analysis and calculation that EOC distribution of crack indications at the TSP elevations will result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of a LOCA [loss-of-coolant accident] and SSE [safe-shutdown earthquake] on the SG component (as required by GDC 2), it has been determined that tube collapse may occur in the SGs at some plants. The postulated tube collapse results from a deformation of TSPs as a result of lateral loads at the wedge supports at the periphery of the plate. The lateral loads result from the combined effects of the LOCA rarefaction wave and SSE loadings. The resulting pressure differential on the deformed tubes may then cause some of the tubes to collapse.

There are two issues associated with a postulated SG tube collapse. First, the collapse of SG tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, since the LBB methodology is applicable to the Cook Nuclear Plant Unit 1 reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. Loss of coolant accident loads for the primary pipe breaks were used to bound the Cook Nuclear Plant Unit 1 smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in SG tube collapse or significant deformation.

Addressing RG 1.83 [In-Service Inspection of PWR Steam Generator Tubes] considerations, implementation of the bobbin coil probe, voltage-based interim tube plugging criteria of 2 volts is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the TSP elevation per T/S, and MRPC [motorized RPC] inspection requirements for the larger indications left in-service to characterize the principal degradation as ODSCC.

As noted previously, implementation of the TSP elevation plugging criteria will decrease the number of tubes which must be repaired. The installation of SG tube plugs reduces the RCS flow margin. Thus, implementation of the plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the FSAR or any Bases of the plant T/Ss.

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: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

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NRC Project Director: Mark Reinhart, Acting

Northern States Power Company, Docket No. 50-282, Prairie Island Nuclear Generating Plant, Unit No. 1, Goodhue County, Minnesota

Date of amendment request: July 15, 1996

Description of amendment request: The proposed amendment would allow the use of the moveable incore detector system for measurement of the core peaking factors with less than 75% and greater than or equal to 50% of the detector thimbles available. The amendment request is a one-time only change for Prairie Island, Unit 1, Operating Cycle 18. It is being submitted to allow for continued operation if the number of detector thimbles drops below 75%.

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

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

The proposed changes do not involve an increase in the probability of an accident previously evaluated. The moveable incore detector system is used only to provide confirmatory information on the neutron flux distribution and is not required for the daily safe operation of the core. The system is not a process variable that is an initial condition in the accident analyses. The only accident that the moveable incore detector system could be involved in is the breaching of the detector thimbles which would be enveloped by the small break loss of coolant accident (LOCA) analysis. As the proposed changes do not involve any changes to the system’s equipment and no equipment is operated in a new or more harmful manner, there is no increase in the probability of such an accident.

The proposed amendments would not involve an increase in the consequences of an accident previously evaluated. The moveable incore detector system provides a monitoring function that is not used for accident mitigation (the system is not used in the primary success path for mitigation of a design basis accident). The ability of the reactor protection system or engineered safety features system instrumentation to mitigate the consequences of an accident will not be impaired by the proposed changes. The small break LOCA analysis (and thus its consequences) continues to bound potential breaching of the system's detector thimbles.

With greater than or equal to 50% and less than 75% of the detector thimbles available, core peaking factor measurement uncertainties will be increased, which could impact the core peaking factors and as a result could affect the consequences of certain accidents. However, any changes in the core peaking factors resulting from increased measurement uncertainties will be compensated for by conservative measurement uncertainty adjustments in the Technical Specifications to ensure that pertinent core design parameters are maintained. Sufficient additional penalty is added to the power distribution measurements such that this change will not impact the consequences of any accident previously evaluated.

Therefore, based on the conclusions of the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendments would not create the possibility of a new or different kind of accident previously evaluated as they only affect the minimum complement of equipment necessary for operability of the moveable incore detector system. There is no change in plant configuration, equipment or equipment design. No equipment is operated in a new manner. Thus the changes will not create any new or different accident causal mechanisms. The accident analysis in the Updated Safety Analysis Report remains bounding.

Therefore, based on the conclusions of the above analysis, the proposed changes will not create the possibility of a new or different kind of accident.

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

The proposed changes will not involve a significant reduction in a margin of safety. The reduction in the minimum complement of equipment necessary for the operability of the moveable incore detector system could only impact the monitoring/calibration functions of the system. Reduction of the number of available moveable incore detector thimbles to the 50% level does not significantly degrade the ability of the system to measure core power distributions. With greater than or equal to 50% and less than 75% of the detector thimbles available, core peaking factor measurement uncertainties will be increased, but will be compensated for by conservative measurement uncertainty adjustments in the Technical Specifications to ensure that pertinent core design parameters are maintained. Sufficient additional penalty is added to the power distribution measurements such that this change does not impact the safety margins which currently exist. Also, the reduction of available detector thimbles has negligible impact on the quadrant power tilt and core average axial power shape measurements. Sufficient detector thimbles will be available to ensure that no quadrant will be unmonitored.

Based on these factors, the proposed changes in this license amendment will not result in a significant reduction in the plant's margin of safety, as the core will continue to be adequately monitored.

Based on the evaluation above, and pursuant to 10 CFR 50, Section 50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR 50, Section 50.92.

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

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

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

NRC Project Director: Mark Reinhart, Acting

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

Date of amendment request: May 31, 1996

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to add a Limiting Condition for Operation (LCO) for trisodium phosphate (TSP) and increase the minimum required amount of TSP contained in the containment sump mesh baskets.

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.

Trisodium Phosphate Dodecahydrate (TSP) is stored in the containment sump to raise the pH of the sump and spray water following a loss of coolant accident (LOCA). As the pH of the water increases, more radioactive iodine is kept in solution and the possibility of airborne radioactivity leakage is decreased. An additional advantage of a higher pH is the beneficial reduction in chloride stress corrosion cracking (SCC) of austenitic stainless steel components in the containment following a LOCA.

This chemical is an accident mitigator, not an accident initiator in that it is not used until after an accident (i.e., a LOCA) has occurred. At the time it begins to go into solution, the accident has occurred, containment spray has been activated and water is collecting in the containment sump. Therefore, increasing the Technical Specification (TS) minimum amount of TSP verified to be in containment will not involve a significant increase of the probability of an accident previously evaluated.

The Updated Safety Analysis Report (USAR), Section 14.15, ''Loss of Coolant Accident,'' does not take credit for a post-LOCA minimum containment sump pH adjustment to 7.0 for the iodine removal and retention calculation until ten hours after initiation of the event. Increasing the amount of TSP (based on recent re-analysis) in the containment sump ensures that a pH greater than or equal to 7.0 is achieved and therefore does not increase the consequences of any accident previously evaluated.

The proposed change to TS 2.3(4) represents a new Limiting Condition for Operation (LCO) which is added to establish overall consistency with the CE STS [Combustion Engineering Standard Technical Specifications] for TSP requirements. The proposed change establishes a minimum TSP volume that must be maintained during operating Modes 1 and 2 to ensure that a pH greater than or equal to 7.0 is achieved within four hours following a LOCA; as well as, establishing times for accomplishing corrective actions should the LCO not be met. Therefore, this change does not significantly increase the probability or consequences of any accident previously evaluated.

The proposed change to TS 3.6(2)d(i) revises the required surveillance inventory of the TSP baskets consistent with the aforementioned calculation to ensure that a pH greater than or equal to 7.0 is achieved. Therefore, this change does not increase the consequences of any accident previously evaluated.

The proposed change to TS 3.6(2)d(ii) moves the surveillance test amounts of chemical and water used from the Specification to the Basis section. This relocation will not alter the test method or acceptance criteria.

In the Basis, the amount of TSP used in the test is changed to reflect the ratio of TSP to water that would be found in the containment sump following a LOCA. The specified concentration of boron in the test reflects the highest concentration that could be found in the containment sump following a LOCA. The test temperature is changed to 115 - 125°F, which is well below the temperature expected to be found in the containment sump following a LOCA. The decanting of the solution does not change the intent of the test method since the dissolving period will still be conducted without agitation. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

TSP is currently present in the containment sump. The addition of TSP ensures that a pH greater than or equal to 7.0 is achieved following a LOCA. The increase in TSP inventory will be accomplished via a modification to be installed during the 1996 Refueling Outage.

The proposed change to TS 2.3(4) represents a new LCO which is added to establish overall consistency with the CE STS for TSP requirements. The proposed change establishes a minimum TSP volume that must be maintained during operating Modes 1 and 2 to ensure that a pH greater than or equal to 7.0 is achieved following a LOCA, as well as, establishing corrective action term limits should the LCO not be met. This proposed change does not create a possibility of a new or different kind of accident from any previously analyzed.

The proposed change to TS 3.6(2)d(ii) moves the surveillance test amounts of chemical and water used from the Specification to the Basis section to be consistent with the CE STS. This relocation will not alter the test method or acceptance criteria. In the Basis section, the amount of TSP used in the test is changed to reflect the ratio of TSP to water that would be found in the containment following a LOCA. The specified concentration of boron in the test reflects the highest concentration that could be found in the containment sump following a LOCA. The test temperature is changed to a range of 115 - 125°F which is well below the temperature expected to be found in the containment sump following a LOCA. The decanting of the solution does not change the intent of the test method since the dissolving period will still be conducted without agitation. Therefore, these changes will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

TSP is stored in the containment lower level to raise the pH of the containment sump and recirculated spray water following a LOCA. As the pH of the water increases, more radioactive iodine is kept in solution and the possibility of airborne radioactivity leakage is decreased. Additionally, a higher pH has the beneficial effect of reducing the possibility of chloride stress corrosion cracking of austenitic stainless steel components in the containment.

The proposed change to TS 2.3(4) represents addition of a new LCO for TSP requirements during power operations and hot standby consistent with CE STS. This change does not involve a significant reduction in a margin of safety.

TS 3.6(2)d(i) requires verification that a minimum volume of TSP is contained in the storage baskets in containment. This change proposes to increase that volume consistent with the latest ABB/CE calculation. The increased volume will ensure that the containment sump, when filled with water from the Reactor Coolant System, Safety Injection Refueling Water Tank, Safety Injection Tanks and Boric Acid Storage Tanks, will have a pH greater than or equal to 7.0 within four hours following a LOCA. Therefore, this change does not involve a reduction in a margin of safety.

The proposed change to TS 3.6(2)d(ii) would move the surveillance test amounts of chemical and water used from the Specification to the Basis section. This relocation is consistent with the CE STS and will not alter the test method or acceptance criteria. In the Basis, the amount of TSP used in the test is changed to reflect the ratio of TSP to water that would be found in the containment following a LOCA. The specified concentration of boron in the test reflects the highest post-LOCA concentration that could be found in the containment. The test temperature is changed to a range of 115 - 125°F which is well below the temperature expected to be found in the containment sump following a LOCA. The decanting of the solution does not change the intent of the test method since the dissolving period will still be conducted without agitation. Therefore, these changes do not involve a significant reduction in a margin of safety.

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

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NRC Project Director: William H. Bateman

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

Date of amendment request: July 15, 1996

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to allow the use of either zircaloy or ZIRLO cladding and add a reference to Westinghouse Topical Report, WCAP-12610, June 1990.

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

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

The proposed revision to TS 4.3.2 is based on improved STS 4.2 of NUREG-1432. ZIRLO is similar in chemical composition, physical and mechanical properties to Zircaloy-4, but features improved corrosion performance and dimensional stability. These characteristics ensure that fuel rod cladding integrity and fuel assembly structural integrity are maintained. Fuel assemblies manufactured with ZIRLO clad fuel rods meet the same design bases requirements as fuel assemblies manufactured with Zircaloy-4 cladding and the regulatory requirements of 10 CFR 50.46 are applicable to either material.

No concerns have been identified pertaining to reactor operation with a core comprised of fuel assemblies manufactured with Zircaloy-4 clad rods and fuel assemblies manufactured with ZIRLO clad rods. ZIRLO clad fuel rods do not require a change to the FCS [Fort Calhoun Station] reload design and safety analysis limits. Radiological consequences of previously evaluated accidents are not increased because the safety analysis dose predictions are not sensitive to the type of cladding material used. The proposed limited substitution of zirconium alloy or stainless steel filler rods in accordance with NRC-approved fuel rod configurations will allow leaking fuel rods (or potential leakers) to be removed. Therefore, the radiological consequences of accidents previously evaluated in the FCS Updated Safety Analysis Report (USAR) are not increased by this change.

The revisions to TS 4.3.2 listed above will not result in a change to any of the process variables that might initiate an accident or affect the radiological release for an accident. The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC-approved methodology. There are no physical changes to the plant associated with the change to TS 4.3.2 other than the changes to the fuel assemblies. Therefore, this revision does not involve a significant increase in the probability or consequences of an accident previously evaluated because the safety analysis to be performed for each cycle will continue to demonstrate compliance with all fuel safety design bases.

The proposed revision of TS 4.3.2 is supported by Westinghouse Topical Report, WCAP-12610, ''VANTAGE + Fuel Assembly Report,'' dated June 1990 (Westinghouse Proprietary). This topical report describes the fuel rod design bases, criteria and models, which are affected by the use of ZIRLO cladding. Consequently, WCAP-12610 is proposed for addition to the list of analytical methods located in TS 5.9.5b that are used to determine the core operating limits.

Based on the above discussion, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Fuel assemblies manufactured with ZIRLO clad fuel rods must meet original design criteria and thus they will not be an initiator for any new or different kind of accident. All design and performance criteria will continue to be met by fuel assemblies manufactured with ZIRLO clad fuel rods and no new single failure mechanisms have been found.

The use of fuel assemblies manufactured with ZIRLO cladding does not involve any alterations to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors. The substitution of zirconium alloy, stainless steel filler rods, or lead test assemblies for fuel rods will be limited to NRC-approved fuel rod configurations. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created by this change.

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

The use of fuel assemblies manufactured with ZIRLO clad rods does not change the proposed FCS reload design and safety analysis limits. The normal operating conditions allowed for in the Technical Specifications will be taken into consideration for the use of these fuel assemblies. For each cycle reload core, the fuel assemblies will be evaluated using NRC-approved reload design methods to include consideration of the core physics analysis peaking factors and core average linear heat rate effects.

NRC-approved methods will also be used to analyze each configuration of zirconium alloy or stainless steel filler rods in fuel assemblies to demonstrate continued safe operation within the limits that assure acceptable plant response to accidents and transients. Therefore, this 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: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William H. Bateman

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: June 21, 1996

Description of amendment request: The proposed amendment would change the frequency of instrument channel calibrations in Table 4.1-1, ''Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels'' to accommodate operation with 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 operation with the proposed license amendment involve a significant increase in the probability or consequences of any accident previously evaluated?

Response:

The proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated. The proposed changes are being made to extend the calibration frequency to 24-months for the:

Pressurizer Pressure; Accumulator Level and Pressure; andVolume Control Tank Level.

These changes are being made, using the guidance of Generic Letter 91-04, to accommodate a 24-month operating cycle. The proposed changes in the calibration frequencies do not involve any plant hardware changes (other than alarm adjustments) or the way the systems function. The results of the instrumentation drift analysis, loop accuracy/set point calculations and the evaluation of channel uncertainties indicate the calibrations can be safely extended to accommodate the 24-month operating cycle.

The four pressurizer pressure channels are used for high and low pressure protection (i.e., reactor trip and safety injection) and for overpower-overtemperature protection. Three of the pressure channels are also used for pressure control and compensation signals for rod control. Pressurizer pressure indication is also provided in the control room for use during normal operation and while using the EOPs (emergency operating procedure). The loop accuracy/setpoint calculations confirm that sufficient margin exists between the pressurizer high and low pressure reactor trip, low pressurizer pressure SI [safety injection], and overtemperature delta-temperature analytical limits and the existing field trip settings based on an extended calibration interval. A small increase in pressurizer pressure normal indication uncertainty due to increased sensor drift is within the readability of the indicator and has been incorporated into the pressurizer pressure initial conditions used in the evaluation of channel uncertainties (Reference 15) [see application dated June 21, 1996]. The post-accident indication uncertainties remain bounded by the existing uncertainties used in the EOPs. Assurance that the RPS [reactor protection system] and ESF [engineered safety feature] instrumentation and protection logic relays will function as required is also provided by on-line surveillance (channel checks performed each shift and quarterly channel functional tests) that are designed to detect potential instrument failures and verify operability of pressurizer pressure channels.

Water level and pressure in each accumulator is monitored by two redundant channels designed to provide indication in the control room. High and low level alarm functions alert the operator to initiate operations to maintain the accumulator water volume or pressure within the Technical Specifications limits. The level and pressure instrumentation do not provide an active protective or control function and are not required to mitigate an accident condition. The level (or volume) and pressure limits are important since they are initial conditions assumed in the safety analysis. The loop accuracy/setpoint calculations for accumulator level and pressure were updated to include conservative values for 30-month calibration uncertainties using Westinghouse sensor drift values and extrapolated vendor specified uncertainties for rack and indicating components consistent with industry methods. The increased indicator uncertainty has been evaluated for both input parameters (accumulator level and pressure) assumed for the LOCA [loss-of-coolant accident] and Containment Integrity events (Reference 15) and a non significant increase in both the peak clad temperature and containment pressure was identified.

The volume control tank (VCT) level instrumentation is not required to mitigate the consequences of an accident. The instrumentation provides control room indication and initiates automatic actions of the chemical and volume control system (e.g., diverts letdown to the holdup tanks on high level, initiates makeup on low level, changes the charging pump suction on low low level). The loop accuracy/setpoint calculation for VCT level, updated based on the increased drift and uncertainty, determined that the existing setpoints remain valid to ensure the VCT instrumentation can perform the required design function.

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

Response:

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes extend the calibration frequency to 24 months for the Pressurizer Pressure, Accumulator Pressure and Level, and Volume Control Tank Level instrumentation to accommodate a 24-month operating cycle. The proposed changes in calibration frequencies do not involve any plant hardware changes, nor do they change the way that the systems function.

The extension of the calibration and surveillance test intervals were evaluated and the results, documented in Reference 15, indicate that the calibrations can be safely extended to accommodate the 24-month operating cycle.

3. Does operation with the proposed license 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. The proposed changes extend the calibration frequency to 24 months for the Pressurizer Pressure, Accumulator Pressure and Level, and Volume Control Tank Level instrumentation to accommodate a 24-month operating cycle.

The proposed changes result in an increased instrument channel uncertainty for the pressurizer pressure. An evaluation (Reference 15) has determined that: all current cycle 9 safety analysis limits based on pressurizer pressure uncertainties remain bounding for extended surveillance intervals (high and low pressure trips); the safety analysis limits for K1 (a constant used in the overtemperature [DELTA] T trip setpoint) remain applicable; and, Engineered Safety Feature Actuation System trip settings based on pressurizer pressure uncertainty remain bounding (low pressure safety injection).

The proposed changes result in an increased instrument channel uncertainty for the accumulator level and pressure. An evaluation (Reference 15) has determined that increasing the uncertainty results in non-significant (defined by 10 CFR 50.46(a)(3)(i) as less than 50°F) increases in the total peak clad temperature (less than 35°F) for the large break and small break LOCA but the values remain well within regulatory acceptance criteria. The evaluation also determined that the peak calculated pressure in containment following a LOCA would increase due to the lower bound on pressure and the higher bound on volume in the accumulators. An assessment of the approximate effect on the peak containment pressure determined that the Technical Specification integrated leak rate testing value of 42.42 psig (the licensing basis peak pressure) remains bounding.

The proposed changes result in an increased instrument channel uncertainty for the VCT level but there are no changes to any margins of safety because this instrumentation supports a control function.

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

Southern California Edison Company, et al., Docket No. 50-206, San Onofre Nuclear Generating Station, Unit No. 1, San Diego County, California

Date of amendment request: December 22, 1995

Description of amendment request: The proposed change would revise the San Onofre Unit 1 License Condition to delete a reference to License Condition 2.C(4) from License Condition 2.D. This change is being requested to eliminate a reporting requirement for violations of the physical protection plans that is redundant to reporting requirements in 10 CFR 73.71 and 10 CFR 73 Appendix G.

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

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

No. The proposed change is considered an administrative change. It has no impact on the probability or consequences of any of the accidents previously evaluated. This change revises License Condition 2.D to remove the burden of duplicate reporting requirements. This change does not affect the physical protection program as previously approved by the Nuclear Regulatory Commission (NRC).

A reporting requirement in License Condition 2.D is being revised to remove the reference to License Condition 2.C(4) for the physical protection program. The reporting requirements for the physical protection program are located in the regulations, 10 CFR 73.71 and 10 CFR 73 Appendix G.

Therefore, the probability and consequences of an accidently previously evaluated are not affected by these proposed changes.

2. Will operation of the facility according to this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated.

No. This proposed change is considered an administrative change. It has no impact on equipment, systems, or structures such that a new or different kind of accident is created. This change revises License Condition 2.D to remove duplicate and unnecessary reporting requirements for the physical protection program. There is no change associated with the implementation and maintenance of the physical protection program as previously approved by the NRC.

Therefore, the possibility of a new or different kind of accident from an accident previously evaluated is not created.

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

No. This proposed change is considered an administrative change only. It has no impact on the margin of safety associated with the physical protection program. This change revises License Condition 2.D to remove duplicative and unnecessary reporting requirements for the physical protection program. The maintenance and implementation of the physical protection program is not affected by this change.

Therefore, there will not be a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis of the licensee 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: Main Library, University of California, P.O. Box 19557, Irvine, California 92713

Attorney for licensee: James A. Beoletto, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770

NRC Project Director: Seymour H. Weiss

Southern California Edison Company, et al., Docket No. 50-206, San OnofreNuclear Generating Station, Unit No. 1, San Diego County, California

Date of amendment request: March 13, 1996

Description of amendment request: The proposed change would revise San Onofre Unit 1 License Condition 2.D in the Operating (Possession Only) License to remove a reporting requirement that is redundant to reporting requirements in 10 CFR 50.72 and 50.73. Additionally, the proposed change would make administrative and editorial changes in the Permanently Defueled Technical Specifications, which constitute Appendix A of the Operating (Possession Only) License.

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

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

No. San Onofre Nuclear Generating Station, Unit 1 (SONGS 1) has been permanently shut down with its reactor defueled and spent fuel from the reactor stored in the spent fuel pool. The proposed change will not modify any of the existing plant configurations, controls, procedures, or Permanently Defueled Technical Specifications (PDTS) requirements necessary to assure the integrity and safe operation of the spent fuel pool.

The requested change to License Condition 2.D will result in not requiring violations of the PDTS to be reported based on License Condition 2.D. The basis for this change is that all types of reportable events applicable to a defueled plant are covered by 10 CFR 50.72 and 50.73, which SONGS 1 is required to implement. Any other reporting requirements imposed through a license condition are redundant to reporting requirements contained in 10 CFR 50.72 and 50.73. Therefore, this change is administrative.

The requested changes to the PDTS are also administrative in nature. They consist of changes to reflect the current nuclear organization and responsibilities, modify administrative requirements relating to the Onsite Review Committee, modify a requirement relating to Final Safety Analysis Report documentation using NRC guidance, and make editorial corrections and improvements in the text. Since these changes are administrative, they have no effect on the accidents previously evaluated.

Therefore, operation of the facility in accordance with this proposed change will not involve a significant increase in the probability or consequences of an accidently previously evaluated.

2. Will operation of the facility according to this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated.

No. The proposed changes do not alter the design, configuration, or method of operation of the plant. The changes to License Condition 2.D and the PDTS are administrative or editorial.

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

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

No. The proposed changes do not alter the design, configuration, or method of operation of the plant. Since the proposed changes are administrative or editorial, the existing plant safety margins are not reduced.

Therefore, operation of the facility in accordance with this proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis of the licensee 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: Main Library, University of California, P.O. Box 19557, Irvine, California 92713

Attorney for licensee: James A. Beoletto, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770

NRC Project Director: Seymour H. Weiss

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 amendment requests: May 29, 1996

Description of amendment requests: The licensee proposes to revise improved Technical Specification (TS) 3.5.1, ''Safety Injection Tanks (SITs),'' to increase the minimum boron concentration in the safety injection tanks from 1850 parts per million (ppm) to 2200 ppm. This TS change is being requested to support the planned increase in the operating cycle length.

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.

Southern California Edison (Edison) is increasing the minimum boron concentration to maintain the ability of the Safety Injection Tanks (SITs) to perform their intended safety function consistent with the increase in fuel enrichment up to 4.8 weight percent (w/o) Uranium-235 and changing the burnable poison from B_4C to Erbia (Erbium-Oxide Er_2O_3 and fuel mixture) to increase the length of the operating cycle. Increasing the minimum boron concentration in the SITs will maintain the ability of the Emergency Core Cooling System (ECCS) to control core reactivity during and following an accident.

No change is being made to the design of the safety injection system. Consequently, there will be no impact on the probability of initiating an accident which has been previously evaluated.

Increasing the boron concentration in the SITs will ensure the ability of this system to mitigate the accidents for which it is required. No other accident conditions, design conditions, Technical Specifications, or Technical Specification Bases are affected by this proposed change in boron concentration.

Therefore, the operation of the facility in accordance with this proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

There is no change in plant design or operational methodology imposed by the increase in SIT boron concentration. This increase in boron concentration is required because Edison is increasing the fuel enrichment up to 4.8 w/o Uranium-235 and changing the burnable poison from B_4C to Erbia to achieve a longer cycle length. Therefore, additional negative reactivity is required at the beginning of the fuel cycle for these alternate coolant sources.

Edison believes this change in the SIT minimum boron concentration limit is, in essence, an administrative change. The SITs are filled from the refueling water storage tank (RWST), which has a technical specification minimum boron concentration requirement of 2350 ppm. Edison maintains the RWST boron concentration higher than the minimum limit. As a result, for the past several years the SIT boron concentration has been approximately 2500 ppm, even though the technical specification lower limit is 1850 ppm. The maximum boron concentration limit is not being changed. Increasing the SIT minimum boron concentration limit of the technical specification narrows the existing operating band, and maintaining the boron concentration between 2200 ppm and 2800 ppm will keep the boron concentration between the current band of 1850 ppm to 2800 ppm. Therefore, changing the SIT minimum boron concentration from 1850 ppm to 2200 ppm does not involve a physical change to the plant.

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

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

With the increase in fuel enrichment up to 4.8 w/o Uranium-235 and changing the burnable poison from B_4C to Erbia to increase the length of the operating cycle, increasing the minimum boron concentration in the SITs is required to maintain the current margins of safety.

The calculations were performed to ensure the core remains subcritical (i.e., conservatively 1% shutdown) with the proposed boron concentration. In addition to the conservative assumptions used in the calculation, 50 ppm was added to the results.

Therefore, the operation of the facility in accordance with this proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

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

Date of amendments request: June 12, 1996

Description of amendments request: The proposed amendments would revise the reactor core safety limits, Overtemperature delta T (OTDT) and Overpressure delta T (OPDT) reactor trip setpoints and allowable values, and the power distribution limits associated with implementation of Relaxed Axial Offset Control (RAOC) and F_Q surveillance. The proposed amendments also include changes to the Bases associated with these specifications and surveillances.

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 safety limits, reactor trip setpoints, HNF [high neutron flux] setpoints for MSSVs [main steamline safety valves] out of service, F[delta]H for LOPAR [low parasitic], and RAOC strategy changes do not increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The core safety limits and trip setpoints were determined using the NRC reviewed and approved DNB [departure from nucleate boiling] methodologies, namely RTDP, and approved DNB correlations. No new performance requirements are being imposed on any system or component in order to support the revised core limits. Overall plant integrity is not reduced. The DNB sensitive transients that are protected by [OPDT] and [OTDT] were reanalyzed or evaluated. The DNB design criterion continues to be met. None of these changes directly initiate an accident; therefore, the probability of an accident has not increased. No new performance requirements are imposed on any safety-related equipment. The acceptance criteria for the reanalyses continue to be met; therefore, the consequences of accidents previously evaluated in the FSAR are not significantly changed. All dose consequences have been evaluated for these changes and all acceptance limits continue to be met. All safety analyses that use the revised [OTDT] and [OPDT] setpoints continue to meet all acceptance criteria. [Loss-of-coolant accident] LOCA analyses are not affected by any of these proposed changes.

2. The proposed Technical Specifications changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed Technical Specifications changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. The DNB design criterion continues to be met. The use of the revised core limits, reactor trip setpoints and RAOC have been shown to allow FNP [Farley Nuclear Plant] to operate in a safe configuration. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed Technical Specifications changes do not involve a significant reduction in a margin of safety. All accident analysis acceptance criteria continue to be met. The DNB design criterion remains unchanged. The DNBR [departure from nucleate boiling ratio] design limit values have not changed. Therefore, the DNB design limit values associated with the DNB methodology and correlations, upon which the Technical Specifications changes are based, do not result in a significant reduction in the margin of safety because the DNB design criterion continues to be met. The proposed revisions to the Technical Specifications result in an operating configuration consistent with the analytic assumptions (including LOCA analyses) used to form the bases of the Technical Specifications.

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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

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

Date of amendments request: June 20, 1996

Description of amendments request: The proposed amendments would revise the Technical Specifications (TS) to incorporate the requirements of 10 CFR Part 50, Appendix J, Option B. The Administrative Controls portion would be revised to establish and reference a ''Containment Leakage Rate Testing Program'' in accordance with the NRC's Regulatory Guide 1.163 dated September 1995.

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

1. The proposed changes do not involve a significant increase in the probability of consequences of an accident previously evaluated. The proposed changes provide a mechanism within the TS for implementing a performance-based leakage rate test program which was promulgated by the revision to 10 CFR [Part] 50 to incorporate Option B to Appendix J. The proposed changes do not involve any physical or operational changes to structures, systems or components. The proposed TS Limiting Conditions for Operation (LCO) are consistent with 10 CFR [Part] 50, Appendix J requirements and are equivalent to the current LCO requirements. The current safety analyses and safety design basis for the accident mitigation functions of the containment, the airlocks, and the containment isolation valves are maintained. Since the allowable containment leakage is still maintained within the analyzed limit assumed in the accident analyses, there is no adverse effect on either onsite or offsite dose consequences. Furthermore, containment leakage is not an accident initiator. Therefore, these changes will not increase the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed. The proposed changes do not involve any physical or operational changes to structures, systems or components. No new failure mechanisms beyond those already considered in the current plant safety analyses are introduced. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. The proposed changes do not involve a significant reduction in the margin of safety. Extending Type A, B, and C test intervals from those currently provided in the TS to those provided for in 10 CFR [Part] 50 Appendix J, Option B slightly increases risk due to an increased likelihood of containment leakage corresponding to the increased testing intervals. However, this is somewhat compensated by the corresponding risk reduction benefits received from the reduction in component cycling, stress, and wear associated with the increased intervals. When considering the total integrated risk, which includes all analyzed accident sequences, the additional risk associated with increasing test intervals is negligible.

The NRC letter to NEI [Nuclear Energy Institute] dated November 2, 1995, recognizes that changes similar to the proposed changes at FNP [Farley Nuclear Plant] are required to implement Option B of 10 CFR [Part] 50, Appendix J. In NUREG-1493, ''Performance-Based Containment Leak-Test Program,'' dated September 1995, which forms the basis for the Appendix J revision, the NRC concludes that adoption of performance-based test intervals for Appendix J testing will not significantly reduce the margin of safety. The containment leak rate data and component performance history at FNP are consistent with the conclusions reached in NUREG-1493 and NEI 94-01. Thus, the proposed license amendments do not involve a significant reduction in a margin of safety and will continue to support the regulatory goal of ensuring an essentially leak-tight containment boundary.

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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: May 28, 1996

Description of amendment request: The proposed amendment would increase the test interval for Technical Specification (TS) 3/4.3.1.1, Reactor Protection System Instrumentation from monthly on a staggered test basis to semiannually on a staggered test basis for the control rod drive trip breakers and the reactor trip module logic. Additionally, the proposed amendment would increase the test interval from monthly to semiannually for the output logic of the anticipatory reactor trip system (ARTS) instrumentation as specified in TS 3/4.3.2.3.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below: (1)

Operation of the DBNPS in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Increasing the surveillance interval will not affect the probability or consequences of an accident previously evaluated since performance of the surveillance test only ensures operability of the particular trip function at the time of the test. The licensee evaluated the maintenance history and surveillance test results of the control rod drive trip breakers, reactor trip module logic, and ARTS output logic to show these components have consistently met their design and operational requirements over the past 8 years.

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

The proposed changes do not modify or affect system design, function, operation, or manner of testing.

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

The licensee has performed a reliability evaluation that indicates insignificant change in reactor trip system unavailability and a reduction in the potential for spurious trips resulting from testing which support the conclusion that a significant reduction in a margin of safety will not occur.

Based on the NRC staff review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 28, 1996

Description of amendment request: The proposed amendment would revise the Technical Specifications for shutdown margin to allow calculational determination of the highest worth control rod. Editorial changes are also included.

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) During refueling, maintenance may be performed on either the control rods or the control rod drive mechanisms. Controls, such as refueling interlocks, are provided to assure inadvertent criticality does not occur during this maintenance. There are no proposed revisions to these controls except to lower the threshold for applicability, which constitutes a more restrictive change.

These controls also continue to assure that the new, higher minimum shutdown margin is maintained to ensure the reactor can be returned to a subcritical condition should an inadvertent criticality occur. The proposed alternate calculational method for highest worth control rod has additional conservatism to account for any uncertainties in the calculation and provides equivalent margin. Therefore, this change will not significantly increase the probability or consequences of any previously analyzed accident.

(2) The proposed change does not necessitate a physical alteration of the plant in that no new or different type of equipment will be installed. The proposed change does propose a higher minimum shutdown margin and a lower threshold of applicability for CRD [control rod drive] maintenance, both of which are more restrictive. The proposed change will provide effective methods to preserve the safety functions associated with the prevention or automatic mitigation of design basis accidents. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed changes to the controls provided to allow control rod withdrawal for the purposes of maintenance are more restrictive and thus preserve the safety functions associated with the prevention or automatic mitigation of design basis accidents. The addition of a higher minimum shutdown margin requirement and the proposed calculational alternative for highest worth rod, does not decrease any of the safety controls or functions to prevent inadvertent criticalities and provides equivalent or higher margins. Therefore, this change will not significantly reduce 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301

Attorney for licensee: R. K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110-2624

NRC Project Director: Jocelyn A. Mitchell, Acting Directorboro, VT 05301

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

Date of amendment request: July 3, 1996

Description of amendment request: The proposed amendment would modify Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) Section 4.2.b, ''Steam Generator Tubes,'' to: revise the plugging criteria for tubes in the tubesheet crevice region; add new inspection criteria for tubes evaluated using the new plugging criteria; add definitions of terms used in the new plugging criteria; and add reporting requirements.

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

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

The revised plugging criteria ensure that tubes in the tubesheet with indication(s) are sufficiently inspected and evaluated and, if necessary, rolled to meet the proposed acceptance criteria based on the new definitions of acceptable distance between the indication and the rolled area. With sufficient distance between the indication(s) and the hard rolled region of the tube in the tubesheet, tube rupture probability and the consequences of tube rupture are the same as previously analyzed. Additionally, the potential for leakage is within previously analyzed limits.

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

Implementation of the proposed tube plugging criteria and proposed inspection acceptance criteria based on the proposed definitions does not introduce any significant changes to the plant design basis. Use of these criteria will not introduce a mechanism that will result in an accident initiated outside of the tubesheet crevice region. Any hypothetical accident as a result of tube indications in the tubesheet crevice region of the tube will be bounded by the existing tube rupture analysis. Therefore, application of the revised acceptance criteria for indication(s) within the tubesheet crevice region will not create the possibility of a new or different kind of accident.

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

The use of the proposed inspection criteria and tube plugging acceptance criteria will maintain the integrity of the tube bundle commensurate with the requirements of Regulatory Guide 1.121 under normal and postulated accident conditions. The safety factors used in verification of the strength of tube(s) evaluated under the new plugging criteria are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used for steam generator design. The leak testing acceptance criteria are based on the primary-to-secondary leakage limits in the TSs and the Updated Safety Analysis Report accident analyses will be maintained. Therefore, the proposed TS change will not result in a significant reduction in the margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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 NoSignificant 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.

Northeast Utilities Service Company, Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London, Connecticut

Date of amendment request: July 3, 1996Brief

Description of amendment request: The proposed amendments would provide a one-time change to Technical Specification 3.9.1, ''Refueling Operations, Boron Concentration.'' The proposed change would remove the requirement that the boron concentration in all filled portions of the Reactor Coolant System be ''uniform.''

Date of publication of individual notice in Federal Register: July 11, 1996 (61 FR 36583)

Expiration date of individual notice: August 12, 1996

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

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 amendment: June 3, 1996, as superseded by application dated June 25, 1996

Brief description of amendment request: The proposed amendment would revise Technical Specifications 3.3.11, ''Post Accident Monitoring Instrumentation,'' and 5.5.2.13, ''Diesel Fuel Oil Testing Program.'' The amendment would reinstate provisions of the current San Onofre Nuclear Generating Station, Unit Nos. 2 and 3 technical specifications that were revised as part of Amendment Nos. 127 and 116. These amendments adopted the recommendations of NUREG-1432, ''Standard Technical Specifications Combustion Engineering Plants.''

Date of individual notice in Federal Register: July 2, 1996 (61 FR 34452)

Expiration date of individual notice: August 1, 1996

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

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

Date of application for amendment: April 25, 1995

Brief description of amendment request: The proposed amendment would add a reactor water cleanup system high blowdown containment isolation trip function and associated limiting condition for operation and surveillance requirements.

Date of individual notice in Federal Register: June 28, 1996 (61 FR 33777)

Expiration date of individual notice: July 29, 1996

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

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

Date of application for amendment: June 6, 1995, as supplemented by letter dated April 22, 1996.

Brief description of amendment request: The proposed amendment would make administrative and editorial changes to Section 6.0 of the technical specifications for WNP-2.Date of individual notice in Federal Register: June 28, 1996 (61 FR 33779)

Expiration date of individual notice: July 29, 1996

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

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: March 29, 1996.

Brief description of amendment: The amendment revises the technical specifications (TS) to add an allowance to complete a TS-required surveillance within 24 hours of discovery of a missed surveillance in accordance with the guidance of Generic Letter (GL) 87-09, ''Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements.''

Date of issuance: July 8, 1996

Effective date: July 8, 1996

Amendment No. 170

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

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25669) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 8, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: November 2, 1994

Brief description of amendments: The amendments delete the content of Appendix B, ''Environmental Protection Plan (EPP) (Nonradiological),'' and modify License Condition 2.C.(2) to delete that portion which refers to the EPP.

Date of issuance: July 8, 1996

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

Amendment Nos.: 149 and 143

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Environmental Protection Plan and License Conditions.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25702) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 8, 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

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

Date of application for amendment: April 18, 1996

Brief description of amendment: The amendment deleted a restriction on the 24-hour emergency diesel generator operation test in Surveillance Requirement 3.8.1.14 of the Technical Specifications for the Grand Gulf Nuclear Station, Unit 1. The deletion allows the test to also be conducted during power operation (i.e., during Modes 1 and 2), instead of the current requirement to only conduct the test when the plant is shut down.

Date of issuance: July 15, 1996

Effective date: July 15, 1996

Amendment No: 124

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20847) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 15, 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.

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: May 6, 1996

Brief description of amendment: The amendment reflects that the name of Mississippi Power & Light Company (MP&L) has been changed to Entergy Mississippi, Inc. The amendment revises Operating License No. NPF-29 and the Antitrust Conditions for the Grand Gulf Nuclear Station, Unit 1 (GGNS) to (1) add the phrase ''(now renamed Entergy Mississippi, Inc.)'', (2) replace the name of Mississippi Power & Light Company (MP&L) by the name Entergy Mississippi, Inc., and (3) replace a footnote by the statement: ''Amendment 125 resulted in a name change for Mississippi Power & Light Company (MP&L) to Entergy Mississippi, Inc.''.

Date of issuance: July 16, 1996

Effective date: July 16, 1996

Amendment No: 125

Facility Operating License No. NPF-29. Amendment revises the operating license.

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28613) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 16, 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, et al., Docket No. 50-335 St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendments: June 1, 1996

Brief description of amendments: Revise Technical Specifications to reflect reduced reactor coolant system flows resulting from increased percentage of plugged steam generator tubes.

Date of Issuance: July 9, 1996

Effective Date: July 9, 1996

Amendment Nos.: 145

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

Date of initial notice in Federal Register: June 7, 1996 (61FR29140). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 9, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

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: July 26, 1995

Brief description of amendments: The amendments modify the Technical Specifications to allow operation with up to plus or minus 18 steps of rod misalignment at or below 90 percent power.

Date of issuance: July 12, 1996

Effective date: July 12, 1996

Amendment Nos. 186 and 180Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60FR47616) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 12, 1996.No significant hazards consideration comments received: No

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

GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: May 7, 1996 (TSCR 247)

Brief description of amendment: The amendment adopts the provisions of the Standard Technical Specifications, NUREG-1433, Rev. 1 which clarify surveillance requirement applicability and allow a maximum period of 24 hours to complete a surveillance requirement upon discovery that the surveillance has been missed.

Date of Issuance: July 15, 1996

Effective date: July 15, 1996

Amendment No.: 185

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

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28615). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 15, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

PECO Energy Company, Public Service Electric and Gas Company Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: December 21, 1995

Brief description of amendments: The amendments modify the Peach Bottom Atomic Power Station Units 2 and 3 Facility Operating Licenses to provide for elimination of outdated or superseded material regarding, among other things, environmental monitoring and modifications to the low pressure coolant injection system, and for making the FOLs for both units consistent.

Date of issuance: July 15, 1996

Effective date: Units 2 and 3, as of the date of issuance, to be implemented within 30 days.

Amendments Nos.: 215 and 220

Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Facility Operating Licenses.

Date of initial notice in Federal Register: March 13, 1996 (61 FR 10396) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 15, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Public Service Electric & Gas Company, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of application for amendment: May 7, 1996, as supplemented June 14, 1996

Brief description of amendment: The amendment made a one-time change to Technical Specification 3/4.7.6, ''Control Room Emergency Air Conditioning System,'' which permits refueling of Unit 2 with the Control Room Emergency Air Conditioning System (CREACS) inoperable in Modes 5 and 6. This change will expire after the completion of the Control Room and CREACS upgrade, currently in progress, and the restart and entry into Mode 4 of Unit 2 from the current outage.

Date of issuance: July 10, 1996

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

Amendment No. 165

Facility Operating License No. DPR-75: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25710) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 10, 1996.No significant hazards consideration comments received: No

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

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 application for amendments: April 22, 1996, as supplemented June 12, 1996

Brief description of amendments: The amendments change the Technical Specifications to implement 10 CFR Part 50, Appendix J, Option B, for the Type A test by referring to Regulatory Guide 1.163, ''Performance Based Containment Leakage-Test Program.''

Date of issuance: July 11, 1996

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

Amendment Nos. 184 and 166

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20856) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 11, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: June 24, 1996

Brief description of amendments: The amendments approve a unit cycle specific (Unit 1, Cycle 14 and Unit 2, Cycle 11) Technical Specification change to Note 4 of Table 4.3-1 that permits continued operation of both Farley units without performing the required surveillance of the manual safety injection input to the reactor trip circuitry for the current operating cycle until the next unit shutdown, following which, this testing has to be performed prior to entering Mode 2.

Date of issuance: July 19, 1996

Effective date: July 19, 1996

Amendment Nos.: 120 and 112

Facility Operating License Nos. NPF-2 and NPF-8: The amendments revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: Yes. (61 FR 34880 dated July 3, 1996). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by August 2, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated July 19, 1996.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, P.O. Box 1369, Dothan, Alabama

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

Date of application for amendment: January 2, 1996, as supplemented by letter dated April 12, 1996.

Brief description of amendment: The amendment would revise TS 3.9.4 and its associated Bases to allow the containment personnel airlock doors to be open during core alterations and movement of irradiated fuel in containment.

Date of issuance: July 15, 1996

Effective date: July 15, 1996, to be implemented within 30 days of the date of issuance.

Amendment No.: 114

Facility Operating License No. NPF-30: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5819). The April 12, 1996, supplemental letter provided clarifying information and did not change the original no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 15, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

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

Date of application for amendment: April 4, 1996

Brief description of amendment: The amendment revises the Technical Specifications regarding secondary containment integrity including addition of required actions in the event secondary containment integrity is not maintained when required. It also requires surveillance of the secondary containment isolation valves under the licensee's in-service testing program.

Date of issuance: July 10, 1996

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

Amendment No.: 147

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20859) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 10, 1996No significant hazards consideration comments received: No

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

Dated at Rockville, Maryland, this 24th day of July 1996.

For the Nuclear Regulatory Commission

Steven A. Varga, Director,

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

[Doc. 96-19317 Filed 7-30-95; 8:45 am]

BILLING CODE 7590-01-F

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–22097; File No. 812–9992]

Continental Assurance Company, et al.

July 25, 1996.

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

ACTION: Notice of application for exemptions under the Investment Company Act of 1940 (''1940 Act'').

APPLICANTS: Continental Assurance Company (''CAC''), Valley Forge Life Insurance Company (''VFL,'' together with CAC, the ''Companies''), Continental Assurance Company Variable Life Separate Account (''CAC Account''), Valley Forge Life Insurance Company Variable Life Separate Account (''VFL Account''), and CNA Investor Services, Inc.

RELEVANT 1940 ACT SECTIONS: Sections 6(c), 27(a)(3), 27(c)(2), and 27(e), and Rules 6e–3(T)(b)(13)(ii), 6e–3(T)(b)(13)(vii), 6e–3(T)(c)(4)(v), and 27e–1 thereunder.

SUMMARY OF APPLICATION: Applicants seek an order to the extent necessary to permit them or any other variable life insurance separate account established in the future by the Companies (''Future Accounts,'' collectively with the CAC Account and the VFL Account, the ''Accounts'') to support certain flexible premium variable life insurance policies offered currently or in the future through the Accounts (collectively, ''Policies'') to: (1) deduct from premium payments received under the Policies a charge that is reasonable in relation to each Company's increased federal tax burden related to the receipt of such premium payments that results from the application of Section 848 of the Internal Revenue Code of 1986, as amended, (''Code''); (2) deduct sales charges from premium payments received in connection with Policies in a manner that results, in some instances, in sales charges on subsequent premium payments exceeding sales charges on prior premium payments; (3) compute sales surrender charges on such premium payments in a manner that results, in some instances, in sales surrender charges on subsequent premium payments exceeding sales surrender charges on prior premium payments; and (4) refrain from sending owners of Policies a written notice of certain refund and withdrawal rights.

FILING DATE: The application was filed on February 14, 1996.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the SEC 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 August 16, 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 writer's interest, the reason for the request, and the issues contested.