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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 15, 1995, through May 25, 1995. The last biweekly notice was published on Tuesday, May 23, 1995 (60 FR 27334).

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 Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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 July 7, 1995, 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, 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, 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.

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

Date of amendment requests: March 24, 1995.

Description of amendment requests: The proposed amendments would make numerous changes to Technical Specification (TS) 3/4.8.1, ''A.C. Sources,'' and the associated TS Bases, for Palo Verde Units 1, 2, and 3. The proposed amendments would implement recommended changes from NUREG–1432, ''Standard Technical Specifications: Combustion Engineering Plants''; Generic Letter (GL) 94–01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators''; and GL 93–05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' The proposed changes are intended to increase emergency diesel generator (EDG) reliability by reducing the stresses on the EDGs from unnecessary testing. Additional changes have also been proposed to TS 3/4.8.1 to further enhance EDG reliability, to achieve consistency with NUREG–1432, Combustion Engineering Standard TS, and to improve the TS presentation.

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 changes to TS 3/4.8.1 and the associated Bases affect the required actions in response to inoperable offsite and onsite AC sources, surveillance requirements for the EDG, and reporting requirements for EDG failures. The majority of the proposed changes are based on the recommendations of NUREG 1432, GL 94–01, and GL 93–05. These proposed changes have been extensively reviewed by the NRC during the preparation of these documents, and by APS during the development of this request for TS amendment. The proposed changes are expected to result in improvements in EDG performance and reduce EDG aging due to excessive testing. The proposed changes will permit the elimination of the unnecessary mechanical stress and wear on the EDGs while ensuring that the EDGs will perform their design function. The elimination of mechanical stress and wear will improve reliability and availability of the EDGs which will have a positive effect on the ability of the EDGs to perform their design function. The proposed changes to [do] not affect the availability or the testing requirements of the offsite circuits.

Because the proposed changes do not affect the design or performance of the EDGs or their ability to perform their design function, the changes are expected to result in a decrease in the probability or consequences of an accident previously evaluated. The proposed changes will increase EDG reliability, thereby increasing overall plant safety. Because these changes do not affect the probability of accident precursors (EDGs do not initiate any accidents), the proposed type license amendment does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed changes to TS 3/4.8.1 and the associated Bases do not introduce any new modes of plant operation or new accident precursors, involve any physical alterations to plant configurations, or make any changes to system setpoints which could initiate a new or different kind of accident. The proposed changes do not affect the design or performance characteristics of any EDG or its ability to perform its design function. No new failure modes have been defined nor new system interactions introduced for any plant system or component, nor has any new limiting failure been identified as a result of the proposed changes. The proposed changes will eliminate unnecessary EDG testing, increasing EDG reliability and availability, and thereby having an overall positive affect on plant safety. Accidents concerning loss of offsite power and a single failure (e.g., loss of an EDG) have previously been evaluated. These changes are intended to improve plant safety, decrease equipment degradation, and remove unnecessary burden on personnel resources by reducing the amount of testing that the TS requires during power operation. Therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Under the proposed changes to TS 3/4.8.1 and the associated Bases, the EDGs will remain capable of performing their safety function. The changes do not affect the design or performance of any EDG, but will increase EDG reliability and availability by reducing the stresses and the effects of aging on the EDG by eliminating unnecessary testing. This will result in an overall increase in plant safety. Since the ability of the EDGs to perform their safety function will not be degraded, the proposed license amendment does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on that 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: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

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

NRC Project Director: William H. Bateman.

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

Date of amendment requests: March 31, 1995.

Description of amendment requests: The proposed amendment would clarify the shutdown margin definition, change the shutdown margin applicability and surveillance requirements to comply with safety analysis assumptions for subcritical inadvertent control element assembly withdrawal (UFSAR Section 15.4), and expand the applicability for core protection calculator (CPC) operability. In addition, the proposed amendment would add a reference to the Core Operating Limits Report (COLR) for the MODE 6 refueling boron concentration limit. The proposed amendment would also change the power calibration requirements for the linear power level, the CPC delta T power, and CPC nuclear power signals to allow more conservative settings than presently required.

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

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

These changes are being made to ensure compliance with the safety analysis assumptions for subcritical inadvertent CEA [control element assembly] withdrawal. These changes also ensure that the boron concentration in the reactor is sufficient to prevent criticality if an inadvertent withdrawal of a shutdown CEA bank were to occur with all other CEAs inserted. Therefore, the consequences of the inadvertent CEA withdrawal is no greater than those of the event previously evaluated. This change also has no affect on the probability of an accident since it is not introducing or changing any accident initiating mechanism.

The analysis of uncontrolled CEA withdrawal from MODES 2 and 3 subcritical with four RCPs [reactor coolant pumps] running is presented in UFSAR Section 15.4.1 as an anticipated operational occurrence. The consequences of this event are that the acceptable fuel design limits are not exceeded (General Design Criterion 25 as specified in the NRC Standard Review Plan). The proposed change to TS requiring that either the CPCs or Logarithmic Power Level—High trip (trip setpoint lowered to 10^{-4}% of Rated Thermal Power) are Operable in MODES 3, 4, and 5, ensures that an inadvertent CEA withdrawal with less than four pumps operating, results in consequences no greater than those of the previously evaluated uncontrolled CEA withdrawal event.

The revised TS will also ensure that the reactivity worth of any full-length CEAs not capable of being inserted is accounted for in the determination of the shutdown margin. This change will ensure the shutdown margin will continue to be within safety analysis assumptions for previously evaluated accidents.

The proposed changes to TS, replacing the MODE 6 boron concentration specification with the requirement to maintain the boron concentration within the limit specified in the COLR, will not affect the probability or consequences of an accident, because it is not changing the MODE 6 reactivity requirement of K_{eff} less than or equal to 0.95, but provides a specific boron concentration value in the COLR to ensure the MODE 6 required K_{eff} value of less than or equal to 0.95 is met.

The proposed changes will reduce the amount of non-conservatism presently allowed for the linear power level, the CPC delta T power and CPC nuclear power signals. Changing the tolerance range from plus or minus 2% to between −0.5% and 10% between 15% and 80% RATED THERMAL POWER, except during initial post refueling power ascension and restricting recalibration, will allow more conservative settings than currently required.

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

The changes revising the mode applicabilities are being made to comply with safety analysis assumptions for subcritical CEA withdrawal. The SR [surveillance requirement] ensures that the shutdown margin is within the safety analysis assumptions when the reactor trip breakers are open and any full-length CEA is not fully inserted. No new or different kind of accident will be initiated since this change will ensure that the required shutdown margin is maintained when the reactor trip breakers are closed.

The proposed change to TS, requiring either the CPCs or Logarithmic Power Level—High trip to be operable, will provide protection from inadvertent CEA withdrawal when less than four RCPs are operating. No new or different kind of accident will be initiated by this change, since this change incorporates TS limitations to ensure protection for an existing accident scenario.

The revised TS shutdown margin definition ensures that the reactivity worth of any full-length CEAs not capable of being inserted is accounted for in the determination of the shutdown margin. This ensures the shutdown margin will continue to be within safety analysis assumptions. Maintaining the shutdown margin within the safety analyses assumption will not create any new or different kind of accident.

The proposed changes to TS power calibration tolerance limits are conservative relative to the current TS requirements and therefore will not create any new or different kind of accident.

The proposed change to TSs replacing the MODE 6 boron concentration specification with the requirement to maintain the boron concentration within the limit specified in the COLR does not create the possibility of a new or different kind of accident from any accident previously analyzed. The proposed change is not changing the MODE 6 reactivity requirements of less than or equal to 0.95 while providing a specific boron concentration value in the COLR to ensure the MODE 6 required K_{eff} value of less than or equal to 0.95.

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

The proposed change to TS adds an additional requirement for the CPCs or Logarithmic Power Level—High trip to be operable in MODES 3, 4, and 5. This change maintains the margin of safety in the safety analysis by providing a TS that will ensure appropriate protection is provided in the event of an inadvertent CEA withdrawal with less than four RCPs operating.

The proposed changes to TS (Boration Control, Shutdown Margin), revising the mode applicabilities, maintains the margin of safety provided in the TS by ensuring that the safety analysis assumptions for subcritical CEA withdrawal are met. The new SR does not reduce the margin of safety since the shutdown margin assumed in the safety analysis will be maintained by this TS.

The revised TS shutdown margin definition ensures that the reactivity worth of any full length CEAs not capable of being inserted is accounted for in the determination of shutdown margin. This ensures shutdown margin will continue to be within safety analysis assumptions. This change maintains the margin of safety that is currently provided by TS.

The proposed changes to TS, reducing the amount of non-conservatism in the safety system power indications, maintains the margin of safety for design basis events which take credit for the linear power level, the CPC delta T power, and CPC nuclear power signals.

The proposed change to TS moves the specific MODE 6 boron concentration value to COLR. The proposed change does not change the MODE 6 reactivity requirement of K_{eff} of less than or equal to 0.95, but provides a specific boron concentration value in the COLR to ensure the MODE 6 required K_{eff} value of less than or equal to 0.95 is met. Therefore, the margin of safety is not affected by the proposed change.

The NRC staff has reviewed the licensees' analysis and, based on that 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: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

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

NRC Project Director: William H. Bateman.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: December 8, 1992, as supplemented on September 10, 1993, and May 17, 1995.

Description of amendment request: As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom Technical Specifications (TS) used.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG–0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both Dresden and Quad Cities TS to the STS contained in NUREG–0123.

The TSUP for Dresden and Quad Cities is not a complete adaptation of the STS. The TSUP focuses on (1) Integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GLs), and (4) relocating specific items to more appropriate TS locations.

The December 8, 1992, application, as supplemented on September 10, 1993, and May 17, 1995, proposed to upgrade only Section 3/4.1 (Reactor Protection System) of the Dresden and Quad Cities TS.

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

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

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes to the current Technical Specifications (CTS) represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.1 are based on BWR–STS (NUREG–0123, Revision 4 ''Standard Technical Specifications General Electric Plants BWR/4) guidance or NRC accepted changes at later operating BWR plants. Any deviations from BWR–STS and CTS requirements do not significantly increase the probability or consequences of any previously evaluated accident for Dresden and Quad Cities Station. These proposed changes are consistent with the current safety analyses and have been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems that make up the Reactor Protection System are not assumed in any safety analysis to initiate any accident sequence for both Dresden and Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. These more prescriptive surveillance requirements increase the probability that the Reactor Protection System will perform its intended function. Therefore, the proposed TS will improve the reliability and availability of all affected systems and reduce the consequences of any accident previously evaluated.

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

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these changes provide for additional restrictions which are in accordance with the current safety analyses, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.1 is based on BWR–STS guidelines or NRC accepted changes at later operating BWR plants. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the BWR–STS or later operating BWRs. Any deviations from BWR–STS or CTS requirements do not create the possibility of a new or different kind of accident than previously evaluated for Dresden and Quad Cities Stations. No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications or adopt requirements that have been used at other operating BWRs with design similar to Dresden and Quad Cities. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that make up the Reactor Protection System are not assumed in any safety analysis to initiate any accident sequence for Dresden and Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the Reactor Protection System are generally more prescriptive than the current requirements specified within the Technical Specifications; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the later individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain within their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 3/4.1 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the BWR–STS. Any deviations from BWR–STS and CTS requirements do not significantly reduce the margin of safety for Dresden and Quad Cities Stations. The proposed changes are intended to improve reliability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found acceptable for use at Dresden and Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and Quad Cities and maintain necessary levels of system or component readability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the Reactor Protection System when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: July 29, 1993.

Description of amendment request: The proposed amendment would extend the instrument calibration intervals for selected plant instrumentation from 18 months to 36 months.

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 to extend to 36 months the calibration interval of selected instrumentation does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The purpose of the proposed Technical Specification change is to extend calibration interval testing requirements for selected instrumentation. However, because of the continued application of redundant Technical Specification requirements such as channel checks, channel functional tests, and logic system functional tests, the performance of these instruments will be maintained within the acceptance limits assumed in plant safety analyses and required for the successful mitigation of an initiating event. The proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended function within their instrument settings.

These other tests are sufficient to identify failure modes or degradations in instrument performance and ensure operation of the associated systems within acceptance limits. There are no credible failure modes that can be detected by instrument calibration that cannot also be detected by other Technical Specification tests.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. As discussed above, the proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended function within the acceptance limits assumed in plant safety analyses and required for successful mitigation of an initiating event. All plant systems continue to operate in an identical manner. No new accident modes are created.

(3) Involve a significant reduction in a margin of safety. The current Technical Specification allowable values are based on the maximum analytical limits assumed in the plant safety analyses. These analyses conservatively establish the margin of safety. The proposed Technical Specification changes do not affect the capability of the associated systems to perform their function within the instrument settings used as the basis for the plant safety analyses. Plant and system settings to an initiating events will remain in compliance within the assumptions of the safety analyses, and therefore the margin of safety is not affected.

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: Cynthia A. Carpenter, Acting.

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

Date of amendment request: December 15, 1994.

Description of amendment request: The proposed amendment would relocate, revise, or delete various Technical Specification (TS) provisions. Administrative controls on working hours in TS 6.2.2.f, the Independent Safety Engineering Group requirements in TS 6.2.3, the unit staff qualification requirements in TS 6.3, the reportable event requirement for the Onsite Review Organization (OSRO) in TS 6.6.1.b, the radiation protection program requirements in TS 6.11, the record retention requirements in TS 6.10, and the review and audit functions in TS 6.5 (with the exception of TS 6.5.2.8), would be relocated to Chapter 13 of the Updated Final Safety Analysis Report (UFSAR). The review and approval process for temporary changes to each TS 6.8.1 plant procedure listed in TS 6.8.4 would also be relocated to Chapter 13 of the UFSAR.

The requirements of TS 6.5.2.8, the review and approval process for administrative procedures in TS 6.8.2, and the review and approval process for plant procedures in TS 6.8.3, would be relocated to the Fermi 2 Quality Assurance program. The in-plant radiation monitoring program requirements in TS 6.8.5.b, and the high radiation area requirements in TS 6.12 would be relocated to Chapter 12 of the

UFSAR. The radiological environmental monitoring program requirements in TS 6.8.5.f would be relocated to Chapter 11 of the UFSAR. The Process Control Program (PCP) requirements in TS 6.13 would be relocated to the PCP.

The requirements for OSRO to review the Security Plan in TS 6.5.1.6.j and to have Security Plan implementing procedures in TS 6.8.1.e would be relocated to the Fermi 2 Security Plan. The requirements for OSRO to review the Emergency Plan in TS 6.5.1.6.k and to have Emergency Plan implementing procedures in TS 6.8.1.f would be relocated to the Fermi 2 Emergency Plan.

The unit staff qualification requirements, as specified in the H. R. Denton (NRC) letter of March 29, 1980, in TS 6.3, would be deleted. The licensee states these have been superseded by 10 CFR Part 55 and Generic Letter (GL) 87–07. The training requirements in TS 6.4 would be deleted. The licensee states that other Section 6.0 TS and NRC regulations provide sufficient control of these training requirements. The submittal requirement of the annual radioactive effluent release report in TS 6.9.1.8 would be revised from ''within 90 days after January 1 * * *'' to ''prior to May 1. * * *''

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

(1) The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes are administrative in nature. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR transient analyses. No Limiting Condition for Operation, ACTION statement or Surveillance Requirement is affected by any of the proposed changes. Also, these proposed changes, in themselves, do not reduce the level of qualification or training such that personnel requirements would be decreased. Therefore, this change is administrative in nature and does not involve a significant increase in the probability or consequences of an accident previously evaluated. Further, the proposed changes do not alter the design, function, or operation of any plant component and therefore, do not affect the consequences of any previously evaluated accident.

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

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

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

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

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment request: April 26, 1995.

Description of amendment request: The proposed amendment would add a requirement to Technical Specification (TS) 4.5.2.a to periodically verify that the High Head Safety Injection (HHSI) pump minimum flow valve, 2CHS*MOV373, is maintained open during plant operation in Modes 1, 2, and 3. Valve 2CHS*MOV373 must be maintained open to provide a minimum flowpath for the HHSI pumps and thereby minimize the likelihood of HHSI pump damage due to operating the pumps with insufficient flow. The proposed change would allow flexibility for local verification of valve position or flow indication if the control room indication is not available. The proposed amendment would also make several editorial changes to TS 3/4.5.2 for consistent format with other TSs.

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

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

Maintaining 2CHS*MOV373 in a de-energized locked open position ensures charging/High Head Safety Injection pump (HHSI pump) minimum flow remains available for normal operation and design basis accidents. It has been determined that with 2CHS*MOV373 in the open position there is no significant increase in radiation levels and no change to the existing environmental qualification or personnel access routes. Sufficient injection flow to the core will be maintained during events requiring a Safety Injection (SI) actuation. Potential HHSI pump damage due to low flow will be prevented during periods of high Reactor Coolant System (RCS) pressure following a steam line break and SI. It has also been determined that the HHSI pumps will remain capable of performing their safety function with a continuous minimum flow. There is no impact on analysis assumptions or radiological consequences of an accident.

There are no postulated events in the Updated Final Safety Analysis Report (UFSAR) which require that 2CHS*MOV373 be closed. Thus, the decision to de-energize and lock open the valve ensures adequate minimum flow for the HHSI pumps.

The proposed addition of 2CHS*MOV373 to Technical Specification 3.5.2 enhances the operator's ability to verify the valve position. The proposed surveillances and footnote will be used to monitor the valve position, the status of motor operator, and the valve position indicating lights. Therefore, the proposed change to the technical specification will ensure that the HHSI pump minimum flow is always available.

Several editorial changes were also made to Technical Specification 3.5.2. These changes do not alter the intent of the technical specification and as such have no impact on previously evaluated accident scenarios.

Therefore, the proposed change will not involve a significant 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?

The proposed addition of 2CHS*MOV373 to the technical specifications does not involve changes to the physical plant. The proposed change adds surveillance requirements and a footnote which monitor the valve position, the lack of power to the motor operator, and the valve position indicating lights. This assures that the minimum flow valve is open to maintain the HHSI pumps operable under all conditions.

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 proposed change provides additional action to ensure that 2CHS*MOV373 remains open and minimum HHSI pump flow remains available. Safety limits and limiting safety system settings are not affected by this proposed change. There are no changes to the offsite dose consequences resulting from this request.

Therefore, use of the proposed technical specification would 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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Attorney for licensee: Gerald Charnoff, Esquire, 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. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: July 18, 1991, as supplemented by letters dated March 16, and December 2, 1994, and March 9, 1995.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) on control Room Air Conditioning System (CRACS) by separating the current composite requirements of TS 3.7.6 into four TSs covering three separate functions; control room emergency air filtration system (two mode sets), control room air temperature, and control room isolation and pressurization. The changes also increase the allowed outage time to identify and correct breaches to the control room envelope, adds requirements for make-up air flow rate to be used in conjunction with existing differential pressure requirements, and adds toxic gas specifications for Modes 5 and 6. The amendment is related to a revision to the Technical Specification Bases approved by the NRC in a letter dated August 9, 1988. The March 16, and December 2, 1994, and March 9, 1995 submittals provided additional information and included some additional restrictions in proposed changes by original application dated July 18, 1991. The original notice was published on September 4, 1991 (56 FR 43808). The additional submittals do not change the no significant hazard consideration determination previously made by the licensee.

Basis for proposed no significant hazards consideration determination: The proposed change would create new Specifications as follows: 3/4.7.6.1 Emergency Air Filtration, Modes 1–4; 3/4.7.6.2 Emergency Air Filtration, Modes 5 and 6; 3/4.7.6.3 Control Room Air Temperature; 3/4.7.6.4 Control Room Isolation and Pressurization. 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 limiting accidents against which the CRACS protects are:

- All Chapter 15 scenarios involving a release of radiation to the environment outside the containment,
- Toxic gas releases, and
- Smoke resulting from control room envelope fires.

Limiting accidents against which the emergency air filtration system protects are all Chapter 15 scenarios involving release of radiation to the environment outside the containment.

The probability and consequences of any of the limiting accidents listed above are unchanged by the specialization of the plant TSs. As pointed out in the description of the change, TSs 3/4.7.6.1 and 3/4.7.6.2 have retained all requirements from the existing TS with the addition of one action statement based on the inoperability of both trains, and the exception of one action statement based on one inoperable train in Modes 5 or 6. This action statement is unnecessary since it is only applicable in a mode unlikely to experience the limiting design basis accidents against which this system protects. Therefore, the protection of the original specification is uncompromised for the function of emergency air filtration.

There are two differences between the existing TS and the proposed TS 3/4.7.6.3 regarding control room air temperature. The first is the three hour outage allowed when both air conditioning units are inoperable [this was withdrawn by licensee's March 9, 1995, letter].

This corrects most types of failures. Although three hours are less restrictive than TS 3.0.3, it is not significantly less and therefore, does not seriously reduce the protection of the original specification. The other change is the reduction of the surveillance temperature from 110°F to 80°F. This is more restrictive than the existing version. All other requirements for air conditioning are retained in the proposed TS.

Proposed TS 3/4.7.6.4, which concerns control room isolation and pressurization, allows more limited continued plant operation than the existing TS. When compared to existing actions required for continued operation with a known breach, the proposed specification recognizes the potential consequences that could arise from operation with an unidentified breach in the envelope and imposes more restrictive actions.

Engineering analysis also shows that, for most of the time, toxic chemical concentrations in the control room envelope after a postulated release are largely the result of in-leakage from the RAB [reactor auxiliary building] after isolation. This has the effect of reducing the chemical concentration of gas leaking into the control room by at least an order of magnitude and ultimately results in a control room chemical concentration buildup rate slower than previously assumed. These characteristics make it likely that the operators would have sufficient time to don the breathing apparatus installed in the control room. It is also noteworthy that this emergency breathing apparatus is considered by Regulatory Guide 1.78 to provide sufficient operator protection for those cases where chemical toxicity limits might be exceeded.

The limited continued operation allowed by the proposed change, the design characteristics of the control room, and the installed breathing apparatus provides a reasonable level of protection for plant personnel. Some new restrictions are identified for the control room isolation and pressurization. These were not previously identified and therefore offer enhanced protection to the TS. All existing requirements specific to the isolation and pressurization function are retained in the proposed version. As such, the proposed specification offers more protection than the existing TS.

Based on the above, these revisions to the TS will not adversely affect the reliability or performance of any installed equipment. There are no design changes associated with this proposed amendment, consequently, all aspects of the safety analysis will remain unchanged and there will be no physical change to the facility, and operation of Waterford 3 in accordance with these proposed changes will not involve a significant increase in the probability or consequence of any accident previously evaluated.

To create a new or different kind of accident, these changes must introduce a new failure path. In this regard, these revisions are benign since they do not alter the system or its operation. With a few exceptions, all existing TS restrictions have been retained. The exceptions have been shown to have insignificant impact. Furthermore, several additional restrictions, not in the existing specification, have been added.

Based on the above information, these changes do not introduce a new failure path and therefore, cannot create a new, unevaluated sequence of events. The current plant safety analyses are bounding and this revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Safety margins related to the control room envelope air systems are established for control room temperature and the habitability of the control room following all credible accidents. This change does not modify the equipment installed in the plant or its operation. Therefore, existing margins of safety are retained, and the operation of Waterford 3 in accordance with this 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.

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

Date of amendment request: May 12, 1995.

Description of amendment request: The proposed change modifies surveillance requirements associated with containment leakage Technical Specification (TS) 3.6.1.2 by removing scheduler requirement for Type A tests to be performed specifically at 40 plus or minus 10 month intervals and, instead, reference Type A testing in accordance with 10 CFR part 50, appendix J. The proposed change adopts the wording for primary containment integrated leak rate testing that is consistent with the requirements of the Combustion Engineering Improved Standard Technical Specifications (NUREG–1432). The proposed change also includes several administrative changes. The May 12, 1995, submittal superseded the November 16, 1993, submittal in its entirety. The November 16, 1993, submittal was noticed in **Federal Register** on January 5, 1994 (59 FR 619).

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 will not affect the assumptions, design parameters, or results of any accident previously evaluated. The proposed change does not add or modify any existing equipment. The proposed Type A test schedule will continue to be consistent with 10 CFR 50 Appendix J. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change does not involve modifications to any existing equipment. The proposed change will not affect the operation of the plant or the manner in which the plant is operated. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The margin of safety for the containment barrier is, in part, preserved by compliance with 10 CFR 50 Appendix J. Although the proposed change will allow greater flexibility in meeting Appendix J requirements, the TS will continue to preserve compliance with 10 CFR Appendix J. 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, Louisiana 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.

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.

Description of amendment request: The proposed amendment would provide a special test exception that would allow an extension of the standby diesel generator (SDG) allowed outage time for a cumulative 21 days on each SDG once per fuel cycle, and it would also allow an extension of the essential cooling water (ECW) loop allowed outage time for a cumulative 7 days on each ECW loop once per fuel cycle. These extended allowed outage times will be used to perform required inspections and maintenance on the SDGs and the ECW system during power operation.

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 is inoperable there are still two complete trains available to mitigate the consequences of any accident. If the Essential Cooling Water loop is not operable during the 21 days the Standby Diesel Generator is inoperable, the Standby Diesel Generator's Engineered Safety Features bus and equipment in the train 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.

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 and Essential Cooling Water Allowed Outage Times once per fuel cycle 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 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 estimate of 1.7 x 10(−4), a second one submitted in 1992 to meet the IPE requirements from the Level 2 PSA/IPE with a CDF estimate of 4.4 x 10(−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 Technical Specifications, Plant Specific Data and incorporate the Emergency Transformer into the model. This update resulted in a CDF estimate of 2.07 x 10(−5). When the requested changes are modeled along with the compensatory actions, the resulting CDF estimate is 2.30 x 10(−5). While this is slightly higher (approx. 11%) than the updated results, it is still significantly lower (approx. 46%) than the previous Risk-Based Evaluation of Technical Specification submitted in 1993. Therefore, it is concluded that there is no significant reduction in the margin of safety.

Based on the above evaluation, Houston Lighting & Power has concluded that these changes do not involve any significant hazards considerations.

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., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036.

NRC Project Director: William D. Beckner.

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 2, 1995.

Description of amendment request: The proposed amendment would revise Technical Specifications 3.4.2.2 and 3.7.1.1 (Table 3.7–2) by relaxing the lift setting tolerances of the pressurizer safety valves from plus or minus 1% to plus or minus 2% and the main steam safety valves from plus or minus 1% to plus or minus 3%, respectively. In addition, a footnote would be added to require that the pressurizer safety valves and main steam safety valves setpoint tolerances be restored to within plus or minus 1% whenever a lift setting is determined to be outside plus or minus 1% following valve 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, which is presented below:

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

The proposed changes increase the ''as-found'' setpoint tolerances for the Pressurizer Safety valves from plus or minus 1% to plus or minus 2% and the Main Steam Safety valves from plus or minus 1% to plus or minus 3%. The proposed changes do not involve any hardware modifications to plant structures, systems, or components. An evaluation has determined that the proposed changes do not significantly affect the structural integrity of either the reactor coolant system or the main steam system.

The proposed setpoint tolerance of plus or minus 2% for the Pressurizer Safety valves and plus or minus 3% for the Main Steam Safety valve ''as-found'' condition was previously evaluated as part of the evaluation for the transition to VANTAGE 5H fuel. The evaluation was reviewed and approved by the NRC Staff as part of License Amendment Nos. 61 and 50 to Operating License NPF–76 and NPF–80. Since the VANTAGE 5H fuel evaluation incorporated these proposed changes, the calculated radiological release associated with that evaluation is unaffected. Similarly, this applies to the radiological dose associated with a steam generator tube rupture.

Additionally, the proposed change [sic] are consistent with the guidance provided by Section III and XI of the ASME Code.

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

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

Since the lift setting of a Pressurizer Safety valve or Main Steam Safety valve will be restored to plus or minus 1% whenever it is determined to be outside plus or minus 1%, the ''as-left'' setpoint tolerances for the Pressurizer Safety valves and Main Steam Safety valves are unchanged. The ''as-left'' setpoint will continue to satisfy the current technical specification requirement on lift setting tolerance. As such, there is no change in plant operation or equipment performance. Since neither plant operation or equipment performance is affected by the proposed changes, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Since the proposed changes are consistent with the guidance provided by Section III and XI of the ASME Code, and the proposed lift setting tolerance of plus or minus 2% for the Pressurizer Safety valves and plus or minus 3% for the Main Steam Safety valves has been incorporated into the design basis accident analyses, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the safety evaluation presented above for the proposed changes, Houston Lighting & Power has determined that the health and safety of the public will not be jeopardized. Therefore, the proposed 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, Texas 77488.

Attorney for licensee: Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036.

NRC Project Director: William D. Beckner.

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

Date of amendment request: April 13, 1995.

Description of amendment request: The proposed amendment would modify the Technical Specifications to allow use of laser-welded sleeves to repair defective steam generator tubes.

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

Conformance of the proposed amendments to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following:

(1) Operation of CNP [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.

The TS [tubesheet] or TSP [tube support plate] intersection LWS [laser-welded sleeve] configuration has been designed and analyzed in accordance with the requirements of the ASME [American Society of Mechanical Engineers] Code and RG [Regulatory Guide] 1.121. Fatigue and stress analyses of the sleeved tube assemblies produced acceptable results. Mechanical testing has shown that the structural strength of the Alloy 690 sleeves under normal faulted and upset conditions is within acceptable limits. Leak testing has demonstrated that primary to secondary leakage is not expected during all plant conditions, including the case where the seal weld is not produced in the lower joint of the TS sleeve. Testing shows that non-welding TS sleeve lower joints remained leaktight at temperature and pressure conditions representative of normal and accident conditions. Since laser welding produces a hermetic seal between the tube and sleeve, no leak path can be realized under any condition. Therefore, installation of LWSs will not influence offsite dose calculation for a postulated steam line break event.

The proposed technical specification change to support the installation of Alloy 690 LWSs does not adversely impact any previously evaluated design basis accident or the results of accident analyses for the current technical specification minimum reactor coolant system flow rate. The results of the qualification testing, analyses, and plant operating experience demonstrate that the sleeve assembly is an acceptable means of maintaining tube integrity. These aforementioned analyses and tests demonstrate that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis. Plugging limit criteria are established using the guidance of RG 1.121. Furthermore per RG 1.83 recommendations, the sleeved tube can be monitored through periodic inspections with present eddy current techniques.

Conformance of the sleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests, support the conclusion that installation of laser-welded tube sleeves will not increase the probability or consequences of an accident previously evaluated. Depending upon the break location for a postulated steam generator tube rupture event, implementation of tube sleeving could act to reduce the radiological consequences to the public due to reduced flow rate through a sleeved tube compared to a non-sleeved tube based on the restriction afforded by the sleeve wall thickness.

(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 laser-welded sleeving will not introduce significant or adverse changes to the plant design basis. Stress and fatigue analysis of the repair has shown the ASME Code and RG 1.121 allowable values are met. Implementation of laser-weld sleeving maintains overall tube bundle structural and leakage integrity during all plant conditions at a level consistent to that of the originally supplied tubing. Leak and mechanical testing of sleeves supports the conclusions of the calculations that the sleeve retains both structural and leakage integrity during all conditions. Sleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the sleeves. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. Since the sleeve design does not affect any other component or location of the tube outside of the immediate area repaired, in addition to the fact that the installation of sleeves and the impact on current plugging level analyses is accounted for, the possibility that laser-weld sleeving creates a new or different type of accident is not supported.

The design of thermally treated Alloy 600 and 690 sleeved tube assemblies have performed well historically with regard to corrosion. There are no reported instances of Alloy 600 thermally treated or Alloy 690 sleeve degradation for the greater than 35,000 sleeves that Westinghouse has installed in the U.S. Accelerated corrosion test results show the free span laser-weld joint (LWJ) (with post weld heat treatment) is capable of exhibiting a resistance to corrosion of greater that 10 times that of rolled tube transitions. Most LWS corrosion specimens did not experience degradation and were subsequently removed from the corrosion test media after a substantial testing period (supporting the 10x factor compared to roll transitions) was achieved. Several mill annealed Alloy 600 material heats were used for corrosion specimen preparation. All were documented by previous test to have been highly susceptible to PWSCC. The post weld heat treatment process applied to LWS free span joints is designed to achieve a minimum tube OD wall temperature of 1400°F adjacent to the weld and within the laser weld heat affected zone. Since the target temperature of 1400°F is achieved on the tube OD, a slightly higher temperature is achieved at the tube ID surface, where the weld cooling stresses are concentrated. Also, since the axial length of the laser weld and laser weld heat affected zone are relatively narrow compared to other sleeve welding processes, a narrower section of tube is required to be heat treated. Since the length of tube required to be heat treated is shorter in the LWS process than with other sleeving processes, lower residual stresses in the tube can be expected. Accelerated corrosion tests also show that non-heat treated laser-weld free span joints exhibit resistance to stress corrosion cracking equal to or greater than rolled tube transitions. An extensive data base exists on LWS joint performance in foreign plants in which the free span joints are not heat treated. Of the approximately 18,000 non-heat treated joints in service, none has exhibited a rapid corrosion potential. Corrosion testing of the TS sleeve lower joint LWJs exhibit a resistance to corrosion cracking of three to four times that of rolled tube transitions. These factors suggest postulated sleeve/tube assembly degradation would occur at a rate less than rolled transitions, and the potential for a sleeve/tube assembly with accelerated degradation rate characteristics more severe than rolled transitions, and the potential for a sleeve/tube assembly with accelerated degradation rate characteristics more severe than roll transitions is negligible.

Approximately 800 LWSs are currently in operation in the U.S. Some of these have been in service since April 1992. The plants in which these sleeves are installed have not experienced any adverse operational issues (such as primary to secondary leakage) as has been detected at other plants with sleeves which have experienced rapid corrosion of the parent tube.

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

The laser-welded sleeving repair of degraded steam generator tubes as identified in WCAP–13088 Rev. 3 has been demonstrated to restore the integrity of the tube bundle under normal and postulated accident conditions. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors the ASME Boiler and Pressure Vessel Code used in steam generator design. The plugging limit criteria for the sleeve has been established using the methodology of RG 1.121. The design of the sleeve joints have been verified by testing to preclude leakage during normal and postulated accident conditions. Implementation of laser-welded sleeving will reduce the potential for primary to secondary leakage during a postulated steam line break while maintaining available primary coolant flow area in the event of a LOCA. By removing from service degraded intersections through repair, the potential for tube leakage during a steam line break is reduced. These degraded intersections now are returned to a condition consistent with the design basis. While the installation of a sleeve causes a reduction in flow, the reduction is far below the reduction incurred by plugging. Therefore, far greater primary coolant flow area is maintained through sleeving. Use of RG 1.121 criteria assures that the margin of safety with respect to structural integrity is the same for the sleeves as for the original steam generator tubes.

The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of RG 1.83. Portions of the tube bridged by the sleeve joints are effectively isolated from the pressure boundary, and the sleeve then forms the pressure boundary in these areas. The areas of the sleeved tube assembly which require inspection are defined in Attachment 4 [WCAP–13088, ''Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report, Laser Welded Sleeves,'' January 1994].

In addition, since the installed sleeve represents a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with sleeves installed. As stated previously, weld fusion zone width is established using UT testing. The minimum acceptable weld width as determined by UT examination is approximately 50% wider than the minimum weld width which satisfies the stress conditions of the ASME Code.

The generic evaluation uses the pressure stress equation of Section NB 3224.1 of the ASME Code which is used to establish the minimum required wall thickness for the sleeve design and subsequently used to determine the level of sleeve wall degradation (depth by eddy current determination) that would require the sleeve to be removed from service. Using the [Delta]$P_{\text{Norm. Op.}}$ value of 1530 psi from Attachment 4 [WCAP–13088, ''Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report, Laser Welded Sleeves,'' January 1994] the limiting minimum required sleeve wall thickness is established. The sleeve wall plugging limit (using Attachment 4 [WCAP–13088, ''Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report, Laser Welded Sleeves,'' January 1994]) of 25% is subsequently established, and includes an allowance of 10% for eddy current uncertainty and 10% for growth, although sleeve wall degradation has not been observed to date in Westinghouse sleeves. The generic evaluation used the ASME Code minimum property values to establish the sleeve plugging limit. Certified material test reports indicate that the sleeve material properties are significantly higher than the ASME Code minimum values. The generic evaluation considered a primary to secondary pressure differential of 1530 psia, with a steam pressure of 720 psia, for normal operating conditions. CNP Units 1 can operate at full power with a reduced T_{hot} value and RCS pressure of 2250 psi. Steam pressure can be maintained as low as 650 psi (to keep T_{hot} as low as possible), but cannot go lower than 650 psi or the steam generator operating requirement of a primary to secondary [Delta]P of 1600 psi (max) will be exceeded. At this [Delta]$P_{Norm.\ Op.}$ value of 1600 psi, the sleeve minimum wall thickness requirement (and subsequently sleeve pressure boundary plugging limit) using ASME Code minimum material properties can be recalculated. For this condition (normal operating [Delta]P equal to 1600 psi), the sleeve minimum wall plugging limit is defined to be 23%. An allowance for eddy current uncertainty and continued degradation are included in this value. The minimum required wall thickness is determined by examining plant conditions at normal, upset, faulted, and test conditions. For Model 51 steam generators, the normal operating condition results in the limiting minimum wall thickness requirement.

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.

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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: December 20, 1993, as supplemented July 19, 1994, and February 28, 1995.

Description of amendment requests: The proposed amendments would revise the Technical Specifications to change Train A and B emergency loads from 8 hour to composite 4 hour, delete a load on the Train B batteries load list, and revise the operational loads on the Train N batteries. The supplemental submittals, made in response to NRC staff concerns, would also add surveillance requirements for a battery with signs of degradation and modify performance testing 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 was published in the **Federal Register** on February 2, 1994 (59 FR 4939). This analysis was not changed by the supplemental submittals.

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, including the supplemental submittals, 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: Cynthia A. Carpenter, Acting.

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: March 31, 1995.

Description of amendment requests: The proposed amendments would revise the technical specifications to provide increased flexibility in the operation of the containment personnel airlocks during core alterations.

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

Per 10 CFR 50.92, a proposed change does not involve a significant hazards consideration if the change does not:

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 a margin of safety.

Criterion 1

The design basis fuel handling accident is the rupture of the highest rated fuel assembly. As discussed previously [in the application], the consequences of an accident inside containment (i.e., site boundary dose) with both airlock doors are bounded by the existing fuel handling accident currently presented in our UFSAR [Updated Final Safety Analysis Report].

Since the containment airlock doors do not affect the failure mechanism of a fuel assembly during a fuel handling accident, we believe that this amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, no credit was taken for containment closure in the accident analysis. Therefore, based on these considerations, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

As stated in response to criterion one, the position of the containment airlock doors in no way affects the mechanism by which a spent fuel assembly is damaged during a fuel handling accident. Thus, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The margin for safety as defined in 10 CFR 100 has not been reduced. As discussed previously, the existing fuel handling accident analysis for an event inside containment takes no credit for the isolation of containment. As a result, the position of the airlock doors has no impact on the analyzed site boundary doses resulting from such an accident. Based on these considerations, it is concluded that the changes do not involve a significant reduction in a margin of safety. The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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: Cynthia A. Carpenter, Acting.

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

Date of amendment request: May 2, 1995.

Description of amendment request: This license amendment request revises Surveillance Requirement (SR) 4.7.A.2.f.1 to allow a one-time schedular extension of the two year Type B Local Leak Rate Test (LLRT) interval required for the Drywell Head and Manport (penetrations DWH and X–4 respectively). This extension will allow the Type B testing of penetrations DWH and X–4 to be deferred from the current due date of July 17, 1995, until Refueling Outage No. 16 (RE–16), which is currently scheduled to commence in October 1995.

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

The enclosed Technical Specifications change is judged to involve no significant hazards based on the following:

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

This license amendment request revises Surveillance Requirement (SR) 4.7.A.2.f.1 to allow the one-time schedular extension of the two year Type B Local Leak Rate Test (LLRT) interval required for the Drywell Head and Manport (Penetrations DWH and X–4 respectively). This extension will allow Penetrations DWH and X–4 to be Type B tested during Refueling Outage No. 16 (RE–16), which is currently scheduled to commence October 1995. Currently, the two year maximum interval for these penetrations comes due July 17, 1995. The District has concluded that a one-time extension of approximately six months beyond the two year limit will not result in a significant increase in the probability of these penetrations failing to perform their safety function. This conclusion is based on the previous LLRT surveillance history of Penetrations DWH and X–4, which have not failed an LLRT in the last 19 years. The surveillance history demonstrates that these penetrations are not subject to leak related failures.

Additionally, the seals associated with these penetrations will not have experienced significantly more radiation and heat exposure by the conclusion of the proposed extension than they would have during the current two year interval. Although some radiation and heat is present during plant shutdowns, the seal degradation resulting from these conditions is significantly slower. Because seal degradation is a function of heat and radiation, and is generally not a function of time, the District has concluded that the one-time extension will not result in a significant increase of seal degradation. Because seal failure for these penetrations is largely based on the rate of seal degradation, the probability of the failure of these penetrations is not significantly increased. Therefore, a significant increase in the probability or consequences of an accident is not created.

This proposed change does not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. The change does not degrade the performance of any safety system assumed to function in the accident analysis. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This license amendment request involves the one-time schedular extension of the LLRT interval requirement for Penetrations DWH and X–4. SR 4.7.A.2.f.1 is being revised to extend the surveillance test interval for Penetrations DWH and X–4 to coincide with RE–16, currently scheduled to commence October 1995. A one-time extension of the subject surveillance interval does not involve the creation, deletion, or modification of the function of any structure, system, or component, nor does this change introduce or change any mode of plant operation. This proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

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

This license amendment request involves the one-time extension of the two year maximum surveillance test interval for Penetrations DWH and X–4 from the current due date of July 17, 1995, to instead coincide with RE–16, which is scheduled to commence October 1995. By the time these tests are performed, the penetration seals will not have experienced significantly more radiation and heat than they would have during the previous test intervals. Therefore, the penetration seals will not have experienced significant degradation as a result of the extended interval. Furthermore, Penetrations DWH and X–4 have not failed an LLRT in the last 19 years. The surveillance history demonstrates that these penetrations are not subject to leak related failure. This proposed change does not involve any change to plant design, equipment instrument setpoints, or operation. Therefore, this proposed change does not create 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: Auburn Public Library, 118 15th Street, Auburn, NE 68305.

Attorney for licensee: Mr. G.D. Watson, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Project Director: William D. Beckner.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: March 29, 1995.

Description of amendment request: The request will revise Technical Specification Section 3.10.5 to allow more than one control bank to be fully withdrawn from the core simultaneously for rod drop time response testing. Specifically, the change will delete, (1) the limiting condition for operation (LCO) 3.10.5.a and (2) a reference to the full length shutdown rods from LCO 3.10.5. The change will also add a statement that ''The SHUTDOWN MARGIN requirement of Section 3.1.1.1.2 shall be met without credit for withdrawn control rods.'' Other editorial changes are to be made for consistency.

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 changes do not involve an SHC because the changes would not:

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

The proposed changes cannot initiate an event since the plant will be maintained shutdown at all times. Thus, there is no increase in the probability of occurrence of an accident previously evaluated.

The proposed changes do not degrade the performance of any safety system nor do they alter any assumptions made in the accident analyses. Currently, the technical specifications allow the rod position indication system to be disabled for each control bank while performing this test. In addition, this system is not a safety system credited in the accident analyses. Therefore, allowing more than one bank to have its indication removed during the test does not degrade any safety system. Since the shutdown margin will be maintained without crediting these rods, there is no change to the assumptions made in the accident analyses. Thus, there is no increase in the consequences of an accident previously evaluated.

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

The proposed changes do not position the control rods into any new configurations or sequence not previously analyzed. Ejected rod worths are evaluated for ARI–1 (all rods in with the most reactive rod out) and, therefore, bound the test configuration. In addition, the reactivity state of the system is maintained shut down by the margin required in Technical Specification 3.1.1.1.2 without crediting the control rods. Therefore, there is no possibility of a new or different type of accident than previously analyzed.

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

The proposed changes do not impact any of the physical protective boundaries, safety systems, or operating conditions. The plant will be maintained shut down without crediting the control rods. The accident analyses is not impacted and, therefore, there is no 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: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L.M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: April 28, 1995.

Description of amendment request: The request will revise the diesel generator (DG) fuel oil testing that is performed on new fuel prior to the addition of the new fuel to the storage tank.

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 changes do not involve an SHC because the changes would not:

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

The proposed changes: correct a typographical error by providing the appropriate range for the Saybolt viscosity; replace the qualitative clear and bright test with a quantitative water and sediment test for new fuel prior to adding it to the storage tank; and clarify that a calculated cetane index may be performed in lieu of obtaining the cetane number for the fuel. The water and sediment test provides a quantitative method for evaluating water and sediment, and will require a more restrictive limit of 0.05 percent by volume of water and sediment than the 0.10 percent recommended by the manufacturer. The cetane index has been shown to be representative of the cetane number for the fuel. The DG capability to start and operate is enhanced by the proposed changes. Therefore, the changes have no negative effect on the consequences of the previously evaluated accidents.

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

The proposed changes do not alter or affect the design, function, failure mode, or operation of the plant. The proposed changes have no adverse effect on the quality of the fuel oil that is utilized by the DG. The proposed changes are administrative in nature and do not involve any physical alteration to any plant system or change the method by which any safety-related system performs its function. For these reasons, there is no possibility of an accident of a different type than previously evaluated.

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

The proposed changes will assure that the DG fuel oil meets DG manufacturer's quality requirements by the performance of the recommended testing of the DG fuel oil. The proposed changes will not impact the margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: April 28, 1995.

Description of amendment request: The proposed revision to the Action Statement of Limiting Condition for Operation (LCO) 3.7.5 would permit Millstone Unit No. 3 to remain in Modes 1 through 4 with the average water temperature of the ultimate heat sink (UHS) greater than 75°F (but lower than 77°F) for 12 hours. An additional action would be added which would require the plant to be placed in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours upon identifying that the UHS temperature is greater than 77°F.

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 changes do not involve an SHC because the changes would not:

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

The proposed addition of a 12-hour period to monitor the UHS temperature to the Technical Specification LCO Action Statement does not involve an increase in the probability of an accident previously evaluated. The probability of an accident previously evaluated is not increased by a short-term increase in the UHS temperature. The probability of FSAR Chapter 15 Condition IV accidents occurring in conjunction with the short duration increase in service water inlet temperature above 75°F is low enough such that they are not risk significant. Further, an evaluation has been performed that safe shutdown will be achieved and maintained for a loss of offsite power event and a steam generator tube rupture event with the additional consideration of a single failure with service water inlet temperatures as high as 77°F. There has been no significant increase in the consequences of these events previously evaluated.

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

The proposed technical specification change does not create the possibility of a new or different kind of accident previously evaluated. The addition of a 12-hour time period to monitor the UHS temperature increases the amount of time that is allowed for the plant to be in HOT STANDBY from 6 to 18 hours should the UHS temperature increase above 75°F. This extension of the time allowed for the plant to be in HOT STANDBY does not change the plant configuration. As such, the change does not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed technical specification change does not involve a significant reduction in the margin of safety. The addition of a 12-hour time period to monitor the UHS temperature increases the time required for the plant to be in HOT STANDBY from 6 to 18 hours should the UHS temperature exceed 75°F. An evaluation has been performed to demonstrate that the risk significance associated with the increased action time is very low. In addition, safe shutdown capability has been demonstrated for service water inlet temperatures as high as 77°F.

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: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L.M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: May 1, 1995.

Description of amendment request: Technical Specifications that specify an 18-month surveillance will be changed to state that these surveillances are to be performed at least once each refueling (i.e., 24 months).

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 does not involve an SHC because the change would not:

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

The proposed change to Surveillance Requirement 4.6.3.2 of the Millstone Unit No. 3 Technical Specifications extends the frequency for verifying that each containment isolation valve actuates to its required position in response to Phase A and Phase B isolation test signals, and for verifying that each containment purge supply and exhaust isolation valve actuates to its required position in response to a containment high radiation test signal. The proposal would extend the frequency from at least once per 18 months to at least once per refueling interval (24 months).

The proposed change to Surveillance Requirement 4.6.3.2 does not alter the intent or method by which the surveillances are conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. As such, the proposed change to the frequency of Surveillance Requirement 4.6.3.2 will not degrade the ability of the containment isolation valves to perform their safety function. Also, the containment isolation valve arrangements are not vulnerable to single failures, because they provide at least two barriers between the atmosphere outside the containment and the atmosphere within the containment, the reactor coolant system, or systems that would become connected to the containment atmosphere or the reactor coolant system as a result of, or subsequent to, a DBA.

Additional assurance of containment isolation valve operability is provided by Surveillance Requirements 4.6.3.1 and 4.6.3.3. Surveillance Requirement 4.6.3.1 requires that a containment isolation valve will be restored to an operable status following the performance of work on the containment isolation valve or its ancillaries. Surveillance Requirement 4.6.3.3 requires the confirmation of the mechanical operability of the containment isolation valves by the inservice inspection program. The proposed change does not modify these requirements.

Additionally, Surveillance Requirements 4.3.2.1 and 4.3.3.1 assure the operability of the automatic isolation logic (Phase A and Phase B isolation signals and containment high radiation signal) for the containment isolation valves by performing tests on a monthly basis. This proposed change does not modify these Surveillance Requirements.

Equipment performance over the last four operating cycles was evaluated to determine the impact of extending the frequency of Surveillance Requirement 4.6.3.2. This evaluation included a review of surveillance results, preventive maintenance records, and the frequency and type of corrective maintenance. It has been concluded that the containment isolation valves are highly reliable, and that there is no indication that the proposed extension could cause deterioration in valve condition or performance.

Based on the above, the proposed change to Surveillance Requirement 4.6.3.2 of the Millstone Unit No. 3 Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed change to Surveillance Requirement 4.6.3.2 of the Millstone Unit No. 3 Technical Specifications extends the frequency for verifying that each containment isolation valve actuates to its required position in response to Phase A and Phase B isolation test signals, and for verifying that each containment purge supply and exhaust isolation valve actuates to its required position in response to a containment high radiation test signal. The proposal would extend the frequency from at least once per 18 months to at least once per refueling interval (24 months).

The proposed change does not alter the intent or method by which the surveillances are conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. As such, the proposed change in the frequency of Surveillance Requirement 4.6.3.2 will not degrade the ability of the containment isolation valves to perform their safety function. Also, the containment isolation valve arrangements are not vulnerable to single failures, because they provide at least two barriers between the atmosphere outside the containment and the atmosphere within the containment, the reactor coolant system, or systems that would become connected to the containment atmosphere or the reactor coolant system as a result of, or subsequent to, a DBA.

Based on the above, the proposed change to Surveillance Requirement 4.6.3.2 of the Millstone Unit No. 3 Technical Specifications will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to Surveillance Requirement 4.6.3.2 of the Millstone Unit No. 3 Technical Specifications extends the frequency for verifying that each containment isolation valve actuates to its required position in response to Phase A and Phase B isolation test signals, and for verifying that each containment purge supply and exhaust isolation valve actuates to its required position in response to a containment high radiation test signal. The proposal would extend the frequency from at least per 18 months to at least once per refueling interval (24 months).

The proposed change does not alter the intent or method by which the surveillances are conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. As such, the proposed change in the frequency of Surveillance Requirement 4.6.3.2 will not degrade the ability of the containment isolation valves to perform their safety function. Also, the containment isolation valve arrangements are not vulnerable to single failures, because they provide at least two barriers between the atmosphere outside the containment and the atmosphere within the containment, the reactor coolant system, or systems that would become connected to the containment atmosphere or the reactor coolant system as a result of, or subsequent to, a DBA.

Additional assurance of the operability of the containment isolation valves is provided by Surveillance Requirements 4.6.3.1 and 4.6.3.2. Also, assurance of the operability of the automatic actuation logic of the containment isolation valves is provided by Surveillance Requirements 4.3.2.1 and 4.3.3.1.

Equipment performance over the last four operating cycles was evaluated to determine the impact of extending the frequency of Surveillance Requirement 4.6.3.2. This evaluation included a review of surveillance results, preventive maintenance records, and the frequency and type of corrective maintenance. It has been concluded that the containment isolation valves are highly reliable, and that there is no indication that the proposed extension could cause deterioration in valve condition or performance.

Based on the above, the proposed change to Surveillance Requirement 4.6.3.2 of the Millstone Unit No. 3 Technical Specifications 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: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment request: May 8, 1995.

Description of amendment request: The proposed amendment would change Technical Specifications 2.3, 3.1, 3.2, 3.3 and 3.6. These changes are in accordance with the guidance of Generic Letter 93–05, ''Line Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993.

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.

GL 93–05, Item 4.2, Control Rod Movement Test; Specification 3.2, Table 3–5, Item 2

Omaha Public Power District (OPPD) proposes to extend the control element assembly (CEA) partial movement surveillance test of Specification 3.2, Table 3–5, Item 2 from a biweekly to a quarterly frequency. This change is based on operating experience and the recommendation of Generic Letter (GL) 93–05, Item 4.2.1. A review of previous surveillance tests and interviews with personnel familiar with the test did not identify any prior surveillance test failures. Industry experience has shown that this test can cause reactor trips, dropped rods and unnecessary challenges to safety systems as stated in NUREG–1366, ''Improvements to Technical Specification Requirements,'' dated December 1992. Therefore, extending the frequency of conducting this surveillance test may be beneficial to plant operations and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

GL 93–05, Item 5.14, Radiation Monitors; Specification 3.1, Table 3–3, Items 3b, 4 and 5b

OPPD proposes to replace descriptive wording in Specification 3.1, Table 3–3, Items 3a/b and 5a/b with defined terms. OPPD also proposes to extend surveillance of the area, post-accident and primary to secondary leak-rate radiation monitors (Specification 3.1, Table 3–3, Items 3b and 5b) from a monthly to a quarterly frequency as recommended by GL 93–05, Item 5.14. Most of these monitors are new (i.e., installed within the last two cycles) or contain many new components. The value of monthly testing is greatly reduced as the new monitors include self checking circuitry that will indicate monitor failure, loss of power, or loss of background. Although post accident radiation monitors RM–091 A/B are not new, Station operating experience has shown that they are reliable. In cases where new components interface with older components, the older components have a history of reliable operation.

Readings and internal test signals are used to verify instrument operation on a daily basis. In addition, the proposed frequency (quarterly) is the same frequency currently specified for the containment radiation high signal (CRHS) monitors (Specification 3.1, Table 3–2, Item 6b), which generate an engineered safeguards signal. Replacing descriptive words with defined terms ensures consistency and that the surveillance test accomplishes its purpose.

A quarterly surveillance conserves resources, increases the availability of the area, post-accident and primary to secondary leak-rate detection radiation monitors and is consistent with CRHS monitor testing. These proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

OPPD proposes to delete Specification 3.1, Table 3–3, Item 4 on surveillance testing of the emergency plan radiation instruments. These are portable instruments stored in specified locations for use by emergency response personnel in the event of an accident. The instruments may be used to survey onsite/offsite areas for radioactivity or to facilitate the decontamination of personnel following an accident. No limiting condition for operation (LCO) action statement is associated with these instruments. As a result, there is no basis for the TS to contain a surveillance requirement for them. In addition, retaining this surveillance in the TS is unnecessary since it does not meet criteria 1 through 4 of the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. Therefore, since these instruments are not utilized until after an accident has occurred, and do not assist in accident mitigation, deleting this surveillance requirement does not involve a significant increase in the probability or consequences of an accident previously evaluated.

GL 93–05, Item 6.1, Reactor Coolant System Isolation Valves; Specification 3.3(2)a

The reactor coolant system (RCS) pressure isolation valves have proven to be very reliable. Therefore, OPPD proposes to extend the time that the plant can be in cold shutdown before the test is required (Specification 3.3(2)a) from 72 hours to 7 days, following the recommendation of GL 93–05, Item 6.1. A review of previous surveillance tests and interviews with personnel familiar with the test did not identify any prior surveillance test failures. This proposed change will reduce radiation exposure and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

GL 93–05, Item 7.4, Accumulator Water Level and Pressure Channel Surveillance Requirements; Specification 2.3(2)g, Specification 3.1, Table 3–2, Item 14a

OPPD proposes to revise Specification 2.3(2)g following the recommendation of GL 93–05, Item 7.4. This revision will clarify that the safety injection tank (SIT) level and/or pressure instrumentation may be inoperable, which does not alter the intent of the Specification, but is more accurate in defining when the Specification applies. This revision also extends the time limit for inoperability of SIT instrumentation from 1 hour to 72 hours, which is justified based upon a review of historical data. As stated in NUREG–1366: ''While technically inoperable, the accumulator [SIT] would be available to fulfill its safety function during this time, and thus, this change would have a negligible increase on risk.''

Currently, Specification 2.3(2)g allows only one hour for SIT level and pressure instrumentation to be inoperable, which is insufficient time to initiate repairs. A review of historical data determined that SIT water level stays relatively constant while pressure decreases slightly over time. It is unlikely that SIT pressure would decrease below the Specification 2.3(1)c limit of 240 psig during the proposed 72-hour LCO, since SIT pressure is normally maintained around 255 psig (Updated Safety Analysis Report (USAR), Section 6.2.3.5).

OPPD's proposal to revise Specification 3.1, Table 3–2, Item 14a to require shiftly verification that SIT level and pressure are within limits and remove reference to verifying ''indications are between independent high and low alarms for level and pressure,'' is consistent with the guidance of GL 93–05, Item 7.4. As stated in GL 93–05, Item 7.4, the operability of SIT instrumentation is not directly related to the capability of a SIT to perform its safety function. OPPD proposes to suspend this surveillance on the affected SIT while the instrumentation is being repaired, since as stated above, SIT level and pressure are expected to stay within the limits of Specification 2.3(1)c during the proposed 72 hour LCO. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

GL 93–05, Item 8.1, Containment Spray System; Specification 3.6(2)b

OPPD proposes to extend the surveillance frequency for verifying that the containment spray nozzles are open (Specification 3.6(2)b) from five to ten years following the recommendation of GL 93–05, Item 8.1. Minor revisions to statements in the basis of Specification 3.6 that refer to conducting this test at five year intervals are proposed also. OPPD has not experienced problems with obstructions in the containment spray nozzles as determined by a review of previous surveillance tests and personnel interviews. Of the three instances reported in NUREG–1366 concerning obstructions of containment spray nozzles, all were problems related to construction errors. Any construction errors in the FCS containment spray system would have been found by previous surveillance tests.

The problem that occurred at San Onofre Unit 1 (clogging of several containment spray nozzles following the application of a coating material to the carbon steel piping) is not a concern at FCS since the FCS containment spray system piping and valves are constructed of stainless steel (USAR Table 6.3–2). Thus, extending the surveillance frequency of Specification 3.6(2)b from five to ten years does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

GL 93–05, Item 4.2, Control Rod Movement Test; Specification 3.2, Table 3–5, Item 2

OPPD's proposal to extend the CEA partial movement surveillance test (Specification 3.2, Table 3–5, Item 2) to a quarterly frequency is based on operating experience and the recommendation of GL 93–05, Item 4.2.1. The proposed change only lengthens the time between surveillance tests and will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

GL 93–05, Item 5.14, Radiation Monitors; Specification 3.1, Table 3–3, Items 3b, 4 and 5b

OPPD proposes to replace unnecessary wording in Specification 3.1, Table 3–3, Items 3a/b and 5a/b with defined terms and to extend the surveillance frequency of Items 3b and 5b from monthly to quarterly based on the recommendation of GL 93–05, Item 5.14. Most of the area, post accident and primary to secondary leak-rate detection radiation monitors are new or contain new components. The new monitors include self checking circuitry that provides failure notification. Although post accident radiation monitors RM–091 A/B are not new, they have an excellent operating history. The proposed changes introduce consistent use of terminology and lengthen the time between surveillance tests and will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

OPPD's proposal to delete Specification 3.1, Table 3–3, Item 4 on surveillance testing of the emergency plan radiation instruments will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. Since these instruments are not utilized until after an accident has occurred, and do not assist in accident mitigation, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

GL 93–05, Item 6.1, Reactor Coolant System Isolation Valves; Specification 3.3(2)a

The RCS pressure isolation valves have proven to be very reliable. As a result, OPPD proposes to extend the time that the plant can be in cold shutdown before the test is required (Specification 3.3(2)a) from 72 hours to 7 days following the recommendation of GL 93–05, Item 6.1. The proposed change will reduce radiation exposure and does not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

GL 93–05, Item 7.4, Accumulator Water Level and Pressure Channel Surveillance Requirements; Specification 2.3(2)g, Specification 3.1, Table 3–2, Item 14a

OPPD's proposal to revise Specification 2.3(2)g following the guidance of GL 93–05, Item 7.4 more accurately states when the specification should apply and extends the time limit for inoperability of SIT instrumentation from 1 hour to 72 hours based upon a review of historical data. The proposed change will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. As stated in NUREG–1366: ''While technically inoperable, the accumulator [SIT] would be available to fulfill its safety function during this time, and thus, this change would have a negligible increase on risk.''

OPPD's proposal to revise Specification 3.1, Table 3–2, Item 14a to require shiftly verification that SIT level and pressure are within limits and remove reference to verifying ''indications are between independent high and low alarms for level and pressure,'' is consistent with the guidance of GL 93–05, Item 7.4. As stated in GL 93–05, Item 7.4, the operability of SIT instrumentation is not directly related to the capability of a SIT to perform its safety function. OPPD proposes to suspend this surveillance on the affected SIT while the instrumentation is being repaired, since SIT level and pressure are expected to stay within the limits of Specification 2.3(1)c during the proposed 72 hour LCO. Therefore, since these proposed changes do not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits, they do not create the possibility of a new or different kind of accident from any accident previously evaluated.

GL 93–05, Item 8.1, Containment Spray System; Specification 3.6(2)b

OPPD's proposal to extend the surveillance frequency for verifying that the containment spray nozzles are open (Specification 3.6(2)b) from five to ten years as recommended by GL 93–05, Item 8.1 is justified by operating experience. OPPD has not experienced problems with obstructions in the containment spray nozzles as determined by a review of previous surveillance tests and personnel interviews. The problem that occurred at San Onofre Unit 1 (clogging of several containment spray nozzles following the application of a coating material to the carbon steel piping) is not a concern at FCS since the FCS containment spray system piping and valves are constructed of stainless steel (USAR Table 6.3–2).

The proposed change only extends the time between surveillance tests and revises associated basis statements to support the extension. The proposed change will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. Therefore, OPPD's proposal to extend the surveillance frequency of Specification 3.6(2)b from five to ten years 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.

GL 93–05, Item 4.2, Control Rod Movement Test; Specification 3.2, Table 3–5, Item 2

OPPD's proposal to extend the CEA partial movement surveillance test of Specification 3.2, Table 3–5, Item 2 to a quarterly frequency is based on operating experience and the recommendation of GL 93–05, Item 4.2.1. A review of previous surveillance tests and interviews with personnel familiar with the test did not identify any prior surveillance test failures. Industry experience has shown that this test can occasionally cause reactor trips, dropped rods and unnecessary challenges to safety systems as stated in NUREG–1366. Therefore, extending the frequency of conducting this surveillance test may be beneficial to plant operations and does not involve a significant reduction in a margin of safety.

GL 93–05, Item 5.14, Radiation Monitors; Specification 3.1, Table 3–3, Items 3b, 4 and 5b

OPPD proposes to replace descriptive wording in Specification 3.1, Table 3–3, Items 3a/b and 5a/b with defined terms and to extend the surveillance frequency of Items 3b and 5b from monthly to quarterly based on the recommendation of GL 93–05, Item 5.14. Most of the area, post accident and primary to secondary leak-rate detection radiation monitors are new or contain new components. Post accident radiation monitors RM–091 A/B are not new but have a history of reliable operation. The value of monthly testing is greatly reduced since the new monitors include self checking circuitry that provides failure notification. The proposed changes introduce consistent use of terminology and lengthen the time between surveillance tests and therefore do not involve a significant reduction in a margin of safety.

OPPD's proposal to delete Specification 3.1, Table 3–3, Item 4 is justified because the emergency plan radiation instruments are portable instruments that are not utilized until after an accident has occurred. The instruments are checked for proper operation before use and since these instruments do not assist in accident mitigation, the deletion of this surveillance requirement does not involve a significant reduction in a margin of safety.

GL 93–05, Item 6.1, Reactor Coolant System Isolation Valves; Specification 3.3(2)a

The RCS pressure isolation valves have proven to be very reliable. Therefore, consistent with the guidance of GL 93–05, Item 6.1, OPPD proposes to revise Specification 3.3(2)a and extend the time that the plant is allowed to be in cold shutdown before this surveillance test is required from 72 hours to 7 days. This change will reduce radiation exposure and does not involve a significant reduction in a margin of safety.

GL 93–05, Item 7.4, Accumulator Water Level and Pressure Channel Surveillance Requirements; Specification 2.3(2)g, Specification 3.1, Table 3–2, Item 14a

OPPD's proposal to revise Specification 2.3(2)g following the guidance of GL 93–05, Item 7.4 more accurately states when the specification applies and extends the time limit for inoperability of SIT instrumentation from 1 to 72 hours based upon historical data. As stated in NUREG–1366: ''While technically inoperable, the accumulator [SIT] would be available to fulfill its safety function during this time, and thus, this change would have a negligible increase on risk.''

OPPD's proposal to revise Specification 3.1, Table 3–2, Item 14a to require shiftly verification that SIT level and pressure are within limits and remove reference to verifying ''indications are between independent high and low alarms for level and pressure,'' is consistent with the guidance of GL 93–05, Item 7.4. As stated in GL 93–05, Item 7.4, the operability of SIT instrumentation is not directly related to the capability of a SIT to perform its safety function. OPPD proposes to suspend this surveillance on the affected SIT while the instrumentation is being repaired, since SIT level and pressure are expected to stay within the limits of Specification 2.3(1)c during the proposed 72 hour LCO. Therefore, these proposed changes do not involve a significant reduction in a margin of safety.

GL 93–05, Item 8.1, Containment Spray System; Specification 3.6(2)b

OPPD's proposal to extend the surveillance frequency for verifying that the containment spray nozzles are open (Specification 3.6(2)b) from five to ten years as recommended by GL 93–05, Item 8.1 is justified by operating experience. OPPD has not experienced problems with obstructions in the containment spray nozzles as determined by a review of previous surveillance tests and personnel interviews.

The problem that occurred at San Onofre Unit 1 is not a concern at FCS since the FCS containment spray system piping and valves are constructed of stainless steel (USAR Table 6.3–2). Therefore, OPPD's proposal to extend the surveillance frequency of Specification 3.6(2)b from five to ten years and revise associated basis statements 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: James R. Curtiss, Winston & Strawn, 1400 L Street, Washington, DC 20005–3502.

NRC Project Director: William H. Bateman.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of amendment request: April 10, 1995.

Description of amendment request: The proposed amendment would revise License No. DPR–7, to permit the provisions of 10 CFR 50.59 to be applied with respect to changes to the facility or procedures described in the Decommissioning Plan or changes to the Decommissioning Plan, and the conduct of tests or experiments not described in the Decommissioning Plan.

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?

The probability or consequences of an accident previously evaluated will not be effected by the ability to perform safety analyses. As outlined in 10 CFR 50.59, the impact of performing special tests, experiments, and modifications would be evaluated to verify there would be no impact on previously evaluated accidents or increase the probability or consequences of an accident occurring.

Therefore, the proposed change does not involve a significant 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?

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because there is no physical alteration to any plant system, nor is there a change in the method in which any quality-related activities are performed or any direct change in equipment or system function or operation. The proposed change is administrative in nature.

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 proposed change to the HBPP License does not affect the margin of safety of any accident analysis since it does not affect the parameters for any accident analysis, and has no effect on the current operating methodologies or actions that govern plant performance.

Therefore, the proposed change does 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: Humboldt County Library, 636 F Street, Eureka, California 95501.

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

NRC Project Director: Seymour H. Weiss.

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

Date of application for amendments: January 17, 1995 as supplemented by letter dated March 30, 1995.

Description of amendment request: The proposed change revises the Peach Bottom Atomic Power Station, Units 2 and 3 technical specifications to reflect the replacement of the source range monitor (SRM) and intermediate range monitor (IRM) systems with a new system referred to as the wide range neutron monitoring system (WRNMS).

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

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

The use of the WRNMS as discussed herein will not increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of design basis accidents (DBAs) occurring is not affected by the WRNMS. The only plant safety analysis affected by WRNMS is the Rod Withdrawal Error (RWE) at low power, and a reanalysis assuming use of WRNMS shows that the criteria of 170 cal/gm for fuel enthalpy increase under RWE is satisfied; thus, RWE is not a limiting event. Scram setpoints (equipment settings that initiate automatic plant shutdowns) will be established such that there is no increase in scram frequency due to the WRNMS. No new challenges to safety-related equipment will result from WRNMS.

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

As summarized below, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The components of the WRNMS will be supplied to equivalent or better design and qualification criteria than is currently required for the plant. Equipment that could be affected by WRNMS has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, system interaction, or equipment failure mode was identified. Therefore, the WRNMS will not adversely affect plant equipment.

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

All the SRM/IRM functions required in the Technical Specifications are replaced with equivalent (more reliable) WRNMS functions. The accuracy and response times of the WRNMS are superior to those of the SRM/IRM subsystems. Implementation of the WRNMS does not affect any fuel or safety limit. The applicable Bases of the Technical Specifications have been rewritten, and the new Bases maintain the equivalent margin of safety as was provided by the SRM/IRM Bases.

The WRNMS (a) does not decrease a channel trip occurrence beyond its acceptable limit, (b) does not increase a channel response time beyond its acceptable limit, (c) increases indicated accuracies, and (d) does not cause any plant parameter for any analyzed event to fall outside of its acceptable limit(s).

The surveillance test frequency change of 7 to 31 days is based on the WRNMS having (1) fixed in-core detectors, (2) greater reliability than the SRMs and IRMs, and (3) self test features. The 13 second allowable value for the WRNM Period-Short surveillance, and the surveillance test frequency change of 184 days to 24 months is based on trip setpoint calculations using GE's standard (NRC approved) setpoint methodology.

The WRNMS will not involve a reduction in a margin of safety, as loads on plant equipment will not increase, and reactions to or results of transients and postulated accidents will not increase from those presently approved by the NRC.

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

Attorney for Licensee: J.W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, Pennsylvania 19101.

NRC Project Director: John F. Stolz.

Pennsylvania Power and Light Company, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: April 11, 1995.

Description of amendment request: This amendment would extend on a one time basis the allowed outage time in the Susquehanna Steam Electric Station Technical Specification 3.8.1.1 from 3 to 7 days for one offsite circuit being out of service. This change will provide additional time if needed to complete modifications to an offsite circuit.

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

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

The consequences of losing offsite power have been evaluated in the FSAR [Final Safety Analysis Report] and the Station Blackout evaluation. Increasing the AOT [allowed outage time] for T–10 [an offsite power source] from 3 to 7 days does not increase the consequences of a LOOP [loss of offsite power] event nor change the evaluation of LOOP events as stated in the FSAR or Station Blackout evaluation.

Allowing T–10 to be removed from service for an additional 4 days does increase slightly the possibility of a LOOP event as shown in PP&L's [Pennsylvania Power & Light Company's] engineering study. However, implementing the following compensatory actions reduces the probability of a LOOP event:

1. prohibiting high risk activities within the confines of the plant or the grid system that may result in a loss of T–20 [the second offsite power transformer] during the T–10 outage,

2. performing the modification during the Fall when the frequency of grid and weather related LOOPs are reduced,

3. requiring a unit shutdown if the HPCI [high pressure core injection] system becomes inoperable during the T–10 outage,

4. requiring a unit shutdown if the SLCS [standby liquid control system] becomes inoperable during the T–10 outage,

5. requiring that within 24 hours prior to taking T–10 out of service, Surveillance 4.8.1.1.2.a.4 be successfully completed on the aligned diesel generators, and

6. maintaining the following equipment operable during the T–10 work window and restoring any failed system/component to operable status as soon as possible (The failed system/component shall be worked around the clock):

- Both CRD [control rod drive] pumps,
- Diesel fire pump, yard fire hydrant (1FH122) and associated hydrant hose station,
- RHR [residual heat removal system]/RHRSW [residual heat removal service water system]/ESW [emergency service water system] for suppression pool cooling,
- RHR/RHRSW cross tie valves,
- RCIC [reactor core isolation cooling]
- CIG [containment instrument gas] 150 psig header and bottles,
- Turbine Building Closed Cooling Water System (one pump and heat exchanger),
- Portable diesel generator,
- HV–141–F019.

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

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

Allowing the AOT for T–10 to increase from 3 to 7 days is a one time exemption in order to install the new T–10 tap and 230 kV switch yard. The accident analyses affected by this extension are the LOOP events. The remaining portions of the station and equipment are not altered by this change. The potential for the loss of other plant systems or equipment to mitigate the effects of an accident are not altered. One offsite source of power will be out of service for an additional 4 days and compensatory actions will be initiated to lessen the effect of having the offsite power source out of service for an additional 4 days. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change allows, on a one time basis, T–10 to be out of service for an additional 4 days. This increase in AOT for T–10 results in a slight decrease in the margin of safety (defined as core damage frequency) with respect to having two offsite sources available per Specification 3.8.1.1. By implementing the compensatory measures as described in Item 1 above, the margin of safety is increased to be the equivalent of allowing the offsite power source (T–10) to be out of service for 3 days as is allowed by the existing Specification. Therefore, this one time exemption will not involve a significant reduction in safety margin.

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

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: April 10, 1995.

Description of amendment request: This amendment would relocate response time limit tables from the Susquehanna Steam Electric Station Unit 1 and Unit 2 Technical Specifications (TS) to the Final Safety Analysis Report. This modification is a line item improvement to the TS as described in Generic Letter 93–08, ‘‘Relocation of the Technical Specification Tables of Instrument Response Time Limits,’’ dated December 29, 1993.

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

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

The purpose of the proposed Tech. Spec. [Technical Specification] change is to delete and subsequently relocate Tech. Spec. Table 3.3.1–2, Table 3.3.2–3, and Table 3.3.3–3, to the SSES FSAR consistent with the guidance provided in Generic Letter 93–08. This is a line item Tech. Spec. improvement change recommended by the NRC in Generic Letter 93–08. This change will allow PP&L [Pennsylvania Power & Light Company] to administratively control subsequent changes to the response time limits in accordance with 10CFR50.59. The procedures that contain the various response time limits are also subject to the change control provisions in the Administrative Controls section of the Tech. Specs. The proposed change only relocates the existing response time limits; the surveillance requirements and associated Actions are not affected and remain in the Tech. Specs. Relocating the response time limit information does not affect the analysis of any design basis accident. The response times of these systems will be maintained within the acceptance limits assumed in SSES [Susquehanna Steam Electric Station] safety analyses and required for successful mitigation of an initiating event. Also, since any subsequent changes to the FSAR or procedures will be evaluated in accordance with 10 CFR 50.59, no increase in the probability or consequences of an accident previously evaluated will occur. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As discussed above, the proposed Tech. Spec. changes do not affect the capability of the associated systems to perform their intended function within the acceptance limits assumed in SSES safety analyses and required for successful mitigation of an initiating event. The proposed change does not involve a physical modification of the plant or changes in methods governing normal plant operations. The proposed change will not impose any different operational or surveillance requirements. This change only proposes to relocate these requirements to other plant documents whereby adequate control of information will be maintained. No new failure modes will be introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumption. The proposed change does not alter the scope of equipment currently required to be OPERABLE or subject to testing, nor does the proposed change affect any instrument setpoints or equipment safety functions. Since any future changes to these requirements in the FSAR or procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will occur. Therefore, the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: May 2, 1995.

Description of amendment request: The amendments would eliminate the manual start for auxiliary feedwater from the Technical Specification for Engineered Safety Feature (ESF) Actuation System Instrumentation. The manual start will be tested during the quarterly pump test. This change is consistent with NUREG–1431, ‘‘Standard Technical Specifications-Westinghouse Plants.’’

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

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

The change to the ESF Actuation Instrumentation specification to eliminate the requirements for manual initiation of the [Auxiliary Feedwater] (AFW) Pumps does not change any operating characteristics of the plant. The change will eliminate unnecessary AFW Pump starts which increase wear on system components. Manual initiation is not credited in the Salem safety analyses. Manual initiation is verified quarterly on a staggered test basis by performance of specification 4.7.1.2.b. Therefore, the proposed changes do not 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.

The proposed technical specification modifications do not change system configurations, plant equipment or safety analyses. Therefore, the proposed modifications will not increase the possibility of a new or different kind of accident from any accident previously identified.

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

The proposed change to the ESF Actuation Instrumentation Specification does not affect the ability of the AFW System to perform its design function. The manual initiation of the AFW Pump is not credited in the Salem safety analyses. Manual initiation is verified quarterly by performance of specification 4.7.1.2.b. Therefore, these changes do not involve a significant reduction in any margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: January 4, 1995 (TS 355).

Description of amendment request: The proposed amendment changes the applicability and surveillance requirements for the intermediate range monitor (IRM), average power range monitor (APRM), and APRM Inoperative Trip functions. The proposed amendment adopts provisions of the Improved Standard Technical Specifications (NUREG–1433).

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

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

The proposed change revises the frequency of functional tests for the IRM and APRM High Flux (15% Scram) Trip Functions and eliminates operability requirements for the IRM, APRM High Flux (15% Scram), and APRM Inoperative Trip Functions in certain modes of operation. The operation of these trip functions is not a precursor to any design basis accident or transient analyzed in the Browns Ferry Updated Final Safety Analysis Report. Therefore, this change does not increase the probability of any previously evaluated accident.

The proposed change will eliminate the requirement to re-perform the functional tests for these trip functions prior to each startup if the test is within its periodicity (once per 7 days). It will also eliminate the operability requirement for the IRM High Flux Trip Function in the Shutdown Mode and IRM, APRM High Flux (15% Scram), and APRM Inoperative Trip Functions during the Refuel Mode except when any control rod is withdrawn from a core cell containing one or more fuel assemblies. The Specifications will still provide for operability of the equipment in Modes where credit is taken in the safety analysis. Therefore, this change does not increase the consequences of any previously evaluated accident.

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

The proposed change to the Technical Specification requirements for the IRM, APRM High Flux (15% Scram) and APRM Inoperative Trip Functions does not involve a modification to plant equipment. No new failure modes are introduced. There is no effect on the function of any plant system and no new system interactions are introduced by this change. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will eliminate the requirement to re-perform the functional test for the IRM and APRM High Flux (15% Scram) Trip Functions prior to each startup if the tests are within their periodicity (once per 7 days). The proposed change will also eliminate operability requirements for modes of operation in which the IRM, APRM High Flux (15% Scram) and APRM Inoperative Trip Functions provide no useful function. Since the ability of the trip functions to perform their safety function will not be degraded, the proposed amendment does not involve a reduction in a margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: March 31, 1995 (TS 349).

Description of amendment request: The proposed amendment changes the reactor pressure vessel pressure-temperature (P–T) curves, lowering the temperature at which the reactor vessel head bolting studs may be tensioned.

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

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

The proposed Units 1, 2, and 3 change deals exclusively with the reactor vessel P–T [pressure-temperature] curves, which define the permissible regions for operation and testing. Failure of the reactor vessel is not a design basis accident. Through the design conservatism used to calculate the P–T curves, reactor vessel failure has a low probability of occurrence and is not considered in the safety analyses. These changes do not alter or prevent the operation of equipment required to mitigate any accident analyzed in the BFN Browns Ferry Nuclear Plant] Final Safety Analysis Report. Therefore, this change does not increase the probability or consequences of any previously evaluated accident.

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

The proposed change to the Units 1, 2, and 3 reactor vessel P–T curves does not involve a modification to plant equipment. No new failure modes are introduced. There is no effect on the function of any plant system and no new system interactions are introduced by this change. The calculation of the proposed P–T curves was in accordance with Regulatory Guide 1.99, Revision 2, and the requirements of 10 CFR 50, Appendix G. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The ductile to brittle transition temperature is shifted approximately 10°F at higher temperatures and approximately 30°F at lower temperatures on the proposed P–T curves. While this represents a decreased margin against non-ductile fracture during heatup, cooldown and hydrotesting, the proposed curves conform to the guidance contained in Regulatory Guide 1.99, Revision 2, and maintain the safety margins specified in 10 CFR 50, Appendix G. Therefore, the proposed amendment 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: Athens Public Library, South Street, Athens, Alabama 35611.

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: May 11, 1995 (TS 359).

Description of amendment request: The proposed amendment adds a scram pilot air header low pressure reactor trip to Browns Ferry Unit 3. The proposed amendment also clarifies a note regarding reactor protection system instrumentation requirements for all three units.

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

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

The scram pilot air header low pressure switches perform the same function as the high water level switches in the scram charge instrument volume. They automatically initiate control rod insertion (SCRAM) in the event that degraded conditions are detected in the BWR [boiling water reactor] CRD [control rod drive] System. Since the scram pilot air header pressure trip function ensures that the CRD System is available to mitigate the consequence of an accident or transient, and the addition of the scram pilot air header low pressure trip scram function does not affect the precursors for any accident or transient analyzed in Chapter 14 of the BFN Updated Final Safety Analysis Report (UFSAR), there is no increase in the probability of any accident previously evaluated.

The design criteria for the scram system is contained in the generic SER [safety evaluation report], which was transmitted by NRC letter to All BWR Licensees, dated December 9, 1980, BWR Scram Discharge System. The scram pilot air header pressure trip function ensures that the CRD System is available to mitigate the consequence of an accident or transient, and the overall scram system design, with the addition of the scram pilot air header low pressure trip function, satisfies the criteria contained in the generic SER. Since the scram function would be successfully performed, the addition of the scram pilot air header low pressure trip scram function does not involve a significant increase in the consequences of any accident previously evaluated.

The clarification of the description of the SDV [scram discharge volume] high water level bypass in the RPS [reactor protection system] does not, by itself, reflect a modification to plant equipment, maintenance activities, or operating instructions. The revised description does not effect the precursors for any accident or transient analyzed in Chapter 14 of the BFN UFSAR or equipment used in the mitigation of these accidents or transients. Therefore, there is no increase in the probability of any accident previously evaluated nor an increase in the consequences of any accident previously evaluated.

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

The scram pilot air header low pressure trip performs the same protective function as the SDV high water level trip. Both trip functions ensure that a reactor scram is initiated while sufficient volume remains in the SDV to accept discharged water from the CRDs.

The scram inlet and outlet valves are held closed by the air pressure in the scram air header. The scram outlet valves begin to unseat as the air pressure drops below 43 psig (which is higher than the pressure that scram inlet valves begin to unseat). The scram pilot air header low pressure switches detect losses in air pressure and initiate an anticipatory scram to ensure the scram is complete prior to the possible onset of hydraulic locking in the SDV. The proposed trip level setting of 50 psig is conservative and assures a trip signal and successful reactor scram is accomplished prior to hydraulic locking occurring in the SDV as a result of significant flow past the scram outlet valves.

The overall scram system design, with the addition of the scram pilot air header low pressure trip function is in conformance with the generic SER. No new system failure modes are created as a result of adding the scram pilot air header low pressure trip scram function. Therefore, the addition of the scram pilot air header low pressure trip scram function does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The clarification of the description of the SDV high water level bypass in the RPS does not, by itself, reflect a modification to plant equipment, maintenance activities, or operating instructions. No new external threats, system interactions, release pathways, or equipment failure modes are created. Therefore, the clarification of this description does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The overall scram system design, with the addition of the scram pilot air header low pressure trip function is in conformance with the generic SER. Since the scram system would successfully operate to mitigate the consequences of accidents and transients previously analyzed, the proposed amendment does not involve a significant reduction in the margin of safety.

The clarification of the description of the SDV high water level bypass in the RPS does not, by itself, reflect a modification to plant equipment, maintenance activities, or operating instructions. There is no change to the licensing or design basis of the RPS. Therefore, the revised description 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: Athens Public Library, South Street, Athens, Alabama 35611.

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

NRC Project Director: Frederick J. Hebdon.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: April 28, 1995.

Description of amendment request: The proposed amendment would remove the license conditions for the Transamerica Delaval, Inc. emergency diesel generators specified by paragraph 2.C.(9) and defined in Attachment 2 to the Operating 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change involves the removal of license conditions associated with teardowns and certain inspections on the Transamerica Delaval, Inc. (TDI) Emergency Diesel Generators (EDG). A failure of an EDG is not an initiating event for any Updated Safety Analysis Report (USAR) Chapter 15 accident scenario. Accordingly, there could be no increase in the probability of any accident previously evaluated. The availability and reliability of the EDGs will remain within the limits previously assumed in the safety analyses. Eliminating the disassembly and specified inspections would actually tend to decrease the consequences of an accident because, as indicated in Topical Report TDI–EDG–001–A, ''Basis for Modification to Inspection Requirements for Transamerica Delaval, Inc., Emergency Diesel Generators,'' this action will improve the availability of the engines for service, especially during outages, while maintaining current reliability levels. Therefore, removal of the existing conditions from the operating license will not result in an increase in the consequences of an accident previously evaluated.

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

The proposed removal of the license conditions associated with the TDI diesel generators does not affect the design or function of any plant system, structure, or component, nor does it change the way plant systems are operated. No modifications or additions to plant equipment are involved. Therefore, removal of the existing conditions from the operating license will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed removal of the EDG license conditions from the Operating License does not affect any parameters which would result in a significant reduction in margin of safety because the results of the operational data and inspections have demonstrated that the additional license conditions are not required to ensure that the EDGs will be maintained with a reliability consistent with that assumed for the safety analyses. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

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

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: May 2, 1995.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Table 15.4.1, ''Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels.'' The radiation monitoring system channel requirements would be deleted, the main steam line radiation channel requirements would be added, and the containment high range radiation channel requirements would be changed. Administrative changes, consistent with the proposed modifications, would also be made.

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 radiation monitors being removed from Table 15.4.1–1 are not directly involved with mitigating an offsite release in the case of an accident. The surveillance requirements for monitors which would measure and mitigate such a release are listed in Technical Specifications Section 15.7.4, ''Radioactive Effluent Monitoring Instrumentation Surveillance Requirements.'' Post-accident radiation monitors will still be included in Table 15.4.1–1. Monitors to be removed include area and non-RETS [radiological effluent technical specification] required process monitors. These are necessary to monitor plant conditions and will still be subject to surveillance requirements. The removed monitors do not have any safety function with regard to radioactive releases. Therefore, the consequences of an accident will not be increased. The radiation monitors are not initiators for any accident analyses in the FSAR, therefore, the probability of an accident previously evaluated is not increased.

2. The proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated. There is no physical change to the facility, its systems, or its operation, therefore, a new or different kind of accident cannot occur.

3. The proposed change will not involve a significant reduction in the margin of safety. The removal of much of the RMS equipment from the Technical Specifications will not affect the surveillance program already in place. The change in test frequency for the post-accident monitoring instrumentation will not have a significant impact on the margin of safety. Test frequencies continue to meet acceptable standards. RETS required effluent monitors, which are of prime importance due to their release mitigation function, are checked quarterly in accordance with Technical Specifications Section 15.7.4, ''Radioactive Effluent Monitoring Instrumentation Surveillance Requirements.'' Therefore, the margin of safety is not 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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

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

NRC Project Director: Gail H. Marcus.

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

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

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

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

Date of amendment request: May 15, 1995.

Description of amendment request: The proposed amendment would authorize a reconfiguration of the cooling water flow to the reactor building emergency cooling system.

Date of individual notice in the **Federal Register**: May 22, 1995 (60 FR 27144)

Expiration date of individual notice: June 21, 1995.

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

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.

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

Date of application for amendments: November 2, 1994.

Brief description of amendments: These amendments delete the condenser vacuum exhaust release point reference on Figure 5.1–3 and combine it with the plant vent exhaust release point on the revised Figure 5.1–3. In addition to the figure change, Bases Section 3/4.3.3.6 is changed to reflect the removal of radiation monitor RU–142 and the relocation of RU–144 and RU–146 from Table 3.3–13 (deleted by amendments 62, 48, and 34, for Units 1, 2, and 3, respectively) to the Offsite Dose Calculation Manual.

Date of issuance: May 25, 1995.

Effective date: May 25, 1995, to be implemented within 45 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 91; Unit 2—Amendment No. 79; Unit 3—Amendment No. 62.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 21, 1994 (59 FR 65810). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 25, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: April 13, 1994, as supplemented December 20, 1994, January 12, January 31, March 17, and April 5, 1995. Brief description of amendment: The amendment revises TS Sections 3.1.F and 4.13 to allow the repair of steam generator tubes by sleeving using laser welded sleeves.

Date of issuance: May 19, 1995.

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

Amendment No.: 183.

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

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27051). The December 20, 1994, January 12, January 31, March 17, and April 5, 1995, submittals provided clarifying information that did not change the initial no significant hazards determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1995. No significant hazards consideration comments received: No.

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

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

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

Date of application for amendment: December 29, 1994, as supplemented February 2 and May 4, 1995.

Brief description of amendment: This amendment revises the iodine removal system Technical Specification (TS) to reflect replacement of the sodium hydroxide requirements with trisodium phosphate requirements. The revised TS defines operability, applicability, and associated action statements for the new system. Associated surveillance requirements and bases have also been revised.

Date of issuance: May 19, 1995.

Effective date: May 19, 1995.

Amendment No.: 165.

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

Date of initial notice in **Federal Register**: February 1, 1995 (60 FR 6299). The February 2 and May 4, 1995, submittals provided clarifying information which was within the scope of the initial application and did not affect the staff's initial proposed no significant hazards consideration findings. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1995. No significant hazards consideration comments received: No.

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

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 application for amendments: April 19, 1994, as supplemented March 31, 1995.

Brief description of amendments: These amendments revise the Appendix A Technical Specifications (TSs) 3.4.9.3 and 3.4.11 to incorporate changes to the power operated relief valve TSs in accordance with the guidance in Generic Letter 90–06, ''Resolution of Generic Issue 70, ''Power-Operated Relief Valve and Block Valve Reliability,'' and Generic Issue 94, ''Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'' Pursuant to 10 CFR 50.54(f),'' as implemented in the NRC's Improved Standard Technical Specifications (NUREG–1431) with some exceptions and modifications to reflect plant-specific design features. The amendment includes several administrative changes (e.g., renumbering sections, spelling out mathematical symbols, changes in nomenclature for consistency, and relocation of sentences and paragraphs).

Date of issuance: May 15, 1995.

Effective date: May 15, 1995.

Amendment Nos.: 187 and 69.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 6, 1994 (59 FR 34661). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 1995. No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 22, 1993.

Brief description of amendment: The amendment revised the value of the Unit 1 reactor building volume as listed in the technical specifications. The amendment was submitted after a more precise calculation of the reactor building volume was completed.

Date of issuance: May 22, 1995.

Effective date: May 22, 1995.

Amendment No.: 181.

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

Date of initial notice in **Federal Register**: December 22, 1993 (58 FR 76843). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 1995. No significant hazards consideration comments received: No.

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

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: June 25, 1993, as supplemented by letter dated April 13, 1995.

Brief description of amendment: This amendment deleted portions of the current Technical Specifications (TSs) Surveillance Requirements (SRs) for the inboard Main Steamline Isolation Valve Leakage Control System (MSIV–LCS) heaters and blowers. The deleted MSIV–LCS SRs will be relocated to documents that are included by reference in the Updated Final Safety Analysis Report (UFSAR) and are controlled by the licensee under the provisions of 10 CFR 50.59. The change is consistent with the format and content of the Improved Standard Technical Specifications (NUREG–1434, Revision O).

Date of issuance: May 22, 1995.

Effective date: May 22, 1995.

Amendment No. 122.

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

Date of initial notice in **Federal Register**: July 21, 1993 (58 FR 39050). The additional information contained in the supplemental letter dated April 13, 1995, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 1995. No significant hazards consideration comments received: No.

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

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: March 18, 1994, as supplemented by letters dated February 28 and March 17, 1995.

Brief description of amendments: The amendments revise Technical Specification (TS) 3/4.3.3.6, Accident Monitoring Instrumentation, TS 3/4.6.4.1, Hydrogen Monitors, and their associated Bases to incorporate the technical substance of Specification 3.3.3 from NUREG–1431, Revision O (Standard Technical Specifications) for the Westinghouse Owners Group.

Date of issuance: May 15, 1995.

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

Amendment Nos.: 85 and 63.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 28, 1994 (59 FR 22008). The February 28 and March 17, 1995, letters provided clarifying information that did not change the scope of the March 18, 1994, application and initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

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: March 16, 1995.

Brief description of amendments: The amendments revised Technical Specification 4.6.1.2, regarding the test frequency requirements for the overall integrated containment leakage rate tests, so that it references 10 CFR part 50, appendix J and approved exemptions, rather than paraphrase the regulation.

Date of issuance: May 19, 1995.

Effective date: May 19, 1995.

Amendment Nos.: Unit 1—Amendment No. 75; Unit 2—Amendment No. 64.

Facility Operating License Nos. NPF–76 and NPF–80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 26, 1995 (60 FR 20517). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 19, 1995. No significant hazards consideration comments received: No.

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

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: April 14, 1995.

Brief description of amendment: The amendment allows the use of the Westinghouse Electric Corporation sleeving process for repairing steam generator tubes.

Date of issuance: May 22, 1995.

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

Amendment No.: 150.

Facility Operating License No. DPR–36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 22, 1995 (60 FR 19969). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

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

Date of amendment request: January 25, 1995, and oral request of May 16, 1995.

Description of amendment request: This amendment revises the Appendix

A Technical Specifications (TS) relating to the schedule for performing Type A containment Integrated Leak Rate Tests (ILRTs). Specifically, the amendment replaces the prescribed number of ILRTs to be performed and the associated schedule with the requirement to conduct ILRTs at intervals as specified in Appendix J to 10 CFR Part 50.

Date of issuance: May 17, 1995.

Effective date: May 17, 1995.

Amendment No.: 37.

Facility Operating License No. NPF–86. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 15, 1995 (60 FR 8754). The licensee's oral request of May 16, 1995, provided a minor clarifying addition, but does not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of amendment request: October 4, 1993.

Description of amendment request: The amendment revises the Appendix A Technical Specifications (TS) relating to A.C. power sources during operation in Modes 1 through 4. Specifically, the amendment deletes the diesel engine speed specification from Surveillance Requirement (SR) 4.8.1.1.2a.5 and replaces the diesel engine speed requirement with an electrical frequency requirement in SR 4.8.1.1.2g.

Date of issuance: May 19, 1995.

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

Amendment No.: 38.

Facility Operating License No. NPF–86. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 2, 1994 (59 FR 4941). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, 47 Front Street, Exeter, NH 03833.

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

Date of amendment request: October 7, 1994.

Description of amendment request: The amendment modifies Technical Specification (TS) 6.4.1.6, 6.4.3.8, and 6.7.1 relating to Administrative Controls. Specifically, the amendment removes certain audit responsibilities of the Nuclear Safety Audit Review Committee and certain review responsibilities of the Station Operation Review Committee relating to the Emergency Plan and the Security Plan and their implementing procedures, and deletes the requirements for written procedures relating to the Emergency Plan and Security Plan.

Date of issuance: May 19, 1995.

Effective date: May 19, 1995.

Amendment No.: 39.

Facility Operating License No. NPF–86. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63125). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of application for amendment: May 6, 1994, supplemented March 27, 1995.

Brief description of amendment: The amendment incorporates additional sections and their associated surveillance requirements and bases into the Millstone Unit 2 TS that impose additional requirements on components that are credited to provide feedwater isolation in the event of a main steam line break inside containment. In addition, the amendment makes modifications to the TS Bases Sections ¾.3.1 and ¾.3.2 by denoting that the feedwater pumps are assumed to trip immediately upon receipt of a main steam line isolation signal; and makes several miscellaneous editorial changes.

Date of issuance: May 17, 1995.

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

Amendment No.: 188.

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

Date of initial notice in **Federal Register**: June 22, 1994 (59 FR 32232). The March 27, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut Date of application for amendment: April 21, 1994.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.1.2.4, ''Charging Pumps-Operating,'' by adding a note that indicates that the provisions of TS 3.0.4 and 4.0.4 are not applicable for entry into MODE 4 from MODE 5.

Date of issuance: May 18, 1995.

Effective date: As of the date of issuance.

Amendment No.: 189.

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

Public comments requested as to proposed no significant hazards consideration: Yes (60 FR 21558, May 2, 1995). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by June 1, 1995, 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 final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 18, 1995.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: December 9, 1994, as supplemented March 28, 1995.

Brief description of amendment: The amendment eliminates certain surveillance requirements for the emergency diesel generators, in accordance with staff guidance contained in Generic Letter 93–05, ''Line Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing during Power Operation,'' dated September 27, 1993.

Date of issuance: May 12, 1995.

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

Amendment No.: 112.

Facility Operating License No. NPF–49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 15, 1995 (60 FR 8749). The March 28, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: January 18, 1995.

Brief description of amendment: The amendment revises the Technical Specifications to increase the minimum required boron concentration in the boric acid tank (BAT) from 6300 to 6600 ppm. The increase is required to meet the latest analysis for Cycle 6 which includes additional conservatisms which are meant to ensure the new required boron concentration will bound future cycle variations.

Date of issuance: May 17, 1995.

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

Amendment No.: 113.

Facility Operating License No. NPF–49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 15, 1995 (60 FR 8753). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: January 24, 1995, as supplemented March 22 and 29, 1995, and April 25, 1995.

Brief description of amendment: The amendment revises Technical Specification 3.2.3.1.a and Table 2.2–1 to reduce the minimum reactor coolant system (RCS) flow rate by 4%, with corresponding changes in loop flow. The current minimum RCS flow rate of 387,480 gallons per minute (gpm) is reduced to 371,920 gpm for four-loop operation.

Date of issuance: May 23, 1995.

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

Amendment No.: 114.

Facility Operating License No. NPF–49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 1, 1995 (60 FR 11136) and April 12, 1995 (60 FR 18626). The April 25, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 23, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota

Date of application for amendments: January 9, 1995, as supplemented February 7, March 15, March 27, April 3, and April 20, 1995.

Brief description of amendments: The amendments revise the Technical Specifications (TS) for the Prairie Island Nuclear Plant to allow using an alternate steam generator tube plugging criteria (F*) for the part of the tubes within the tubesheet. The amendments incorporate revised acceptance criteria (F*) for tubes with degradation in the tubesheet roll expansion and enable the licensee to avoid unnecessary plugging of steam generator tubes. NRC will issue a separate safety evaluation for the L* criteria at a later date.

Date of issuance: May 15, 1995.

Effective date: May 15, 1995, with full implementation within 30 days.

Amendment Nos.: 118/111.

Facility Operating License Nos. DPR–42 and DPR–60. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 15, 1995 (60 FR 14023). The March 15, March 22, April 3, and April 20, 1995, letters provided updated TS pages and clarifying information in response to NRC's requests for additional information. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 1995. No significant hazards consideration comments received: No.

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

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: August 3, 1994.

Brief description of amendments: The amendments implement a snubber functional test surveillance interval of 24 months. The amendments change the current one-time snubber functional test interval to a permanent interval of 24 months.

Date of issuance: May 16, 1995.

Effective date: May 16, 1995.

Amendments Nos.: 201 and 204.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 18, 1995 (60 FR 3676).The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 16, 1995. 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 Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: September 29, 1994.

Brief description of amendments: The amendments remove from the Technical Specifications the sections entitled ''Seismic Instrumentation'' and ''Meteorological Instrumentation'' and relocate the information and testing requirements to the Salem Updated Final Safety Analysis Report.

Date of issuance: May 22, 1995.

Effective date: May 22, 1995.

Amendment Nos. 167 and 149.

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

Date of initial notice in **Federal Register**: November 23, 1994 (59 FR 60385). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 22, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 3, 1993.

Brief description of amendments: These amendments revise Technical Specification (TS) 3/4.6.3, ''Containment Isolation Valves,'' to require valves listed in Section D of existing Table 3.6–1, ''Containment Isolation Valves,'' to be in an action statement when secured in their engineered safety feature actuation system (ESFAS) actuated position. Bases 3/4.6.3 is also revised to reflect these changes.

Date of issuance: May 17, 1995.

Effective date: May 17, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—Amendment No. 119; Unit 3—Amendment No. 108.

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

Date of initial notice in **Federal Register**: February 16, 1994 (59 FR 7699). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 17, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 25, 1994.

Brief description of amendments: The amendments revise the NA–1&2 Hydrogen Recombiner System surveillance requirements in accordance with Generic Letter 93–05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' Also, the amendments delete the surveillance requirement to operate the containment purge blower and clarifies that the surveillance requirement applies only to the hydrogen recombiner purge blowers.

Date of issuance: May 12, 1995.

Effective date: May 12, 1995.

Amendment Nos.: 192 and 173.

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

Date of initial notice in **Federal Register**: November 23, 1994 (59 FR 60388). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 12, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 2, 1995.

Brief description of amendments: The amendments revise the NA–1&2 Technical Specification 4.6.1.2.a to permit approved exemptions to the containment integrated leak rate test frequency requirements.

Date of issuance: May 15, 1995.

Effective date: May 15, 1995.

Amendment Nos.: 193 and 174.

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

Date of initial notice in **Federal Register***:* April 12, 1995 (60 FR 18629). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 2, 1992.

Brief description of amendment: The amendment revises Figure 3.1.5–2, ''Sodium Pentaborate Tank, Volume Vs. Concentration Requirements,'' to reflect the actual low-volume-alarm and low-limit values for the standby liquid control tank.

Date of issuance: May 17, 1995.

Effective date: May 17, 1995, to be implemented within 30 days of issuance.

Amendment No.: 138.

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

Date of initial notice in **Federal Register**: November 23, 1994 (59 FR 60388). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 1995. No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 21, 1995.

Brief description of amendment: This amendment revises Technical Specification Surveillance Requirement 4.6.2.1.d, ''Containment Spray System,'' to change the surveillance interval specified for the performance of an air or smoke flow test through the containment spray header from at least 5 years to at least once per 10 years.

Date of issuance: May 17, 1995.

Effective date: May 17, 1995, to be implemented within 30 days of issuance.

Amendment No.: 86.

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

Date of initial notice in **Federal Register***:* April 12, 1995 (60 FR 18631). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 1995. No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 31st day of May, 1995.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–13759 Filed 6–5–95; 8:45 am]

BILLING CODE 7590–01–P

[Docket No. 50–389A; DD–95–10]

Florida Power & Light Company' St. Lucie Plant, Unit #2; Issuance of Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC), has issued the Director's Decision concerning the petition dated July 2, 1993, filed by Robert A. Jablon, Esq., et. al, on behalf of the Florida Municipal Power Agency (petitioner). The petitioner requested that the NRC take certain enforcement actions against the Florida Power & Light Company (FPL) for allegedly violating the antitrust license conditions applicable to Unit 2 of the St. Lucie plant.

After consideration and careful review of the facts available to the staff and the decision reached in a parallel proceeding involving the same parties and similar issues before the Federal Energy Regulatory Commission (FERC), the Director has determined that the issues raised by the petitioner that could be remedied by the NRC have addressed and resolved in the FERC proceeding(s) so as to require no further action by the NRC. As a result, no proceeding in response to the petition will be instituted. The reasons for this decision are explained in the ''Director's Decision under 10 CFR 2.206'' (DD–95–10), which is published below.

A copy of the Director's Decision has been filed with the Secretary of the Commission for Commission review in accordance with 10 CFR 2.206(c). The Decision will become the final action of the Commission 25 days after issuance, unless the Commission on its own motion institutes review of the Decision within that time as provided in 10 CFR 2.206(c).

Copies of the Petition, dated July 2, 1993, and the Notice of Receipt of Petition for Director's Decision under 10 CFR 2.206 that was published in the **Federal Register** on September 23, 1993 (58 FR 47919), and other documents related to this Petition are available in the NRC Public Document Room, the Gelman Building, 2120 L Street NW. (Lower Level), Washington, DC 20555 and Local Public Document Room at the Indian River Community College, 3209 Virginia Avenue, Ft. Pierce, FL 33450.

Dated at Rockville, Maryland, this 26th day of May 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 95–13758 Filed 6–5–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket Nos. 50–213, 50–245, 50–336, 50–423]

Northeast Utilities; Haddam Neck Plant and Millstone Nuclear Power Station, Units 1, 2, 3; Issuance of Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has taken action with regard to a Petition dated March 3, 1994, by Mr. Ronald Gavensky (Petition for action under 10 CFR 2.206). The Petition pertains to the Haddam Neck Plant and Millstone Nuclear Power Station, Units 1, 2, and 3.

In the Petition, Petitioner, a quality control receipt inspector raises, numerous concerns regarding receipt inspection activities by Northeast Utilities at both the Haddam Neck Plant and Millstone Nuclear Power Station, Units 1, 2, and 3, Petitioner alleges violations of 10 CFR Part 50, Appendix B, by Northeast Utilities in the receipt inspection area. Petitioner alleges that parts represented as having been inspected and accepted for use were in fact deficient. Petitioner alleges that adequate training, skilled personnel, and necessary tools were not available to perform adequate receipt inspections. Petitioner alleges that he observed unethical and incorrect methods of receipt inspection, and that he sought to identify quality problems within his own department, along with recommendations and solutions, but was not permitted to do so. Finally, Petitioner accuses Northeast Utilities of ''white washing'' his concerns in the receipt inspection area. Petitioner alleges that, on two occasions, Northeast Utilities' management hired investigators to pursue concerns raised by Petitioner only to conclude that there were no problems. Petitioner requests that the licenses of Northeast Utilities be temporarily revoked until after the NRC conducts an investigation of Petitioner's allegations.

The Director of the Office of Nuclear Reactor Regulation has determined to deny the Petition. The reasons for this denial are explained in the ''Director's Decision Pursuant to 10 CFR 2.206'' (DD–95–11), the complete text of which follows this notice, and 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 rooms located at the Russell Library, 123 Broad Street, Middletown, CT 06457 for the Haddam Neck Plant, and at the Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360, for Millstone Nuclear Power Station, Units 1, 2, and 3.

A copy of the Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c) of the Commissions regulations. As provided by this regulation, the Decision will constitute the final action of the Commission 25 days after the date of issuance unless the Commission on its own motion institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 31st day of May 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

I. Introduction

On March 3, 1994, Mr. Ronald Gavensky (Petitioner) filed a Petition with the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 2.206. In the Petition, the Petitioner, a Northeast Utilities (NU) quality control inspector raised concerns regarding receipt inspection activities by NU at the Haddam Neck Plant and the Millstone Nuclear Power Station.[1]

The Petitioner alleged violations of 10 CFR Part 50, Appendix B, by NU in the receipt inspection area. He alleged that parts represented as having been inspected and accepted for use were in

[1] Northeast Nuclear Energy Company (Millstone licensee), an electric operating subsidiary of Northeast Utilities (NU), holds licenses for the operation of Millstone Nuclear Power Station, Units 1, 2, and 3. The Connecticut Yankee Atomic Power Company (Haddam Neck licensee), an electric operating company owned in part by NU, holds the license for the Haddam Neck Plant. Reference in the Petition to the ''license of Northeast Utilities'' refers to the licenses of the Haddam Neck Plant and Millstone Nuclear Power Station, Units 1, 2, and 3.