determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

Statement of Policy

The Commission's policy on Conversion to the Metric System remains essentially as stated in the Federal Register (57 FR 46202) of October 7, 1992.

The NRC supports and encourages the use of the metric system of measurement by licensed nuclear industry. In order to facilitate the use of the metric system by licensees and applicants, beginning January 7, 1993, the NRC will publish the following documents in dual units: New regulations, major amendments to existing regulations, regulatory guides, NUREG-series documents, policy statements, information notices, generic letters, bulletins, and all written communications directed to the public.

Documents specific to a licensee, such as inspection reports and docketed material dealing with a particular licensee, will be in the system of units employed by the licensee. This protocol reflects a general approach that only documents applicable to all licensees, or to all licensees of a given type in which a licensee may operate in the metric system will contain dual units. Otherwise, English or metric units alone are permissible. In dual-unit documents, the first unit presented will be in the International System of Units with the English unit shown in brackets. The NRC will modify existing documents and procedures as needed to facilitate use of the metric system by licensees and applicants. In addition, the NRC will provide staff training as needed. Further, through its participation in national, international, professional, and industry standards organizations and committees and through its work with other industry organizations and groups, the NRC will encourage and further the use of the metric system in formulating and adopting standards and policies for the licensed nuclear industry.

However, if the NRC concludes that the use of any particular system of measurement would be detrimental to the public health and safety, the Commission will proscribe the use of that system by regulation, order, or other appropriate means. In particular, all event reporting and emergency response communications between licensees, the NRC, and State and local authorities will be in the English system of measurement. Further, the NRC will follow the Federal Acquisition Regulation and the General Services Administration metrication program in executing procurements. Lastly, the Commission considers this policy final and conversion to the metric system complete. The Commission does not intend to revisit this policy unless it is causing an undue burden or hardship.

Dated at Rockville, Maryland, this 12th day of June 1996.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

[FR Doc. 96–15397 Filed 6–17–96; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 24, 1996, through June 7, 1996. The last biweekly notice was published on June 5, 1996 (61 FR 28604).

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

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

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

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

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

By July 19, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: April 25, 1996

Description of amendment request: The proposed amendment would change the definition of Operable-Operability, revise Technical Specifications (TSs) and associated Bases Section for TSs 3.5.F.1, ''Core and Containment Cooling systems,'' TSs 3.9.B.1, 3.9.B.2, 3.9.B.3, 3.9.b.4, ''Auxiliary Electrical System,'' and TSs 3.7.B.1.a, c, and e, and 3.7.b.2.a, c, and e, ''Standby Gas Treatment System and Control Room High Efficiency Air Filtration System,'' and delete TSs 4.5.F.1, ''Core and Containment Cooling Systems,'' and 3.7.B.1.f, ''Standby Gas Treatment System and Control Room High Efficiency Air Filtration System.''

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

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

Operation of PNPS [Pilgrim Nuclear Power Station] in accordance with the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

Definition of ''Operable-Operability''

Definitions perform a supporting function for other sections of the TS. The definition of ''Operable-Operability'' affects the manner in which the requirements for a Limiting Condition for Operation (LCO) and its associated remedial actions are applied when a support system is inoperable. This definition re-affirms the principle that a system is operable when it is capable of performing its specified function and when all necessary support systems are also capable of performing their related support functions. The corollary is that a system is inoperable when it is not capable of performing its specified function or when a necessary support system is not capable of performing its related support function.

No changes are being made to the plant design, system configuration, or method of operation. The proposed change does not affect the ability of the AC power sources to perform their required safety functions nor affect the ability of the features they support to perform their respective safety functions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

EDG [Emergency Diesel Generator]

An Individual Plant Examination (IPE) for Internal Events was submitted to the NRC in response to Generic Letter 88–20 in September 1992. The IPE was used to quantify the overall impact of the proposed 14 day allowed outage time on core damage frequency. Part III provides the results of a comprehensive Probabilistic Safety Assessment (PSA) of the impact of the proposed AOTs [allowed outage times] for the EDGs and Startup and Shutdown transformers. As shown in Part III, there is not a significant increase in risk due to the proposed change. Thus the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The existing specification 3.9.B.1 is being separated into two segments (a and b) because of the proposed and different AOTs for the Startup and Shutdown transformers. As a result of the PSA, the AOT for the Startup transformer (a) is reduced from 7 days to 72 hours, while the AOT for the Shutdown transformer (b) remains at 7 days. The reduction of the AOT from 7 days to 3 days is based on the relative risk importance of the Startup transformers support to the balance of plant systems. Similarly, an additional reduction from 72 hours to 48 hours is proposed in the AOT for a simultaneous loss of both the Startup transformer and an EDG (TS 3.9.B.4.b) based upon the Startup transformer's contribution to risk in relation to the EDG 14-day AOT risk assessment analysis and that two power sources have been removed from the associated bus. The AOT reductions represent a measurable decrease in risk as assessed in the PSA. Thus, the probability or consequences of an accident previously evaluated are not significantly increased.

The current technical specifications allow one EDG to be out of service for three days based on the availability of the SUT [startup transformer] and SDT [shutdown transformer] and the fact that each EDG carries sufficient engineered safeguards equipment to cover all design basis accidents. With one EDG out of service and a Loss of Offsite Power (LOOP) condition, the capability to power vital and auxiliary system components remains available via the other EDG, and for one train of ESF equipment via the SDT for all operating, transient and accident conditions. Increasing the EDG AOT to 14 days provides flexibility in the maintenance and repair of the EDGs. The EDG unavailability will be monitored and trended in accordance with the Maintenance Rule. The PSA analyses supports the change to a 14 day AOT for the EDGs based on an insignificant increase in overall risk. Implementation of the proposed change is expected to result in less than a one percent increase in the baseline core damage frequency (2.84E–05/yr), which is considered to be insignificant relative to the underlying uncertainties involved with probabilistic safety assessments. Additional conditions are added to the Standby Liquid Control, Standby Gas Treatment, and Control Room High Efficiency Air Filtration systems requiring the EDG associated with these systems to remain operable while in the 14 day EDG AOT. Thus, the 14 day EDG AOT does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Eliminating the 4.5.F.1 requirement for daily testing of the operable diesel generator when the redundant diesel generator becomes inoperable is consistent with the guidance provided in Generic Letter 93–05. The change does not affect the ability of the emergency diesel generator to perform on demand, and by actually lowering the number of demands to demonstrate operability, reduces the probability of equipment failure. The redundant EDG will remain in service during the entire period of inoperability of the out-of-service EDG. If a common cause failure cannot be ruled out, the redundant EDG will be tested to assure operability. The proposed revisions do not involve a significant change to the plant design or operation, only to the manner in which remaining equipment is confirmed to be operable, which is consistent with NRC guidance. Thus operation of PNPS in accordance with the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The 3.9.B.1 and 2 requirements to demonstrate both EDGs and associated emergency buses operable are deleted. This change is based on the NRC guidance provided in item 10.1 of Generic Letter 93–05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' Revising the methods for verifying EDG and emergency bus operability does not physically alter the plant or have an affect on the probability or consequences of an accident previously evaluated. Deleting the testing requirements for an EDG when the other EDG is inoperable does not increase the probability or consequences of an accident previously evaluated because the reliability program and routinely performed TS surveillances continue to provide the added assurance sought by the testing. The elimination of this testing will serve to improve the overall reliability of the EDGs. Since the proposed change does not affect the design or negatively affect the performance of the EDGs, the change will not result in a significant increase in the consequences or probability of an accident previously analyzed.

SGT [Standby Gas Treatment] and CRHEAF [Control Room High Efficiency Air Filtration]

During normal plant operation, with one SGT or CRHEAF subsystem inoperable, the inoperable subsystem must be restored to operable status in 7 days. In this condition, the remaining operable SGT or CRHEAF subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the operable subsystem could result in the radioactivity release control function not being adequately performed. The 7 day completion time is based on consideration of such factors as the availability of the operable redundant SGT subsystem and the low probability of a DBA [design basis accident] occurring during this period.

If the SGT or CRHEAF subsystem cannot be restored to operable status within 7 days when in the Run, Startup, or Hot Shutdown MODE, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least Hot Shutdown within 12 hours and to Cold Shutdown within 36 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Current TS governing refueling operations restrict fuel movement if one train of SGTS or one train of CRHEAF are inoperable. In this condition the remaining operable SGT and CRHEAF trains are adequate to perform the required radioactivity release control functions. However, the overall system reliability is reduced because a single failure in the operable train could result in the radioactivity release control function of the systems not being adequately performed. New requirements are added that require if one train of SGT or CRHEAF is inoperable, the redundant train of SGT or CRHEAF must be demonstrated to be operable within 2 hours. This substantiates the availability of the operable trains. Fuel handling is limited only to the following 7 days and if the inoperable train is not returned to an operable condition within that time frame, the operable SGT train is placed in operation or fuel handling activities are suspended. For CRHEAF, after 7 days, the operable subsystem is demonstrated operable in accordance with existing surveillances on a daily basis. The proposed changes do not modify system design, use, or configuration in a manner different from their original design and therefore do not involve a significant increase in the consequences or probability of an accident previously analyzed.

The revisions to make the SGT and CRHEAF TS sections similar in wording are made to enhance usability and alleviate possible confusion. These changes are strictly editorial, have no impact, and do not alter technical content or meaning of the specifications. These editorial changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The operation of PNPS in accordance with the proposed license amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated because of the following:

Definition of ''Operable-Operability''

The revised definition redefines the AC power needs to allow either onsite or offsite power available for systems/subsystems to be considered operable. This does not compromise the level of safety already afforded to such systems/subsystems because the functional operability requirements continue to be assured through the technical specifications applicable to such systems/subsystems. AC power availability continues to be assured through existing and proposed surveillances and action statements applicable to AC power systems. Reducing the need for both onsite and offsite power sources in order to consider operable, the systems/subsystems powered by these AC power sources, provides additional operational flexibility by allowing redundant systems/subsystems to still be considered ''operable'' within the requirements of their functional operability requirements. No new change or modes of plant operation are involved. Therefore, operation in accordance with the revised definition does not introduce any new or different kind of accident from any accident previously evaluated.

EDG

The proposed amendment will extend the action completion/allowed outage time for an inoperable emergency diesel generator from 72 hours to 14 days. The EDGs are designed as backup AC power sources for essential safety systems in the event of loss of offsite power. The proposed AOT does not change the conditions, operating configurations or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. The EDGs and AC equipment are not accident initiators. No change is being made in the manner in which the EDG's provide plant protection. No new modes of plant operation are involved. An extended AOT for one EDG does not increase the probability of occurrence of a new or different kind of accident previously evaluated. The PSA results concluded that the risk contribution of the EDG AOT extension is insignificant.

The current Pilgrim Technical Specifications requiring immediate and daily testing of the redundant operable EDG is based on the assumption that the increased testing provides additional assurance that the equipment is available should it be needed. Industry experience indicates that repetitive testing can place demands and wear on the EDG without necessarily providing additional confidence of availability. Also, the new surveillance requires verification that offsite power is available and that a common cause failure is not present. These actions provide assurance that the required emergency buses can be energized with no loss of functions to mitigate accident or transient conditions. In addition, Pilgrim has implemented an EDG reliability program to maintain reliability of EDGs. The proposed change does not introduce any new mode of plant operation or new accident precursors, involve any physical alterations to plant configurations, or make changes to system set points that could initiate a new or different kind of accident. Therefore, operation in accordance with the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The AOT for an inoperable Startup Transformer is reduced from 7 days to 72 hours based upon the PSA that was performed to quantitatively assess the risk impact of the proposed amendment. The proposed reduction in AOT improves overall AC power source availability because the SUT will potentially be inoperable for shorter time periods. Therefore, reducing the AOT does not create the possibility of a new or different kind of accident from any accident previously evaluated.

SGT and CRHEAF

The SGT system is designed to filter radioactive materials from the secondary containment following a postulated DBA or fuel handling accident prior to release to the environment to ensure compliance with 10 CFR 100 limits.

The CRHEAF is designed to filter intake air for the control room atmosphere during conditions when normal intake air may be contaminated.

The proposed revisions do not affect the ability of the SGTS or CRHEAF to perform their intended function, do not create the possibility of a new or different kind of accident from the loss of coolant or fuel handling accidents previously analyzed, and do not modify system configuration, use, or design. Therefore, operating Pilgrim in accordance with this change will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The revisions to make the SGT and CRHEAF TS sections similar in wording are made to enhance usability and alleviate possible confusion. These changes are strictly editorial, have no impact, and do not alter technical content or meaning of the specifications. These editorial changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The operation of PNPS in accordance with the proposed license amendment will not involve a significant reduction in a margin of safety because of the following:

Definition of ''Operable-Operability''

The implementation of the ''Operability'' definition clarifies the relationship between AC power supplies and the operability status of the equipment requiring AC power. No change is being made in which the plant systems relied upon in the safety analyses provide plant protection. Plant safety margins are maintained through the limitations established in the TS LCOs. Since there will be no significant reduction to the physical design or operation of the plant there will be no significant reduction to any of these margins.

EDG

Operation of PNPS in accordance with the proposed license amendment will not involve a significant reduction in a margin of safety. As shown in Part III [of the application dated April 25, 1996], incorporation of the proposed change involves an insignificant reduction in the margin of safety.

The proposed changes do not significantly reduce the basis for any technical specification related to the establishment of, or the maintenance of, a safety margin nor do they require physical modifications to the plant. Additional conditions are added to the Standby Liquid Control, Standby Gas Treatment, and Control Room High Efficiency Air Filtration systems requiring the diesel generator associated with the redundant operable trains of these systems to remain operable while in the 14 day EDG AOT. Moreover, the PSA results showed that the risk contribution of extending the AOT for an inoperable EDG is insignificant. The reduction in the AOT for the SUT could improve availability, therefore, reducing overall risk. Likewise the proposed changes in the deletion of testing have no impact on the safety margin.

As previously stated, implementation of the proposed changes is expected to result in an insignificant increase in: (1) power unavailability to the emergency buses (given that a loss of offsite power has occurred), and (2) core damage frequency. Implementation of the proposed changes does not increase the consequences of a previously analyzed accident nor significantly reduce a margin of safety. Functioning of the EDGs and the manner in which limiting conditions of operation are established are unaffected.

SGT and CRHEAF

SGT and CRHEAF contribute to the margin of safety by supporting the secondary containment system during fuel handling by mitigating the consequences of a fuel handling event. Allowing fuel movement to continue as established in the LCOs does not involve a significant reduction in the margin of safety because the first line of defense, the other SGT and CRHEAF trains will be operable. The proposed change will allow placing the Operable SGT subsystem in operation, or in the case of CRHEAF, conducting daily testing, as an alternative to suspending movement of irradiated fuel. This alternative is less restrictive than the existing requirement, however, the proposed requirements ensure that the remaining subsystem is operable, that no failures that could prevent actuation have occurred, and that any failure would be readily detected. The proposed change does not result in a significant reduction in a margin of safety because it allows operations which have the potential for releasing radioactive material to the secondary containment to continue only if the system designed to mitigate the consequences of this release is functioning. Proper operation of only one SGT or one CRHEAF subsystem is sufficient to mitigate the consequences of any analyzed accident. Therefore, this change does not change any of the assumptions in the accident analysis and does not involve a significant reduction in a margin of safety.

The revisions to make the SGT and CRHEAF TS sections similar in wording are made to enhance usability and alleviate possible confusion. These changes are strictly editorial, have no impact, and do not alter technical content or meaning of the specifications. These editorial changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Jocelyn A. Mitchell, Acting

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: April 22, 1996

Description of amendment request: The licensee is proposing to change the technical specifications to reflect a revision to the overload cutoff limit on the manipulator crane inside the containment at the Haddam Neck Plant. Due to a change in fuel design and supplier, the heaviest fuel assembly design starting in Cycle 20 will be the Westinghouse-supplied LOPAR design. Therefore, the heaviest combination beginning in Cycle 20 will be the Westinghouse LOPAR fuel assembly with a full-length rod cluster control assembly (RCCA) inserted. It will now be used as the standard for the overload cutoff limit on the manipulator crane.

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 will revise the method of determining the overload cutoff limit for the manipulator crane. The actual absolute value of the cutoff limit will not be increased and will not affect the [probability] of any plant accidents.

Since there is no actual increase in the absolute overload cutoff limit, there will be no adverse effects to the crane, cables, or associated hardware. Therefore, there is no impact on the crane's ability to perform its intended function. Even though the net lifting forces on an individual assembly have increased 25 pounds, the limit is within the recommended Westinghouse guidelines with respect to fuel handling and will not result in potential damage to assembly grids during fuel handling activities.

As such, CYAPCO [Connecticut Yankee Atomic Power Company] has concluded that these changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The changes conservatively revise the method of determining the overload cutoff limit for the manipulator crane. There is no impact on the basic functioning of plant systems or equipment. Therefore, the change does not create a malfunction that is different from those previously evaluated.

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

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

The proposed revisions in the methodology for determining the overload cutoff limit for the manipulator crane is conservative and in accordance with vendor standards. The changes do not adversely affect any equipment credited in the safety analysis. Also, the changes do not adversely affect the probability or consequences of any plant accident, including the fuel handling accident or offsite doses associated with those accidents.

As such, the proposed changes have no significant impact on 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: Russell Library, 123 Broad Street, Middletown, CT 06457

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: December 14, 1995, as supplemented by letter dated May 16, 1996

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) to improve the TS Action Statements and Surveillance Requirements for diesel generators in accordance with the recommendations and guidance in Generic Letter 93–05, Generic Letter 94–01, NUREG–1366, and NUREG–1431. The proposed amendments would also incorporate technical and administrative changes.

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

Criterion 1

Operation of the facilities in accordance with the requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Improvements to the LCOs [limiting condition for operation] and surveillance requirements for the emergency diesel generators do not affect their capability to provide emergency power to plant vital instruments and safety related equipment. In fact, these improvements make the diesel generators more reliable since they significantly reduce the amount of wear and stress due to excessive and unnecessary testing. The proposed monthly testing of the diesel generator continues to ensure that the system is ready for service when needed. The fast starts and fast loadings continue to ensure that the timing and loading requirements for engineered safety features actuation are met. The proposed changes do not affect any of the design basis accident analyses previously evaluated. Therefore, these proposed changes do not involve any increase in the probability or consequences of any accident previously evaluated. The proposed changes are fully consistent with the recommendations and guidance contained in GL [Generic Letter] 93–05, GL 94–01, NUREG–1366, NUREG–1431, and are compatible with plant operating experience.

Criterion 2

Operation of the facilities in accordance with the requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes in fact improve the reliability of the diesel generators by eliminating unnecessary wear and stress. Improved reliability decreases the failure probability which also decreases the probability of an accident not previously evaluated. None of the requested amendments increase the common mode failure probability thus would not increase the chance of both EDG's [emergency diesel generators] for a particular nuclear unit being out of service simultaneously. The proposed changes are fully consistent with the recommendations and guidance contained in GL 93–05, GL 94–01, NUREG–1366, NUREG–1431, and are compatible with plant operating experience.

Criterion 3

Operation of the facilities in accordance with the requested amendments will not involve a significant reduction in a margin of safety. The proposed monthly testing of the diesel generators continues to ensure that the system is ready for service when needed. The fast starts and fast loadings continue to ensure that the timing and loading requirements for engineered safety features actuation are met. The proposed changes improve the reliability of the diesel generators. Implementation of the Maintenance Rule also ensures continued reliability of the diesel generators. No margin of safety is decreased as a result of these TS changes.

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

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

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Entergy Operations, Inc., et al., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi and Docket No. 40–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: April 18, 1996, as supplemented by letter dated June 4, 1996

Description of amendment request: The licensee has proposed to (1) amend Limiting Condition for Operation (LCO) 3.10.6 and Surveillance Requirement 3.10.6.3, and (2) add a Surveillance Requirement 3.10.6.4 of the Technical Specifications (TSs) for the Grand Gulf Nuclear Station, Unit 1, and the River Bend Station, Unit 1, to allow another method of fuel movement and loading in the core when control rods are removed or withdrawn from defueled core cells. Currently, LCO 3.10.6 allows only fuel loading as part of the approved spiral reloading sequence to prevent fuel loading into core cells in which the control rod has been removed or withdrawn. This amendment request does not withdraw this approved method, revise the frequency of performing the surveillance during fuel loading, or alter the method of verifying the fuel is being loaded in compliance with the approved method. Grand Gulf Unit 1 and River Bend Unit 1 are both General Electric (GE) Boiling Water Reactor (BWR)-6 plants, the latest version of the GE design series.

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:

Entergy Operations, Inc. [(EOI)] propose[d] to change the current Grand Gulf Nuclear Station (GGNS) and River Bend Station (RBS) Technical Specifications [(TSs)]. The specific proposed change is to add an additional method of performing fuel loading into LCO 3.10.6, ''Multiple Control Rod Withdrawal - Refueling''. The proposed change would allow fuel loading [in the core] if a positive means of assuring fuel assemblies cannot be loaded into a core cell with a withdrawn or removed control rod is in effect. [Currently, the TSs for both plants allow fuel assembles to be loaded in compliance with an approved spiral reload sequence which is used to ensure the reactivity additions are minimized. Spiral loadings encompass reloading a core cell on the edge of a continuous fueled region.]

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Entergy Operations, Inc. [EOI] has evaluated the no significant hazards consideration in its request for this license amendment and determined that no significant hazards consideration results from this change. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. [EOI] is providing the analysis of the proposed amendment against the three standards in 10 CFR 50.92(c). A description of the no significant hazards consideration determination follows:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The refueling interlocks (i.e., the refueling equipment and one-rod-out interlocks) allowed to be bypassed by Technical Specification [TS] LCO 3.10.6 are explicitly assumed in the analysis of the control rod removal error or fuel loading error during refueling. This analysis evaluates the consequences of control rod withdrawal during refueling. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion of fuel, provided all control rods are fully inserted during the fuel insertion. The refueling interlocks accomplish this by preventing loading fuel into the core with any control rod withdrawn, or by preventing withdrawal of a rod from the core during fuel loading.

LCO 3.10.6 allows multiple control rod withdrawals, control rod removals, associated control rod drive (CRD) removal, or any combination of these, and the ''full in'' position indication input to the refueling interlocks is allowed to be bypassed for each withdrawn control rod if all fuel has been removed from the cell. This supports the GGNS Updated Final Safety Analyses Report (UFSAR) and RBS Updated Safety Analyses Report (USAR) analyses since, with no fuel assemblies in the core cell, the associated control rod has no reactivity control function and does not need to remain inserted. Prior to reloading fuel into the cell, however, the associated control rod must be inserted to ensure that an inadvertent criticality does not occur, as evaluated in the analysis.

The Technical Specification [TS] requirements prohibiting fuel loading was placed in the Technical Specifications [TSs] for GGNS and RBS as part of the originally enforced Technical Specification [TS] requirements to resolve NRC concerns identified in IE Information Notice No. 83–35, ''Fuel Movement with Control Rods Withdrawn at BWRs,'' (IEN 83–35). IEN 83–35 details instances where fuel assemblies were loaded into core cells while the control rod was withdrawn and discusses that the General Electric Company (GE) had issued Service Information Letter (SIL) No. 372.

SIL No. 372 discusses a potential event where 8 fuel assemblies are loaded into 2 [two] adjacent core cells where the control rods are withdrawn and no action is taken to recover from the errors. In this SIL GE identified that the probability of such an event occurring was extremely low but potentially slightly higher than 10^{-6} probability of the event even further to where it need not be considered credible (i.e., below 10^{-6} per reactor year), GE recommended that the additional administrative control of prohibiting loading fuel with withdrawn rods be enforced.

The proposed change will only provide an additional way to meet the intent of the original GE recommendation. [The currently approved method is listed in LCO 3.10.6 and Surveillance Requirement 3.10.6.3.]. The proposed change will provide the additional allowance to perform fuel loading only if an additional positive means of assuring fuel assemblies cannot be loaded into a core cell with a withdrawn or removed control rod is in effect. The positive means will entail a physical barrier such that, even if refueling procedures were violated and an attempt was made to load a fuel assembly into a core cell with a withdrawn or removed control rod, the action would be prevented. This requirement provides sufficient additional restrictions to meet the intent of the GE recommendation to add additional administrative controls to prevent the postulated event from occurring.

The probability of an inadvertent criticality occurring will continue to be precluded by the same number of layers of administrative controls [as the currently approved method]; therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The administrative changes in the Technical Specification [TS] requirements do not involve a change in the design of the plant. The proposed requirements will continue to ensure that fuel is not loaded into a core cell that is associated with a removed or withdrawn control rod.

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

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

The margin of safety associated with criticality events during fuel handling is provided by the event being a non credible event. The proposed change will only provide an additional means to meet the same intent of ensuring that the event is of such low probability as to be considered non credible. The proposed change will provide the additional allowance to perform fuel loading only if an additional positive means of assuring fuel assemblies cannot be loaded into a core cell with a withdrawn or removed control rod is in effect. The positive means will entail a physical barrier such that even if refueling procedures were violated and an attempt was made to load a fuel assembly into a core cell with a withdrawn or removed control rod the action would be prevented. This requirement provides sufficient additional restrictions to ensure that the event is of such low probability as to be considered non credible.

The probability of an inadvertent criticality occurring will continue to be precluded by the same number of layers of administrative controls [as the currently approved method]; therefore, this change does not reduce the level of safety imposed by the current Technical Specification [TS] requirements.

Therefore, the proposed changes do not cause 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: (1) Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120, for Grand Gulf Nuclear Station and (2) Government Documents Department, Louisiana State University, Baton Rouge, LA 70803, for River Bend Station.

Attorney for licensee: (1) Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005–3502, for Grand Gulf Nuclear Station and (2) Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005, for River Bend Station.

NRC Project Director: William D. Beckner

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

Date of amendment request: May 9, 1996

Description of amendment request: The amendment request would allow allow the licensee to perform the surveillance of the relief mode of operation of each of the 20 safety/relief valves (S/RVs) on the 4 main steam lines without physically lifting the disk off the seat at power. The proposed changes are to Surveillance Requirements (SRs) 3.4.4.3, Safety/Relief Valves, 3.5.1.7, Automatic Depressurization System Valves, and 3.6.1.6.1, Low-Low Set Valves, of the Technical Specifications, and the changes would state that the required operation of the valve to verify is that the relief-mode actuator strokes when the valve is manually actuated. Each S/RV is a Dikkers, 8 X 10, direct-acting, spring loaded, safety valve with attached pneumatic actuator for relief-mode operation. Eight of the S/RVs use the relief mode to perform the Automatic Depressurization System (ADS) function. Also, six S/RVs, two of which are also ADS S/RVs, use the relief mode to perform the Low-Low Set valve function. The licensee also proposed changes to the Bases of the Technical Specifications that are associated with the above proposed changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below: The Dikkers S/RV provides pressure relief based on the principle of vertically moving the stem that attaches directly to the valve disk. The force that provides the stem movement is provided by one of two sources; the vessel pressure directly against the force of the stem spring (safety mode), or the pneumatic actuator arm against the force of the stem spring (relief mode). ASME Boiler and Pressure Vessel Code requires testing the safety mode of operation once every five year operating cycle. Once a safety valve is installed, the safety mode is never tested while the S/RV is installed in the plant. The testing of the relief mode of operation for a direct-acting S/RV provides verification that the control functions of electrical and pneumatic connections have been properly reconnected, and that the actuator arm will provide the necessary force to operate the S/RV.

This proposed change provides verification of proper control connections by requiring the pneumatic and electrical controls to cycle the actuator arm on each S/RV after installation in the drywell. The test population of S/RVs removed each outage for safety setpoint testing will be tested in the relief mode. This testing will demonstrate that the installed S/RVs will function properly in the relief mode. The remaining installed S/RVs will continue to be tested for proper system function. As presently required by GGNS Technical Specifications and administrative procedures, proper operation of the solenoid control block will be demonstrated by providing an open signal to each S/RV, with a check to verify that each solenoid valve repositions. Verification of proper solenoid valve operation, in addition to the proper relief-mode operation of the test population, provides assurance that the S/RV will perform as expected when control air pressure is applied to the solenoid valve control block.

Entergy Operations, Inc. is proposing that the Grand Gulf Nuclear Station Operating License be amended to perform the surveillance of each safety relief valve (S/RV) relief mode of operation without physically lifting the disk off the seat at power.

During the refueling outage, a sample population of the S/RVs will be removed for safety-mode setpoint testing in accordance with the GGNS IST program, using ASME Boiler and Pressure Vessel Code, Section XI. Each of these removed S/RVs will be tested in the relief mode to verify that the pneumatic actuator functions correctly, and this test sample will be used to provide assurance that the installed S/RV pneumatic actuators will function properly. After the test sample of S/RVs has been replaced with recertified spares, and S/RV controls have been connected, the upper stem nut that couples the valve stem to each newly-installed S/RV's pneumatic actuator will be moved up the stem to allow an uncoupled actuation of the relief-mode actuator. Control air pressure to each actuator will be reduced from normal system pressure to prevent damaging the pneumatic relief-mode actuator. The actuator will be remotely operated from the control room, as required by current test methods, and visual verification will be performed for proper actuator response and range of motion. After proper actuator operation has been verified, the upper stem nut will be returned to its operating stem location. Verification of proper system logic controls and function for every installed S/RV will continue to be performed, as required by Technical Specifications.

The commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards if the 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.

Entergy Operations has evaluated the no significant hazards considerations in its request for a license amendment. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. is providing the following analysis of the proposed amendment against the three standards in 10 CFR 50.92:

a. No significant increase in the probability or consequences of an accident previously evaluated results from this change.

Each refueling outage, a test sample of the population of S/RVs is removed from the plant to perform testing as required by ASME Boiler and Pressure Vessel Code, Section XI. These S/RVs will be stroked in the relief mode during as-found testing, and are therefore verified to operate properly when each S/RV stem is raised by the relief-mode pneumatic actuator. This proposed surveillance verifies proper S/RV relief-mode operation of all installed S/RVs based upon this test sample. This testing, in conjunction with replacement of each S/RV prior to the end of its expected service life, provides reasonable assurance that the installed S/RVs will perform as well as the test population of S/RVs.

After the S/RVs have been replaced in the plant, and after all controls are reconnected, the relief-mode actuator on each newly-installed S/RV will be uncoupled from the S/RV stem, and stroked. This actuator stroke will verify that no damage has occurred to the relief-mode actuator during S/RV transportation from its storage location to its operating location. The direct coupling of the valve stem to disk provides assurance that proper relief actuation will occur when the actuator is operated. The safety-mode components are completely encased within the valve body and bonnet, which provides a rugged structure to prevent damage to these components. The remaining installed S/RVs will continue to be tested for proper control system function as previously required by Technical Specifications. The direct coupling of the S/RV stem to disk provides assurance that proper relief-mode actuation will occur when the actuator is operated. The safety mode of the GGNS S/RVs is not affected by a malfunction of the relief-mode components.

Blockage of each S/RV discharge line will be prevented by the same Foreign Material Exclusion (FME) controls that exist for other reactor vessel and support systems. These FME controls, combined with the horizontal orientation of the S/RV discharge piping mating surfaces, provide reasonable assurance that discharge line blockage will not occur.

Therefore, no significant increase in the probability or consequences of an accident previously evaluated results from this proposed change.

b. This change would not create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change demonstrates that each S/RV will perform its intended relief-mode function, which is the intent of the present surveillance. The relief mode of S/RV operation is demonstrated to be operable based upon successful performance of a test population, S/RV component service life, and existing Technical Specification surveillances. No new failure mechanisms to the relief- mode of operation are introduced, as the proposed surveillance verifies relief actuator operability. Plant FME controls, combined with the horizontal orientation of the S/RV discharge piping mating flange, provides reasonable assurance that discharge line blockage will not occur. This proposed change does not add any new systems, structures, or components, nor does it introduce new S/RV operating modes.

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

c. This change would not involve a significant reduction in the margin of safety.

This proposed change will verify that the relief mode of all installed S/RVs will operate properly based upon demonstrated relief mode performance of a sample of S/RVs. The failure mode of the S/RV relief function would require a failure of either the pneumatic actuator, lifting linkage, or solenoid block. Each of these items has been verified to have a service life exceeding the replacement cycle of each S/RV. Therefore, proper operation of a sample population of S/RVs provides reasonable assurance that the remaining S/RVs would perform identically, within the original margin of expected S/RV operability. In addition, each S/RVFEs solenoid block and control functions will continue to be tested and cycled each refueling outage. The removal of the valve stroke surveillance for all S/RVs does not increase the possibility of valve malfunction, since valve stroke is verified during the as-found testing of the sample population of S/RVs. This proposed surveillance test reduces the number of S/RV actuations, and therefore, reduces challenges to the system both mechanically and thermally. Also, the proposed alternative method of testing reduces the possibility of a stuck-open S/RV, since this proposed method will not stroke the S/RVs with the reactor pressurized during reactor power operations.

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

Based on the above evaluation, Entergy Operations, Inc. has concluded that operation in accordance with the proposed amendment involves no significant hazards considerations.

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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005–3502

NRC Project Director: William D. Beckner

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

Date of amendment request: May 31, 1996

Description of amendment request: The amendment would provide an alternative method to compensate for inoperable refueling equipment interlocks. The alternative method would be to insert a control rod withdrawal block and verify that all control rods are fully inserted; however, the control rods required to be inserted would not apply to those control rods withdrawn in accordance with LCO 3.10.6, ''Multiple Control Rod Withdrawal -Refueling.'' The amendment would add an additional Required Action for Limiting Condition for Operation (LCO) 3.9.1, ''Refueling Equipment Interlocks,'' of the Technical Specifications (TSs) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The alternative method then could be used to respond to inoperable interlocks instead of only the current method of halting in-vessel fuel movement with equipment associated with the inoperable interlock.

The proposed change does not remove the current Required Action method for LCO 3.9.1 and does not change the surveillance requirements on the refueling equipment. The licensee has also provided changes to the Bases of the TSs for the proposed amendment.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The licensee has proposed the amendment for the TSs for both GGNS and River Bend Station (RBS). References made to the RBS TSs and to RBS in the licensee's analysis of no significant hazards consideration have been removed and replaced by [...]. The licensee's analysis is presented below:

Entergy Operations, Inc. proposes to change the current Grand Gulf Nuclear Station (GGNS) [...] Technical Specifications. The specific proposed change adds additional acceptable Required Actions to the Actions of LCO 3.9.1, ''Refueling Equipment Interlocks,'' [for inoperable interlocks]. The additional Required Actions will add an alternative [method] to [the current method of] suspending fuel movement in the reactor vessel when the refueling interlocks are inoperable. The requested alternative is to insert a control rod withdrawal block immediately and verify all control rods required to be inserted are fully inserted. [The control rods required to be inserted would not apply to control rods withdrawn in accordance with LCO 3.10.6, ''Multiple Control Rod Withdrawal—Refueling.'']

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Entergy Operations, Inc. has evaluated the [criteria for] no significant hazards consideration in its request for this license amendment and determined that no significant hazards consideration results from this change. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. is providing the analysis of the proposed amendment against the three standards in 10 CFR 50.92(c). A description of the no significant hazards consideration determination follows:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The refueling interlocks are explicitly assumed in the GGNS Updated Final Safety Analyses Report (UFSAR) [...] analysis of the control rod removal error or fuel loading error during refueling. This analysis evaluates the probability and consequences of control rod withdrawal during refueling. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion of fuel, provided all control rods are fully inserted during the fuel insertion. The refueling interlocks accomplish this by preventing loading fuel into the core with any control rod withdrawn, or by preventing withdrawal of a rod from the core during fuel loading.

When the refueling interlocks are inoperable the current method of preventing the insertion of fuel when a control rod is withdrawn is to prevent fuel movement. This method is currently required by the Technical Specifications. An alternate method to ensure that fuel is not loaded into a cell with the control rod withdrawn is to prevent control rods from being withdrawn and verify that all control rods required to be inserted are fully inserted. The proposed actions will require that a control rod block be placed in effect thereby ensuring that control rods are not subsequently inappropriately withdrawn. Additionally, following placing the control rod withdrawal block in effect, the proposed actions will require that all required control rods be verified to be fully inserted. This verification is in addition to the requirements to periodically verify control rod position by other Technical Specification requirements. These proposed actions will ensure that control rods are not withdrawn and cannot be inappropriately withdrawn because an electrical or hydraulic block to control rod withdrawal is in place. Like the current requirements the proposed actions will ensure that unacceptable operations are blocked (e.g., loading fuel into a cell with a control rod withdrawn [would be blocked]).

The proposed additional acceptable Required Actions provide the same level of assurance that fuel will not be loaded into a core cell with a control rod withdrawn as the current Required Action or the Technical Specification Surveillance Requirement.

Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The change in the Technical Specification requirements does not involve a change in plant design. The proposed requirements will continue to ensure that fuel is not loaded into the core when a control rod is withdrawn except following the requirements of LCO 3.10.6, ''Multiple Control Rod Removal—Refueling,'' which is unaffected by this change.

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

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

As discussed in the Bases for the affected Technical Specification requirements, inadvertent criticality is prevented during the insertion of fuel provided all control rods are fully inserted during the fuel insertion. The refueling interlocks function to support the refueling procedures by preventing control rod withdrawal during fuel movement and the inadvertent loading of fuel when a control rod is withdrawn.

The proposed change will allow the refueling interlocks to be inoperable and fuel movement to continue only if a control rod withdrawal block is in effect and all required control rods are verified to be fully inserted. These proposed Required Actions provide the same level of protection as the refueling interlocks by preventing a configuration which could lead to an inadvertent criticality event. The refueling procedures will continue to be supported by the proposed required actions because control rods cannot be withdrawn and as a result fuel cannot be inadvertently loaded when a control rod is withdrawn.

Therefore, the proposed changes do not cause 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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005–3502

NRC Project Director: William D. Beckner

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

Date of amendment request: May 31, 1996, as supplemented by letter dated May 2, 1996.

Description of amendment request: The amendment request would revise the current reactor vessel material surveillance program schedule for GGNS. This is the schedule for withdrawing surveillance capsules from the reactor vessel for testing to measure the impact of neutron irradiation of the vessel material and is required by Section III.B.3 of Appendix H, ''Reactor Vessel Material Surveillance Program Requirements,'' of 10 CFR Part 50. The schedule must be approved by the Nuclear Regulatory Commission (NRC) before implementation.

For GGNS, there are three surveillance capsules inside the reactor vessel, each of which contains specimens of the reactor vessel material. The first capsule was removed from the reactor vessel on May 7, 1995, during the 7th refueling outage. Because no useful data is expected from testing the material specimens in the first capsule, the request would allow the first capsule to be placed back into the vessel.

As part of revising the schedule, the licensee is also renumbering the three surveillance capsules so that the capsule removed at the 7th refueling outage becomes the third capsule when it is placed back in the vessel. The proposed change would, however, not extend the time that the next capsule (the renumbered first capsule) would be withdrawn from the GGNS reactor vessel.

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:

Entergy Operations, Inc., proposes to change the withdrawal schedule for the reactor vessel material surveillance capsules [and renumber the capsules]. The revised schedule for withdrawal of the surveillance capsules is withdrawal of the first capsule at 24 Effective Full Power Years. The withdrawal schedule for the second capsule is to be determined at a later date. The third capsule which was withdrawn on May 7, 1995 is to be returned to reactor vessel during the Fall, 1996 outage and retained as a standby. [The current schedule for withdrawal of the three capsules is 8 and 24 Effective Full Power Years for the first two capsules, and the third capsule is a spare with no specific schedule for withdrawal.]

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

In consideration of the October 4, 1995, decision of the Atomic Safety and Licensing Board concerning an amendment request from Perry Nuclear Power Plant, Entergy Operations, Inc. has evaluated the no significant hazards consideration in its request for a change to the withdrawal schedule required by 10 CFR 50, Appendix H, and determined that no significant hazards consideration results from this change. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. is providing the analysis of the proposed amendment against the three standards in 10 CFR 50.92(c):

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The change revises the withdrawal schedule for the reactor vessel material surveillance capsules and returns a withdrawn capsule to the reactor vessel. The capsules [only contain specimens of the reactor vessel material and] are not an initiator of any previously analyzed accident. The withdrawal or return of the surveillance capsule does not effect the probability or consequences of any previously analyzed accident. Extending the time for withdrawal of the first capsule and returning the withdrawn capsule to the vessel do not adversely affect the pressure temperature limit curves for the reactor vessel. Regulatory Guide 1.99 [, ''Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials,''] is currently used to prepare the pressure temperature limit curves and is inherently conservative for boiling water reactors (BWRs)[, as GGNS]. The current pressure temperature limit curves will continue to be adhered to. Additionally, [GGNS] participates in the supplemental test program designed to significantly increase the amount of BWR surveillance data. [This program has supplemental capsules which were installed in the Cooper and Oyster Creek Nuclear Power Plants, which contain the limiting GGNS weld and plate vessel material, and which will be withdrawn in 1996, 2000, and 2002.] This program will be used to complement the GGNS surveillance program such that postponement of the capsule withdrawals will have minimal impact on the understanding of the irradiation effects on the GGNS vessel.

[The licensee stated in its May 2, 1996, letter that testing of the specimens in the removed capsule may not provide useful indicators of the damage to the vessel material because the low neutron fluence on the vessel and the good material chemistry will result in a minimal null-ductility temperature shift. Testing the material specimens will destroy them; however, placing the capsule back in the vessel will allow the specimens to have more irradiation until useful data could be obtained from testing the specimens.]

Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

Returning the withdrawn capsule to the vessel and postponing the withdrawal of the first capsule do not contribute to the possibility of a new or different kind of accident or [plant] malfunction from those previously analyzed [in the Updated Final Safety Analysis Report for GGNS]. Failure of the reactor vessel is not a credible accident since the vessel itself is a highly reliable component. This change does not affect that determination. The potential for reactor vessel cracking will be adequately assessed by the proposed withdrawal schedule.

[The licensee stated in its May 2, 1996, letter that testing of the specimens in the removed capsule may not be useful indicators of the damage to the vessel material because the low neutron fluence on the vessel and good material chemistry will result in a minimal shift.]

In addition, the results from the supplemental test program will provide indication of the condition of the vessel until the data from the first GGNS capsule[, withdrawn and tested,] are available. The proposed change provides the same level of confidence in the integrity of the vessel. The pressure temperature curves are currently controlled by the Technical Specifications and are determined using the conservative methodology in Regulatory Guide 1.99. Therefore, the possibility of failure of the reactor vessel is not increased. The proposed change does not involve a change in the design of the plant. The current pressure temperature limit curves are inherently conservative and will continue to be adhered to.

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

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

The current pressure temperature limit curves [for the reactor vessel] are inherently conservative and provide sufficient margin to ensure the integrity of the reactor vessel. The [proposed] changes do not adversely affect these curves. The supplemental test program will be used to complement the GGNS surveillance program such that postponement of the capsule withdrawal [and testing] will have minimal impact on the understanding of irradiation effects on the GGNS vessel. The capsules removed in 1996 as part of the supplemental program will have a [neutron] fluence higher than the 25% of the design life fluence used in establishing the original GGNS [reactor vessel material surveillance program] schedule; therefore, the use of the supplemental test program results will meet the intent of the original test schedule.

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

Based on the above evaluation, Entergy Operations, Inc. has concluded that operation in accordance with the proposed change involves no significant hazards consideration.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005–3502

NRC Project Director: William D. Beckner

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

Dates of amendment request: March 21, 1996, and May 13, 1996

Description of amendment request: The licensee proposed to change the Turkey Point Units 3 and 4 Technical Specifications (TS) to relocate the requirements of the Radiological Effluent Technical Specifications (RETS) to other documents.

The proposed amendments would relocate the LIMITING CONDITIONS FOR OPERATION (LCO) and SURVEILLANCE REQUIREMENTS associated with the RETS in accordance with GL 89–01, NUREG–1301, and NUREG–1431, Rev. 1. The definition in TS 1.15, ''Members of the Public,'' would be deleted since it is already located in 10 CFR Part 20 and has been inserted into the Offsite Dose Calculation Manual (ODCM). The definitions for the ODCM and Process Control Program (PCP) would be relocated to the Administrative Controls section of the TS. TS 3/4.3.3.5 and the radioactive gaseous effluent portion of TS 3/4.3.3.6 and associated tables, instrumentation operational conditions, remedial actions and surveillance requirements would be controlled through the ODCM or PCP and associated procedures. Technical

Specification Administrative Control sections would contain the programmatic controls for the ODCM and PCP. The remaining portion of TS 3.3.3.6 would retain the operational conditions, remedial actions, and surveillance requirements for the explosive gas monitor instrumentation.

The procedural details of the current TS on radioactive effluents and radiological environmental monitoring would be deleted. Associated operational conditions, remedial actions and surveillance requirements presently in the Technical Specifications would be controlled through the ODCM or PCP.

Administrative changes to the TS were also proposed due to paragraph and section numbering changes and relocations associated with the proposed technical changes.

New sections TS 6.8.4f and 6.8.4g were proposed to provide programmatic controls for the Radiological Effluents Controls Program and the Radiological Environmental Monitoring Program.

TS 6.9.1.3 and TS 6.9.1.4 would be simplified and the reporting details now contained in these specifications would be relocated to the ODCM or PCP with the exception of the requirement to report licensee-initiated changes to the PCP in the Annual Radioactive Effluent Release Report.

New record retention requirements changes for the ODCM and PCP would be added to TS 6.10.3q.

In summary, as provided in the guidance, the current technical content of the specifications which would be transferred to the ODCM or the PCP. New programmatic controls for radioactive effluents and radioactive effluent monitoring would be added to the TS, as well as further clarification to the definitions of the ODCM and PCP. The Technical Specification requirements for Gas Decay Tanks and Explosive Gas Mixture would be relocated to the Plant Systems section of the 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)Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes being proposed are administrative in nature in that they relocate Technical Specification requirements associated with RETS from the Technical Specifications to the ODCM or PCP. These changes are in accordance with the recommendations contained in GL 89–01, NUREG 1301, and NUREG 1431 Rev. 1. The only change being made to existing requirements or commitments are administrative in nature. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Since the accident analyses remain bounding, their probability or consequences are not adversely affected. Therefore, the probability or consequences of an accident previously evaluated are not affected.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes being proposed are administrative in nature in that they relocate Technical Specification requirements associated with RETS from the Technical Specifications to the ODCM or PCP. These changes are in accordance with the recommendations contained in GL 89–01, NUREG 1301, and NUREG 1431, Rev. 1. The only change being made to existing requirements or commitments are administrative in nature. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment used to mitigate the consequences of an accident.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The changes being proposed are administrative in nature in that they relocate Technical Specification requirements associated with RETS from the Technical Specifications to the ODCM or PCP. These changes are in accordance with the recommendations contained in GL 89–01, NUREG 1301, and NUREG 1431, Rev. 1. The only change being made to existing requirements or commitments are administrative in nature. All technical content is preserved. The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments.

Therefore, a significant reduction in a margin of safety would not be involved.

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

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

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: Frederick J. Hebdon

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

Dates of amendment request: May 28, 1996

Description of amendment request: The licensee proposed to change the Turkey Point Units 3 and 4 Technical Specifications (TS) to change the licensed qualifications of the Operations Manager. The proposed change would delete the qualification option that the Operations Manger could have held a Senior Reactor Operator License on a boiling water reactor and replace it with an option that this individual could have completed the Turkey Point Nuclear Plant Senior Management Operation Training Course (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level).

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The change being proposed is administrative in nature, addresses organizational personnel qualification issues, and does not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, or Technical Specifications that preserve safety analysis assumptions.

The individual Florida Power & Light Company (FPL) chooses to fill the position of Operations Manager will have extensive educational and management- level nuclear power experience meeting the criteria of ANSI N18.1–1971. The Operations Supervisor and Nuclear Plant Supervisors maintain SRO licenses on Turkey Point. The current Technical Specifications do not require the Operations Manager to hold an SRO License at Turkey Point. The current Technical Specifications permit the Operations Manager to have held an SRO License on another plant. The proposed change will continue to require that the Operations Manager has completed the Turkey Point Nuclear Plant Senior Management Operations Training Course if the incumbent did not previously hold an SRO license. The Turkey Point Nuclear Plant Senior Management Operations Training Course ensures that the Operations Manager has the training on plant-specific systems and procedures at Turkey Point and a knowledge level equivalent to the license requirements for operations management.

The on-shift Operations' organization is, and will continue to be, supervised and directed by the Operations Supervisor, who is currently required by Technical Specification 6.2.2.h. to hold an SRO License.

Additionally, the proposed changes do not impact or change, in any way, the minimum on-shift manning or qualifications for those individuals responsible for the actual licensed operation of the facility as required by 10 CFR 50.54(l).

Based on the above, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change being proposed is administrative in nature, addresses personnel qualification issues, does not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, or Technical Specifications that preserve safety analysis assumptions.

The proposed changes address organizational and qualifications issues related to the criteria used for assignment of individuals to the Operations organization off-shift management chain of command. Since the proposed change does not impact or change, in any way, the minimum on-shift manning or qualifications for those individuals responsible for the actual licensed operation of the facility, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed change addresses organizational and qualification issues related to the criteria used for assignment of individuals to the Operations organization off-shift management chain of command. The proposed change does not impact or change, in any way, the minimum on-shift manning or qualifications for those individuals responsible for the actual licensed operation of the facility.

FPL's operating organization at Turkey Point Plant is shown on Figure 1–2, Appendix A of the NRC-approved FPL Topical Quality Assurance Report (TQAR). Since changes to the TQAR are governed by 10 CFR § 50.54(a)(3), any changes to the TQAR that reduce commitments previously accepted by the NRC require approval by the NRC prior to implementation.

While the Operations Manager is responsible for the plant's operating organization, his responsibilities also include management of the plant's Health Physics and Chemistry departments. The Operations organization is supervised and directed by the Operations Supervisor, who is required by Technical Specification 6.2.2.h. to hold a Senior Reactor Operator License. The Turkey Point Units 3 and 4 Technical Specifications do not require that the Operations Manager maintain an SRO License (nor even that the incumbent has ever held a Senior Reactor Operator License at Turkey Point). The Turkey Point Technical Specification 6.3.1, FACILITY STAFF QUALIFICATIONS, will ensure that, other than license certification, the individual filling the Operations Manager position has the requisite education, training, and experience for the management position.

As a result, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: Frederick J. Hebdon

GPU Nuclear Corporation and Saxton Nuclear Experimental Corporation, Docket No. 50–146, Saxton Nuclear Experimental Facility (SNEF), Bedford County, Pennsylvania

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

Description of amendment request: The proposed amendment would (1) increase the scope of work permitted within the exclusion area at the SNEF to include action preparatory to major component and facility decommissioning limited to asbestos removal, removal of defunct plant electrical services, and installation of decommissioning support facilities and systems such as heating, ventilation, and air conditioning,

(2) eliminate administrative access controls requiring that the grating covering the auxiliary compartment stairwell and rod room door remain locked except for authorized entry, and (3) revise the facility layout diagram to allow the exclusion area to consist of, at a minimum, the containment vessel, and at a maximum, extend to the SNEF outer security fence, and to include on the diagram the footprint of the proposed decommissioning support facilities.

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

The proposed changes do not involve a significant hazards considerations because the changes would not:

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

The SNEF ended power operation in May 1972, and the reactor core has been removed. In its present condition, the only accidents applicable to the site are fire, flooding, and radiological hazard. The additional activities associated with the expansion of the permissible work scope will not involve a significant increase in the probability or consequences of a fire. There is no effect on the probability or consequences of flooding nor would there be a significant increase in the probability or consequences of an offsite radiological hazard. The relocation of administratively controlled accesses in accordance with the revised wording and the proposed clarification of the facility layout diagram would have no affect on analyzed accidents. Activities associated with the construction of the decommissioning support facilities and the existence of the completed buildings depicted on the revised figure will not involve a significant increase in the probability or consequences of a fire, flood, or radiological hazard. The proposed changes identified by this technical specification change request 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 from any previously analyzed.

For the reasons discussed in 1 above, the possibility of a new or different kind of accident from any accident previously evaluated will not be created by the performance of the activities delineated in the proposed revised technical specifications. There is similarly no possibility of a new or different kind of accident from any accident previously evaluated that would result from relocation of administratively controlled accesses within the containment vessel; from the flexibility to relocate/modify the exclusion area fence or from the identification of the footprint, construction and existence of the completed decommissioning support facilities.

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

For the reasons discussed in 1 above, none of the proposed changes involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis of the licensees 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: Saxton Community Library, 911 Church Street, Saxton, Pennsylvania 16678 Attorney for the Licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW, Washington, DC 20037

NRC Project Director: Seymour H. Weiss

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

Date of amendment request: May 20, 1996

Description of amendment request: The proposed amendment would revise the *Facility Operating License No.* NPF–47 and Appendix C to the license to reflect the name change from Gulf States Utilities Company to Entergy Gulf States, Inc.

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. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change documents changing the legal name of the company. The proposed change will not affect any other obligations. The company will still own all of the same assets, serve the same customers, and all existing obligations and commitments will continue to be honored.

Therefore, the proposed change does no significantly increase the probability or consequences of an accident previously evaluated.

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

The administrative changes in the Operating License requirements do not involve any change in the design of the plant.

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

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

The proposed change is administrative in nature, as described above, therefore, this change does not reduce the level of safety imposed by any current requirements.

Therefore, the proposed changes do not cause 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

Northeast Nuclear Energy Company (NNECO), Docket No. 50–245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of amendment request: April 25, 1996

Description of amendment request: The change modifies the calibration requirement for the source range monitors and intermediate range monitors by noting that the sensors are excluded.

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:

Pursuant to 10 CFR 50.92, NNECO has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration (SHC) since the proposed change satisfies the criteria in 10 CFR 50.92(c). That is, the proposed change does not:

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

By removing the requirement for sensor calibration the function and safety performance of these systems will not be affected. Existing surveillances, operator verification of overlap and system interlocks ensure correct system performance without sensor calibration.

Therefore, based on the above, the proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of any previously analyzed accident.

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

This change does not cause the source range monitors (SRM) or the intermediate range monitors (IRM) to function any differently than intended by design and, therefore, does not create the possibility of a new or different kind of accident. The Technical Specification change deletes a Technical Specification requirement which could not literally be complied with for one component and that has no effect on the functional performance of the SRMs or IRMs.

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

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

This change corrects a Technical Specification requirement which could not literally be complied with for one component and that has no effect on the functional performance of the SRMs or IRMs. Instrument calibrations and functional checks are still performed during each refueling outage to assure adequate system performance.

Therefore, this change has no impact on the margin to 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, and Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment requests: February 14, 1996

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2, to revise 30 TS and add two new TS surveillance requirements to support implementation of extended fuel cycles at DCPP, Unit Nos. 1 and 2. The specific TS changes proposed include those for 9 trip actuating device tests, 12 fluid system actuation tests, and 11 miscellaneous tests. Two of the fluid system actuation tests are proposed new TS surveillance requirements. The TS changes also include the addition of a new frequency notation, ''R24, REFUELING INTERVAL,'' to Table 1.1 of the TS. Also, a revision that applies to all subsequent TS changes involves revising the Bases section of TS 4.0.2 to change the surveillance frequency from an 18-month surveillance interval to at least once each refueling interval.

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 surveillance interval notation addition in TS Table 1.1 and the updated TS 4.0.2 Bases section are administrative changes that do not affect the probability or consequences of accidents.

The 30 proposed TS surveillance interval increases from 18 to 24 months do not alter the intent or method by which the inspections, tests, or verifications are conducted, do not alter the way any structure, system, or component functions, and do not change the manner in which the plant is operated. The surveillance, maintenance, and operating histories indicate that the equipment will continue to perform satisfactorily with longer surveillance intervals. Few surveillance and maintenance problems were identified. No problems recurred, with the exception of those associated with the pressurizer heater emergency breakers, which will continue to be surveilled on a quarterly frequency until they are replaced.

There are no known mechanisms that would significantly degrade the performance of the evaluated equipment during normal plant operation. All potential time-related degradation mechanisms have insignificant effects in the timeframe of interest (24 months +25 percent, or 30 months). Based on the past performance of the equipment, the probability or consequences of accidents would not be significantly affected by the proposed surveillance interval increases.

The 24-month surveillance intervals for the two new TS proposed to verify that the CCW [component cooling water] and ASW [auxiliary saltwater] pumps will start automatically are based on an evaluation of historical operation, maintenance, and surveillance data for the pumps. These historical data are available because the pumps have been operated, maintained, and tested on 18- month intervals in accordance with procedures since initial plant startup. These new surveillances represent additional TS requirements to ensure the CCW and ASW pumps start when required. No known degradation mechanisms would significantly affect the ability of the pumps to start over the timeframe of interest (30 months maximum). Based on the past performance of the equipment, these proposed new TS would not affect the probability or consequences of accidents.

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

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

The surveillance interval notation addition in TS Table 1.1 and the updated TS 4.0.2 Bases section are administrative changes that do not affect the type of accidents possible.

For the 30 proposed TS changes involving surveillance interval increases from 18 to 24 months, the surveillance and maintenance histories indicate that the equipment will continue to effectively perform its design function over the longer operating cycles. Additionally, the increased surveillance intervals do not result in any physical modifications, affect safety function performance or the manner in which the plant is operated, or alter the intent or method by which surveillance tests are performed. Only a few problems have been identified and generally have not recurred. All potential time-related degradations have insignificant effects in the timeframe of interest. The proposed surveillance interval increases would not affect the type of accidents possible.

The 24-month surveillance intervals for the two new TS proposed to verify starting of the CCW and ASW pumps are based on an evaluation of historical operation, maintenance, and surveillance data. These new TS represent additional requirements to ensure the CCW and ASW pumps start when required. No known degradation mechanisms would significantly affect the ability of the pumps to start over the timeframe of interest. These proposed new TS would not affect the type of accidents possible.

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

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

The surveillance interval notation addition in TS Table 1.1 and the updated TS 4.0.2 Bases section are administrative changes that do not affect the margin of safety.

For the 30 proposed TS changes involving 18- to 24-month surveillance interval increases, evaluation of historical surveillance and maintenance data indicates there have been only a few problems experienced with the evaluated equipment.

There are no indications that potential problems would be cycle-length dependent or that potential degradation would be significant for the timeframe of interest and, therefore, increasing the surveillance interval will have little, if any, impact on safety. There is no safety analysis impact since these changes will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria. Safety margins would not be significantly affected by the proposed surveillance interval increases.

As previously noted, the 24-month surveillance intervals for the two new TS are based on an evaluation of historical data, represent additional requirements, and are not believed to be significantly affected by potential time-dependent degradation. As such, these proposed new TS would not affect any margin of safety.

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

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment requests: May 9, 1996

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

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

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

Revising the description of the containment spray (CS) initiating signal clarifies the design of the plant and provides uniformity across the Technical Specifications (TS) associated with the CS initiation function. The enhanced description does not affect system operation or performance, nor the probability of any event initiators. The changes do not affect any engineered safety feature actuation setpoints or accident mitigation capabilities.

The administrative changes to TS 3/4.3.2, Table 4.3–2, correct the column headings and restore test frequency notation. The changes only revise the TS to correspond with previously issued license amendments (LAs).

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

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

The administrative changes in the description of the CS initiating signal provide uniformity across the TS associated with the spray system. There are no design, operation, maintenance, or testing changes associated with the administrative changes.

The administrative changes to TS 3/4.3.2, Table 4.3–2, correct the column headings and restore test frequency notation. The changes only revise the TS to correspond with previously issued LAs.

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

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

The administrative changes in CS signal description are not associated with any design, operation, maintenance, or testing revisions.

The administrative changes to TS 3/4.3.2, Table 4.3–2, correct the column headings and restore test frequency notation. The changes only revise the TS to correspond with previously issued LAs.

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

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

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

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

NRC Project Director: William H. Bateman

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 20, 1996 (TS 373)

Description of amendment request: The proposed amendment revises the technical specifications to incorporate a 24-hour delay in implementing the action requirements due to a missed surveillance requirement when the action requirements provide a restoration time that is less than 24 hours. This change also clarifies that the time limit of the action requirements applies from the point in time it is identified a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. The licensee claims this amendment is consistent with generic guidance.

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

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

The proposed amendment to TS definition 1.0.LL is in accordance with the guidance of GL 87–09 and NUREG 1433, Revision 1. The proposed change will allow BFN to continue operation for an additional 24 hours after discovery of a missed surveillance. The change being proposed does not affect the precursor for any accident or transient analyzed in Chapter 14 of the BFN Updated Final Safety Analysis Report. The proposed change does not reflect a revision to the physical design and/or operation of the plant. Therefore, operation of the facility in accordance with the proposed change does not affect the probability or consequences of an accident previously evaluated.

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

The proposed amendment to TS definition 1.0.LL is in accordance with the guidance of GL 87–09 and NUREG 1433, Revision 1. The proposed change will allow the plant to continue operation for an additional 24 hours after discovery of a missed surveillance. The change being proposed will not change the physical plant or the modes of operation defined in the facility license. The change does not involve the addition or modification of equipment, nor do they alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment to TS definition 1.0.LL is in accordance with the guidance of GL 87–09 and NUREG 1433, Revision 1. The proposed change does not affect plant safety analysis or change the physical design or operation of the plant. The proposed change will allow the plant up to 24 hours to perform a missed surveillance. The overall effect is a net gain in plant safety by avoiding unnecessary shutdowns and the associate system transients due to missed surveillance. Therefore, operation of the facility in accordance with the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

Date of amendment request: May 8, 1996

Description of amendment request: The proposed amendment would revise Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 5.3, ''Reactor,'' and TS 5.4, ''Fuel Storage,'' by removing the enrichment limit for reload fuel and imposing fuel storage restrictions on the spent fuel storage racks and the new fuel storage racks. The revised TS are structured consistent with the Westinghouse Standard Technical Specifications and the fuel storage restrictions are based on the criticality analyses used to support TS Amendment 92 dated March 7, 1991.

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

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

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

The criticality analysis which was performed in support of Technical Specification Amendment 92, dated March 7, 1991, demonstrated that adequate margins to criticality can be maintained with fuel enrichments up to 49.2 grams of U^{235} per axial centimeter stored in the New Fuel Storage Racks and enrichments up to 52.3 grams of U^{235} per axial centimeter stored in the Spent Fuel Storage Racks.

The bounding cases of the analysis demonstrated that k_{eff} remains less than 0.95 in the Spent Fuel Storage Racks and the New Fuel Storage Racks if flooded with unborated water. The bounding cases of the analysis also demonstrated that k_{eff} remains less than 0.98 in the New Fuel Storage Racks if moderated by optimally misted moderator. Therefore, the 49.2 grams of U^{235} per axial centimeter enrichment is acceptable for storage in the New Fuel Storage Racks and 52.3 grams of U^{235} per axial centimeter for storage in the Spent Fuel Storage Racks.

The only other accident that needs to be considered is a fuel handling accident. Since the mass of the fuel assembly would not be appreciably altered by the increased fuel enrichment, the probability of this accident occurring is not changed. The consequences of a fuel handling accident also would not be affected by the use of higher fuel enrichment since the fission product inventories in a fuel assembly are not a significant function of initial fuel enrichment. This accident was analyzed in the criticality analysis which was performed in support of Technical Specification Amendment 92, dated March 7, 1991.

It should be noted that any changes in the nuclear properties of the reactor core that may result from higher fuel enrichments would be analyzed in the appropriate reload analysis.

The administrative relocation of information to licensee controlled documents (i.e., USAR) conforms to NRC policy for the content of technical specifications and does not increase the probability or consequences of an accident.

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

As discussed above, the only safety issue significantly affected by the proposed change is the criticality analysis of the Spent Fuel Storage Racks and the New Fuel Storage Racks. Since it has been demonstrated that $k^{G\cong 2eff}$ remains below 0.95 and 0.98, respectively, in those areas, no new or different accident would be created through the use of fuel enrichments up to 52.3 grams of U^{235} per axial centimeter at the Kewaunee Nuclear Power Plant. Administrative controls will ensure that only fuel enriched to 49.2 grams of U^{235} per axial centimeter or less will be placed into the New Fuel Storage Racks.

The relocation of information to licensee controlled documents does not create the possibility of a new or different kind of accident.

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

Since the criticality analyses have shown that increasing the allowable weight percent enrichment to 52.3 grams of U^{235} per axial centimeter would not increase k_{eff} above 0.95 in the Spent Fuel Storage Racks and increasing the allowable weight percent enrichment to 49.2 grams of U^{235} per axial centimeter would not increase k_{eff} above 0.98 in the New Fuel Storage Racks, it is concluded that this proposed change would not reduce the margin of safety. Any changes in the nuclear properties of the reactor core that may result from higher fuel enrichments would be analyzed in the appropriate reload analysis to ensure compliance with applicable reload considerations and requirements.

Relocation of information to licensee controlled documents is an administrative action and therefore does not reduce the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus

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

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

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

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 17, 1996

Description of amendment request: The proposed amendments would modify Technical Specification Section 3/4.4.5, Steam Generators, 3/4.4.6, Reactor Coolant System Leakage, and associate Bases to allow the installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes.

Date of individual notice in the Federal Register: May 29, 1996 (61 FR 26936)

Expiration date of individual notice: June 28, 1996

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

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

Date of application for amendment: April 24, 1996

Brief description of amendment request: The proposed amendment would modify Technical Specifications (TSs) 5.3.1 and 6.9.3.2 to reflect use of new fuel obtained from ABB/Combustion Engineering, and to incorporate staff-approved core reload analysis computer programs (codes). Date of individual notice in Federal Register: May 1, 1996 (61 FR 19326)

Expiration date of individual notice: May 31, 1996

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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: January 5, 1996, as supplemented by letters dated April 19, May 1, and May 10, 1996.

Brief description of amendments: The amendments revise the operating licenses and Technical Specification (TS) Section 1.26 to increase the authorized rated thermal power. The amendments also revise TS 4.1.1.4, 3.1.3.4, and 3.2.6 (Figure 3.2–1) to lower the allowable reactor coolant system cold leg temperature limits for each of the three Palo Verde Nuclear Generating Station units, and TS 3.4.2.1 and 3.4.2.2 to lower the pressurizer safety valve setpoints for Units 1 and 3 to support the increased power operation. The Unit 2 pressurizer safety valve setpoints in TS 3.4.2.1 and 3.4.2.2 were revised in Amendment 78, approved March 28, 1995, to the same values being requested for Units 1 and 3 in this submittal.

Date of issuance: May 23, 1996

Effective date: May 23, 1996, to be implemented for Unit 1 within 30 days of issuance; to be implemented for Unit 2 within 30 days of issuance; to be implemented for Unit 3 within 45 days as of the date of issuance, except for the pressurizer safety valve setpoints change which are effective prior to startup from Unit 3's sixth refueling outage.

Amendment Nos.: Unit 1 - 108; Unit 2 - 100; Unit 3 - 80

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7544) The April 19, May 1, and May 10, 1996, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 23, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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

Date of application for amendment: January 31, 1996.

Brief description of amendment: This amendment revises the Technical Specifications Section 4.4 to allow the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing.

Date of issuance: May 28, 1996

Effective date: May 28, 1996

Amendment No. 169

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7545) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 28, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: November 15, 1995, as supplemented by letters dated March 15, and April 10, 1996

Brief description of amendments: The amendments revise the Technical Specifications and the associated Bases to increase the setpoint tolerance of the main steam safety valves (MSSVs) from plus or minus 1% to plus or minus 3%, to incorporate a requirement to reset the as-left MSSV lift settings to within plus or minus 1% following surveillance testing, and to delete two obsolete footnotes.

Date of issuance: May 31, 1996

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

Amendment Nos.: 146 and 140

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65676). The March 15 and April 10, 1996 letters provided clarifying information that did not change the scope of the November 15, 1995 application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 31, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: January 12, 1995, as supplemented by letter dated June 29, 1995

Brief description of amendments: The amendments revise and clarify portions of Technical Specification Section 6.0, ''Administrative Controls.''

Date of issuance: May 30, 1996

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

Amendment Nos.: 145 and 139

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

Date of initial notice in Federal Register: November 24, 1995 (60 FR 58109) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 30, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: April 3, 1996

Brief description of amendments: The amendments revise the Technical Specifications and the associated Bases to provide that if neither Train A or Train B of the hydrogen igniter is operable in any one containment region, there is an allowance of 7 days to restore one hydrogen igniter to operable status, or be in hot shutdown within the next 6 hours.

Date of issuance: June 3, 1996

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

Amendment Nos.: 147 and 141

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

Date of initial notice in Federal Register: April 16, 1996 (61 FR 16649) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 3, 1996 No significant hazards consideration comments received: No

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

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

Date of amendment request: May 19, 1995, as supplemented by letter dated December 7, 1995

Brief description of amendment: The amendment revised the recombiner surveillance requirements to conform with the staff guidance provided in NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants.''

Date of issuance: June 5, 1996

Effective date: June 5, 1996

Amendment No.: 119

Facility Operating License No. NPF–38. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 180) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 5, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: January 4, 1996

Brief description of amendments: These amendments rectify a discrepancy in Technical Specification 3.5.3, and provide assurance that administrative controls for High Pressure Safety Injection pumps remain effective in the lower operational modes.

Date of Issuance: May 30, 1996

Effective Date: May 30, 1996

Amendment Nos.: 143 and 183

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

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5813) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 30, 1996. No significant hazards consideration comments received: No.

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

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: November 22, 1995

Brief description of amendments: These amendments upgrade existing TS 3/4.4.6.1 for the Reactor Coolant System Leakage Detection Systems by adopting the Standard Technical Specifications for Combustion Engineering Plants to both St. Lucie Units.

Date of Issuance: May 30, 1996

Effective Date: May 30, 1996

Amendment Nos.: 144 and 84

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1629) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 30, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 28, 1996 (TSCR 234)

Brief description of amendment: The amendment modifies Technical Specification pages 3.1–5 and 3.1–16 to indicate 40 percent of the rated reactor thermal power as the anticipatory reactor scram bypass setpoint on turbine trip or generator load rejection.

Date of Issuance: June 4, 1996

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

Amendment No.: 184

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18167) The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated June 4, 1996 No significant hazards consideration comments received: No.

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

Houston Lighting & Power Company, City Public Service Board of San Antonio Central Power and Light Company, City of Austin, Texas, Docket No. 50–498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: January 22, 1996, as supplemented April 4 and May 2, 1996

Brief description of amendment: The amendment modified the steam generator tube plugging criteria in TS 3/4.4.5, Steam Generators, the allowable primary-to-secondary leakage in TS 3/4.4.6.2, Operational Leakage, and the associated Bases. These changes allowed the implementation of alternate steam generator tube plugging criteria for the tube support plate/tube intersections for Unit 1.

Date of issuance: May 22, 1996

Effective date: May 22, 1996

Amendment No.: 83

Facility Operating License No. NPF–76. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: April 16, 1996 (61 FR 16651) as corrected April 22, 1996 (61 FR 17735). The additional information contained in the supplemental letter dated May 2, 1996, 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, 1996. 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, TX 77488

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

Date of application for amendment: July 21, 1995, as supplemented August 8, 1995 and December 15, 1995

Brief description of amendment: The amendment made administrative changes to various sections of the DAEC Technical Specifications (TS). The amendment replaced the surveillance condition when an Emergency Service Water pump or loop is inoperable with an OPERABILITY verification of the opposite train's Emergency Diesel Generator (EDG). The amendment modified the TS to allow credit for demonstration of EDG OPERABILITY that occurred within the previous 24 hours. The amendment revised the format and language of TS Section 5.5

to clarify the requirements and state the capacity of the spent fuel pool and vault storage in order to remove ambiguities in the wording and to be more consistent with the Improved Standard TS guidance. The amendment revised the list of Operations Committee responsibilities (Section 6.5.1.6) to eliminate Committee review of procedures implementing Security and Emergency Plans.

Date of issuance: June 5, 1996

Effective date: June 5, 1996

Amendment No.: 214

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

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49938) and February 2, 1996 (61 FR 3953) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 5, 1996. No significant hazards consideration comments received: No.

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

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: May 4, 1995, as supplemented November 27, 1995, and March 1, 1996

Brief description of amendments: The amendments revise the pressurizer and main steam safety valve lift setting tolerance from plus or minus 1 percent to plus or minus 3 percent (as-found setpoint only), revise the safety limit curves, reformat Section 2, and correct typographical errors.

Date of issuance: May 21, 1996

Effective date: May 21, 1996, with full implementation within 30 days

Amendment Nos.: Unit 1 - 123, Unit 2 - 116

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

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47621) The November 27, 1995, and March 1, 1996, letters provided clarifying information in response to NRC staff questions. 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 21, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 13, 1996

Brief description of amendments: These amendments delete the requirement in Technical Specifications (TS) 4.0.5a for NRC written approval prior to implementation of relief from ASME Code requirements by deleting ''...(g),.except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).'' Also, the amendments add the ASME Section XI definition of ''Biennially or every 2 years - At least once per 731 days,'' in TS 4.0.5b.

Date of issuance: May 28, 1996

Effective date: May 28, 1996

Amendment Nos.: Unit 1 - 112; Unit 2 - 110

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18173) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 28, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 3, 1996

Brief description of amendments: These amendments revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 to revise Technical Specifications 3/4.7.5, ''Control Room Ventilation System;'' 3/4.7.6, ''Auxiliary Building Safeguards Air Filtration System;'' and 3/4.9.12, ''Fuel Handling Building Ventilation System'' to clarify the testing methodology utilized by PG&E to determine the operability of the charcoal and high efficiency particulate air (HEPA) filters in the engineering safeguards features (ESF) air handling units at the Diablo Canyon Power Plant (DCPP).

Date of issuance: May 28, 1996

Effective date: May 28, 1996

Amendment Nos.: Unit 1 - 113; Unit 2 - 111

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18173) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 28, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 8, 1996, as supplemented May 10, 1996, and May 29, 1996, and June 3, 1996.

Brief description of amendment: This amendment modifies the Technical Specifications to correct several typographical errors that were implemented in the Improved Technical Specifications at Ginna Station per Amendment No. 61.

Date of issuance: June 3, 1996

Effective date: As of date of issuance.

Amendment No.: 65

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 24965, dated May 17, 1996). 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 published May 17, 1996, also provided for a hearing by June 17, 1996, but indicated that if a 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 is contained in a Safety Evaluation dated June 3, 1996.

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

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

Date of application for amendment: February 9, 1996 as superseded by letter dated March 22, 1996.

Brief description of amendment: The amendment revises Technical Specification (TS) 1.7, 4.6.1.1, 3.6.1.3, 4.6.1.3, 6.8.4 and the associated Bases section to directly reference Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' as required by 10 CFR 50, Appendix J, Option B for the Type A containment integrated leak rate tests and the Type B and C local leak tests.

Date of issuance: May 28, 1996

Effective date: May 28, 1996, to be implemented within 30 days from the date of issuance.

Amendment No.: 111

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18174) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 28, 1996. No significant hazards consideration comments received: No.

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

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: January 30, 1996

Brief description of amendments: The amendments modify the Technical Specifications to increase the minimal allowable reactor coolant system total flow rate.

Date of issuance: June 5, 1996

Effective date: June 5, 1996

Amendment Nos.: 201 and 182

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7559) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 5, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 24, as supplemented by letter dated May 29, 1996.

Brief description of amendment: The amendment would modify the WNP–2 technical specifications to support Cycle 12 operation, reflect use of new fuel obtained from ABB/Combustion Engineering, and incorporate staff-approved core reload analysis computer programs (codes). Date of issuance: June 4, 1996 Effective date: June 4, 1996, to be implemented within 30 days of issuance.

Amendment No.: 146

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

Date of initial notice in Federal Register: May 1, 1996 (61 FR 19326). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 4, 1996. No significant hazards consideration comments received: No.

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

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a Federal Register notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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 room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By July 19, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 20555–001, 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 20555–001, 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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

Commonwealth Edison Company, Docket No. 50–249, Dresden Nuclear Power Station, Unit No. 3

Date of application for amendment: May 22, 1996

Brief description of amendment: The amendment authorizes, on a one- time temporary basis, operation of Dresden, Unit 3, with the structural steel members in the Low Pressure Coolant Injection (LPCI) corner rooms outside the Updated Final Safety Analysis Report (UFSAR) design parameters, but capable of performing their intended safety function. Following a reactor scram on May 15, 1996, Commonwealth Edison Company (ComEd) performed a Safety Evaluation (SE) in accordance with the requirements of 10 CFR 50.59 to determine if the current configuration of the corner room structural steel members had reduced the margin of safety as described in the UFSAR. The SE determined that the configuration does not reduce the margin of safety with respect to the stress allowables for the structural steel if subjected to a Safe Shutdown Earthquake (SSE). An unreviewed safety question was determined to exist because stress allowables for the structural steel subjected to an Operating Basis Earthquake (OBE) were found outside the UFSAR requirements; however, the current configuration of the corner room structural steel members has not significantly reduced the margin of safety as described in the UFSAR.

Date of Issuance: May 31, 1996

Effective date: May 31, 1996

Amendment No.: 144

Facility Operating License No. DPR–25. The amendment revised the license.

Press release issued requesting comments as to proposed no significant hazards consideration: Yes. Joliet Herald News on May 25, 1996, and the Morris Daily Herald on May 29, 1996. Comments received: No comments were received on the proposed no significant hazards consideration determination; however, comments were received concerning the licensee's timeliness and decision-making in restoring the UFSAR design margin to the structural steel members installed the LPCI corner rooms at Dresden Unit 3.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Illinois and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 31, 1996.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690

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

NRC Project Director: Robert A. Capra

Dated at Rockville, Maryland, this 12th day of June 1996.

For the Nuclear Regulatory Commission

John A. Zwolinski,

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

[Doc. 96–15398 Filed 6–18–96; 8:45 am]

BILLING CODE 7590–01–F

OCCUPATIONAL SAFETY AND HEALTH REVIEW COMMISSION

Sunshine Act Meeting

TIME AND DATE: 10:00 a.m., on June 25, 1996.

PLACE: The Commission's National Office at One Lafayette Centre, 1120 20th St., N.W., 9th Floor, Washington, DC 20036–3419.

STATUS: Under 29 C.F.R. § 2203.4(d) this meeting is subject to being closed by a vote of the Commissioners taken at the beginning of the meeting. Since the only matters to be discussed at this meeting will be specific cases in the Commission's adjudicative process, it is likely that, pursuant to 29 C.F.R. § 2203.3(b)(10), the meeting will be closed upon a proper vote taken.

MATTERS TO BE CONSIDERED: Cases in the Commission's adjudicative process.

CONTACT PERSON FOR MORE INFORMATION: Earl R. Ohman, Jr., General Counsel, (202) 606–5410.

Earl R. Ohman, Jr.,

General Counsel.

[FR Doc. 96–15749 Filed 6–17–96; 8:45 am]

BILLING CODE 7600–01–M

RAILROAD RETIREMENT BOARD

Sunshine Act Meeting

Notice is hereby given that the Railroad Retirement Board will hold a meeting on June 26, 1996, 9:00 a.m., at the Board's meeting room on the 8th floor of its headquarters building, 844 North Rush Street, Chicago, Illinois, 60611.

The agenda for this meeting follows:

Portion Open to the Public

(1) Annual Actuarial Report (Sec. 22 of the Railroad Retirement Act of 1974 and Sec. 502 of the Railroad Retirement Solvency Act of 1983)
(2) Fiscal Year 1996 Budget Allocations
(3) Proposed Reorganization—Bureau of Information Systems
(4) Letters to Congress on H.R. 2942 and S. 1552
(5) Draft Legislation Proposed on April 4, 1996—Draft Legislation to Enhance Debt Collection Efforts
(6) Medicare Part B Services (Contract No. 92RRB006)
(7) Regulations, Claims Manuals, Rulings, and Procedures
(8) Status of Intermodal Services Under the Railroad Retirement and Railroad Unemployment Insurance Acts
(9) Regulations—Part 230 (Reduction and Non-Payment of Annuities by Reason of Work)
(10) Employee Service Determinations:
  A. Maryland Midland Railway, Inc.—James W. Schaeffer, Jr.
  B. Joyce Goss
(11) Labor Member Truth in Budgeting Status Report

Portion Closed to the Public

(A) Request for Change in Position Index (Bureau of Hearings and Appeals)
(B) *Pending Board Appeals*:
  (1) Anderson, Raymond
  (2) Garcia, Fedelina
  (3) Herbert, Harold
  (4) Howard, Alvira M.
  (5) McLeod, Jasper N.
  (6) Trybala, Therese A.

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

Dated: June 14, 1996.

Beatrice Ezerski,

Secretary to the Board.

[FR Doc. 96–15720 Filed 6–17–96; 11:09 am]

BILLING CODE 7905–01–M

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549

Extension:
  Rule 15c2–5
  SEC File No. 270–195
  OMB Control No. 3235–0198

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.), the Securities and Exchange Commission (''Commission'') is publishing the following summary of collection for public comment.

Rule 15c2–5 prohibits a broker-dealer from arranging a loan for a customer to whom a security is sold unless, before the transaction is entered into, the broker-dealer first: (1) delivers to the customer a written statement setting forth certain information about the specific arrangement being offered to him; (2) obtains from the customer sufficient information concerning his or her financial situation and needs so as to determine that the entire transaction is suitable for the customer; and (3) retains in his or her files a written statement setting forth the basis upon which the broker-dealer made such determination. The information required by the rule is necessary for the execution of the Commission's mandate under the Securities Exchange Act of 1934 (''Exchange Act'') to prevent fraudulent, manipulative, and deceptive acts and practices by broker-dealers. There are approximately 50 respondents that require an aggregate total of 600 hours to comply with the rule. Each of these approximately 50 registered broker-dealers makes an estimated 6 annual responses, for an aggregate total of 300 responses per year. Each response takes approximately 2 hours to complete. Thus, the total compliance burden per year is 600 burden hours. The approximate cost per hour is $20, resulting in a total cost of compliance for the respondents of $12,000 (600 hours @ $20).

Written comments are invited on: (a) whether the proposed collection of