23. SYSTEM NAME:

Current Research Information System (CRIS), USDA/CSRS (part of National Archives Record Group 540, Records of the Cooperative State Research, Education, and Extension Service).

SYSTEM LOCATION:

National Archives at College Park, 8601 Adelphi Road, College Park, MD 20740–6001.

CATEGORIES OF INDIVIDUALS COVERED BY THE SYSTEM:

Records in the National Archives cover scientists listed on research projects entered into the CRIS.

CATEGORIES OF RECORDS IN THE SYSTEM:

Records in the National Archives covered by this notice include the Current Research Information System (CRIS) File, 1998 (NARA Accession NN3–540–00–001).

ROUTINE USES OF RECORDS MAINTAINED IN THE SYSTEM, INCLUDING CATEGORIES OF USERS AND THE PURPOSE OF SUCH USES:

Reference by Government officials, scholars, students, and members of the general public. The records in the National Archives of the United States are exempt from the Privacy Act of 1974 except for the public notice required by 5 U.S.C. 552a(l)(1)(3). Further information about uses and restrictions may be found in 36 CFR part 1256 and in the Appendix following this notice.

POLICIES AND PRACTICES FOR STORING, RETRIEVING, ACCESSING, RETAINING, AND DISPOSING OF RECORDS IN THE SYSTEM:

a. Storage: Electronic database stored on magnetic tape.

b. Retrievability: Retrieved by name of project leader or co-investigator.

c. Safeguards: Records are kept in locked stack areas accessible only to authorized NARA personnel.

d. Retention and disposal: Records are retained permanently.

SYSTEM MANAGER AND ADDRESS:

The system manager is the Assistant Archivist for Records Services, Washington, DC (NW), 8601 Adelphi Road, College Park, MD 20740–6001.

NOTIFICATION PROCEDURES:

Individuals desiring information from or about these records should direct inquiries to the system manager.

RECORDS ACCESS PROCEDURES:

Upon request, NARA will attempt to locate specific records about individuals and will make the records available subject to the restrictions set forth in 36 CFR part 1256. Enough information must be provided to permit NARA to locate the records in a reasonable amount of time. Records in the National Archives may not be amended and requests for amendment will not be considered. More information regarding access procedures is available in the *Guide to the National Archives of the United States,* which is sold by the Superintendent of Public Documents, Government Printing Office, Washington, DC 20402, and may be consulted at the NARA research facilities listed in 36 CFR part 1253.

Dated: January 31, 2001.

Michael J. Kurtz,

Assistant Archivist for Records Services, Washington, DC.

Appendix

General Statement About Uses and Restrictions

A record from an accessioned system of records may be made available to any person who has applied for and received a researcher identification card. No special qualifications are required in order to use the records of the National Archives. Rule governing the use of records and procedures for applying for research cards are found in 36 CFR part 1254. However, the use of some of the records is subject to restrictions imposed by statute or Executive order, or by the restrictions specified in writing in accordance with 44 U.S.C. 2108 by the transferring agency. Restrictions currently in effect on access to particular records that have been specified by the transferring agency are known as ''specific restrictions.'' Restrictions on access that may apply to more than one record group are termed ''general restrictions.'' They are applicable to the kinds of information or classes of accessioned records designated regardless of the record group to which they have been allocated or the specific system of records in which they are contained. The restrictions are published in the ''Guide to the National Archives of the United States'' and supplemented by restriction statements approved by the Archivist of the United States and set forth in 36 CFR part 1256.

[FR Doc. 01–3102 Filed 2–6–01; 8:45 am]

BILLING CODE 7515–01–P

NUCLEAR REGULATORY COMMISSION

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 January 29, 2001, through February 9, 2001. The last biweekly notice was published on January 24, 2001 (66 FR 7667).

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 an 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 and Directives Branch, Division of Administrative 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's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By March 9, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first Floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: December 28, 2000.

Description of amendment request: The proposed amendment would decrease the allowed outage time for an inoperable channel of the anticipated transient without scram recirculation pump trip instrumentation.

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

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

The proposed changes to the Technical Specifications are to the allowed outage time(s) specified for instrumentation associated with the Anticipated Transient Without Scram (ATWS) Reactor Recirculation Pump Trip (RPT) system.

The proposed changes do not involve a change to the plant design or operating modes. The changes apply to the ATWS–RPT system, but they have no impact on the failure modes or initiators that potentially cause an ATWS, and thus have no impact on the frequency of occurrence of an ATWS event.

The proposed changes do not involve a change to the design of the ATWS–RPT system, as the proposed changes primarily only affect the allowed outage time of the system and do not otherwise affect the manner in which the system is tested or operated. Thus, the manner in which the ATWS–RPT system is designed to respond to an ATWS event is not affected, so its mitigation design function is not impacted. Although, by design, on-lime testing of the ATWS–RPT requires the system to be rendered unavailable for short periods of time, system unavailability is not significantly impacted by the proposed changes. The proposed changes involve the establishment of a reasonable allowed outage time to support online testing needed to periodically confirm system operability, but which minimizes the overall system average unavailability. All of the proposed allowed outage times are based on the Standard Technical Specifications and as such have been determined to be acceptable for maintaining adequate ATWS–RPT availability and for minimizing plant risk. They thus provide reasonable assurance that the ATWS–RPT system will be available on demand to perform its mitigating function in the event of an accident or transient involving a failure of the primary scram function (i.e., the reactor protection system).

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

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

The proposed changes only affect the outage time allowed for the ATWS–RPT instrumentation. They do not involve any changes to the plant design or operation, and thus do not introduce a new failure mode. Therefore, the proposed changes to the TS do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a change to the plant design or operation including the ATWS–RPT system itself. No change to the setpoints of the ATWS–RPT instrumentation is involved. Since ATWS–RPT availability will be maintained to a sufficiently high degree, and since the ATWS–RPT design (including its associated instrument setpoints) is unaffected, the TS will continue to provide adequate assurance that the ATWS–RPT is capable of performing its intended function.

Based on the above, the proposed changes to the TS do not involve a 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.

Attorney for licensee: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW, Washington, DC 20036–5869.

NRC Section Chief: Anthony J. Mendiola.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: December 20, 2000.

Description of amendment request: The proposed amendment request revises Technical Specification (TS) 5.3.1, ''Reactor Core,'' to permit the use of the Framatome Cogema Fuels (FCF) ''M5'' advanced alloy for fuel rod cladding and fuel assembly spacer grids. The licensee has submitted a related exemption request from the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' Section 50.46, ''Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,'' and associated Appendix K, ''ECCS Evaluating Models,'' which presume the use of zircaloy or ZIRLO cladding. A related Bases change is also made to the Bases for TS 2.1.

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

AmerGen has determined that this license amendment request poses no significant hazards considerations as defined by 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. It has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. Further, there are no evaluated accidents in which the fuel cladding or fuel assembly structural components are assumed to arbitrarily fail as an accident initiator. The fuel handling accident assumes that the cladding does, in fact, fail as a result of an undefined fuel handling event. However, the probability of that undefined initiating event is independent of the properties of the fuel rod cladding. Additionally, in both LOCA [loss-of-cooling accident] and non-LOCA accident scenarios, there will be no significant increase in cladding failure or fission product release, since it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. Therefore, this activity does not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

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 previously evaluated. It has been demonstrated that the material propoerties of the M5 alloy are not significantly different than those of Zircaloy-4. Therefore, M5 fuel cladding and the fuel assembly structural components will perform similarly to those fabricated from Zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator. Therefore, this activity does not create the possibility of a new or different kind of accident from any 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 material properties of the M5 alloy are not significantly different from those of Zircaloy-4 for all normal operating and accident scenarios, including both LOCA and non-LOCA scenarios * * * Therefore, this activity 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.

Attorney for licensee: Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

NRC Section Chief: Marsha Gamberoni.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of amendment request: September 14, 2000, as supplemented on December 21, 2000.

Description of amendment request: The licensee proposes to revise the

Technical Specifications to allow a lead fuel assembly (LFA) with a limited number of fuel rods clad with advanced zirconium-based alloys to be inserted into the core during the next refueling outage.

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

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

Calvert Cliffs Technical Specification 4.2.1, Fuel Assemblies, states that fuel rods are clad with either zircaloy or ZIRLO. This reflects the requirements of 10 CFR 50.44, 50.46, and 10 CFR Part 50, Appendix K, which also restricts fuel rod cladding materials to zircaloy or ZIRLO. Calvert Cliffs Nuclear Power Plant, Inc. proposes to insert a fuel assembly into Calvert Cliffs Unit 2 that have some fuel rods clad in zirconium alloys that do not meet the definition of zircaloy or ZIRLO. An exemption to the regulations has also been requested to allow this fuel assembly to be inserted into Unit 2. The proposed change to the Calvert Cliffs Technical Specifications will allow the use of cladding materials that are not zircaloy or ZIRLO for one fuel cycle once the exemption is approved. To obtain approval of new cladding materials, 10 CFR 50.12 requires that the applicant show that the proposed exemption is authorized by law, is consistent with the common defense and security, will not present an undue risk to the public health and safety, and is accompanied by special circumstances. The proposed change to the Technical Specification is effective only as long as the exemption is effective. The addition of what will be an approved temporary exemption to Unit 2 Technical Specification 4.2.1 does not change the probability or consequences of an accident previously evaluated.

Supporting analyses indicate that since the LFA will be placed in a non-limiting location, the placement scheme and the similarity of the advanced alloys to zircaloy-4 will assure that the behavior of the fuel rods with these alloys are bounded by the fuel performance and safety analyses performed for the zircaloy-4 clad fuel rods currently in the Unit 2 core. Therefore, the addition of these advanced claddings does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

The proposed change does not add any new equipment, modify any interfaces with existing equipment, change the equipment's function, or change the method of operating the equipment. The proposed change does not affect normal plant operations or configuration. Since the proposed change does not change the design, configuration, or operation, it could not become an accident initiator.

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

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

The proposed change will add an approved temporary exemption to the Unit 2 Technical Specifications allowing the installation of a lead fuel assembly. This assembly uses advanced cladding materials that are not specifically permitted by existing regulations or Calvert Cliffs' Technical Specifications. A temporary exemption to allow the installation of this assembly has been requested. The addition of an approved temporary exemption to Technical Specification 4.2.1 is simply intended to allow the installation of the lead fuel assembly under the provisions of the temporary exemption. The license amendment is effective only as long as the exemption is effective. This amendment does not change the margin of safety since it only adds a reference to an approved, temporary exemption to the Technical Specifications.

Supporting analyses indicate that since the LFA will be placed in a non-limiting location, the placement scheme and the similarity of the advanced alloys to zircaloy-4 will assure that the behavior of the fuel rods with these alloys are bounded by the fuel performance and safety analyses performed for the zircaloy-4 clad fuel rods currently in the Unit 2 core. Therefore, the addition of these advanced claddings does not involve a significant reduction in the margin of safety.

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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Marsha Gamberoni.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: December 21, 2000.

Description of amendments request: The amendments would revise Technical Specification 5.2.2.e by removing the reference to the Nuclear Regulatory Commission (NRC) Policy Statement on working hours.

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

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

The proposed change to Technical Specification 5.2.2.e only alters the administrative location of, and the regulatory controls applicable to, unit staff-specific overtime limits and working hours. Overtime limits and working hours will remain controlled by plant administrative procedures. Changes to the relocated overtime limits and working hours will be controlled in accordance with our established procedural control processes. There is no increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed.

There is no increase in the radiological consequences of any accident previously evaluated because the proposed amendment does not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed change does not alter the source term, containment isolation, or allowable radiological releases.

Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment to Technical Specification 5.2.2.e only alters the administrative location of and the regulatory controls applicable to unit staff specific overtime limits and working hours. The proposed amendment does not change the way the plant is operated, and no new or different failure modes have been defined for any plant system or component important to safety. No limiting single failure has been identified as a result of the proposed amendment. No new or different types of failures, accident initiators or scenarios are introduced by the proposed amendment.

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

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

Unit staff overtime is not an input into the calculation of any safety margin in the Technical Specification Safety Limits, Limiting Safety Settings, or other Limiting Conditions for Operation. Unit staff overtime is not an input into the calculation of any safety margin in the Technical Requirements Manual, or any other previously defined margins for any structure, system, or component important to safety. The proposed amendment to Technical Specification 5.2.2.e only alters the administrative location of, and the regulatory controls applicable to unit staff-specific overtime limits and working hours.

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

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

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

NRC Section Chief: Marsha Gamberoni.

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: December 1, 2000.

Description of amendments request: The proposed amendments would change the Technical Specification (TS) 5.6.3, ''Radioactive Effluent Release Report'' date for submittal of the Radioactive Effluent Release Report to ''prior to May 1'' of each year.

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

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

The proposed change is administrative in nature. The date of submittal of the Radioactive Effluent Release Report is not an initiator of any analyzed event. Similarly, the date of submission does not affect the consequences of any accident previously evaluated. The proposed change will not physically alter the plant, and it will not affect plant operation. The proposed change to the submission date of the Radioactive Effluent Release Report will continue to meet the reporting requirement of 10 CFR 50.36a(a)(2) and further clarifies when the report is to be submitted. As such, the proposed change does not involve an increase in the probability or consequence of any accident previously evaluated.

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

The proposed TS change is administrative in nature. It would revise the date by which the Radioactive Effluent Release Report is required to be submitted to the NRC. Revision of the submittal date for the report will not affect any accident initiator or cause any new accident precursors to be created. The proposed change will not affect the types or amounts of radioactive effluents released or cumulative occupational radiological exposures.

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

The proposed change to the submittal requirement for the Radioactive Effluent Release Report is only an administrative change and will have no [effect] on any margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 29, 2000.

Description of amendment request: The proposed amendment would modify the applicability statements of Technical Specification (TS) Limiting Conditions for Operations (LCOs) 3.3.6.2, ''Secondary Containment Isolation Instrumentation,'' 3.3.7.1, ''Control Room Emergency Filtration (CREF) System Instrumentation,'' 3.6.4.1, ''Secondary Containment,'' 3.6.4.2, ''Secondary Containment Isolation Valves (SCIVs),'' 3.6.4.3, ''Standby Gas Treatment (SGT) System,'' 3.7.3, ''Control Room Emergency Filtration (CREF) System,'' 3.7.4, ''Control Center Air Conditioning (AC) System,'' 3.8.2, ''AC Sources—Shutdown,'' 3.8.5, ''DC Sources—Shutdown,'' and 3.8.8, ''Distribution Systems—Shutdown.'' The proposed modifications would require operability of the associated systems only if recently irradiated fuel, which is identified as fuel that has occupied part of a critical reactor core within the previous 7 days, is handled during the first few days of an outage. The 7-day value is based on the results of a revised analysis of a fuel handling accident (FHA) that was performed by utilizing the guidelines contained in NRC Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' dated July 2000.

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

The new ''recently irradiated fuel'' term to describe irradiated fuel assemblies is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. Because the equipment affected by the revised operational conditions is not an initiator to any previously analyzed accident, the proposed change cannot increase the probability of any previously evaluated accident.

The re-analysis of the Fuel Handling Accident concludes that radiological consequences are within the acceptance criteria in Regulatory Guide 1.183 (Reference 3 [of the licensee's application dated December 27, 2000]). The results of the Core Alterations events other than the Fuel Handling Accident remain unchanged from the original design basis, which showed that these events do not result in fuel cladding damage or radioactive release. The FHA re-analysis includes a drop of a non-irradiated fuel assembly over recently irradiated assemblies in the reactor core 24 hours after reactor shutdown. The radiological consequences associated with this scenario, assuming no mitigation credit for Secondary Containment, SGT and CREF Systems, have been shown to satisfy the acceptance criteria in Reference 3. Therefore, the proposed changes do not significantly increase the radiological consequences of any previously evaluated accident.

Based on the above, the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

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

The proposed requirements are imposed when specific activities represent situations where significant radioactive releases are not postulated. The proposed requirements are supported by the revised design basis Fuel Handling Accident analysis. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed change does not create the potential for a new or different kind of accident from any accident previously evaluated.

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

The proposed changes revise the Fermi 2 TS[s] to establish operational conditions where specific activities represent situations during which significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the regulatory guidelines. Safety margins and analytical conservatisms are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed TS Applicability statements continue to ensure that the TEDE [total effective dose equivalent] at both the Control Room and the exclusion area and low population zone boundaries are below the corresponding regulatory guidelines in Reference 3 [of the licensee's application dated December 27, 2000]; therefore, the proposed change will not result in 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.

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: Claudia M. Craig.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: December 28, 2000

Description of amendment request: The proposed amendments would revise the Technical Specification requirements associated with storage of spent fuel in the spent fuel storage pools to account for degradation of the Boraflex panels used in the construction of the storage racks and maintain acceptable margins of subcriticality in the spent fuel storage pools.

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

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

Response: No.

The proposed Oconee Nuclear Station (ONS) Technical Specification (TS) changes described in the License Amendment Request (LAR) do not create a significant increase in the probability or consequence of an accident previously evaluated.

The loss of boron from the Boraflex panels in the Spent Fuel Pool (SFP) racks is offset by the presence of soluble boron in the SFP water for criticality control. The increased surveillance frequency provides assurance the SFP boron concentration limits will be maintained. The handling of the fuel assemblies in the SFP has always been performed in borated water. Fuel assembly placement in the revised fuel storage configurations described in the LAR will continue to be controlled by approved fuel handling procedures to ensure compliance with TS requirements.

The proposed changes do not affect the probability of a dropped fuel assembly accident, accidental misloading of spent fuel, or heavy load drop onto the SFP racks. The criticality analyses show the consequences of such events are not affected by the proposed changes and that the fuel will remain subcritical.

The radiological consequences of a fuel misloading or handling accident in the SFP, or a heavy load drop onto the SFP racks, do not change by taking credit for soluble boron in the pool because the current SFP boron concentration limit is unchanged.

In the unlikely event of significant SFP temperature increases or decreases, the proposed soluble boron limits and increased surveillance frequency of the SFP boron concentration provide assurance the fuel will remain subcritical.

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

Response: No.

Criticality accidents in the SFP are not new or different types of accidents. They have been analyzed as described in Section 9.1.2.3.2 of the Updated Final Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments for fuel enrichments up to 5.00 weight percent U–235. The evaluations described in the LAR demonstrate that the proposed changes do not create a new or different kind of accident from any previously analyzed. The accident analysis in the Updated Final Safety Analysis Report remains bounding.

There are no changes in equipment design or in plant configuration. The revised requirement will not result in the installation of any new equipment or modification of any existing equipment. Therefore, the proposed changes will not result in the possibility of a new or different kind of accident.

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

Response: No.

The proposed TS changes and the resulting spent fuel storage operating limits provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on the ONS spent fuel pool-specific criticality analyses described in the LAR.

The criticality analyses are based on the methodology described in WCAP–14416–NP–A, ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'' Revision 1, November 1996, which has been reviewed and approved by the NRC. This methodology takes partial credit for soluble boron in the SFP and meets the following NRC acceptance criteria (10 CFR 50.68) for preventing criticality outside the reactor:

a. k_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level; and

b. k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

The proposed TS limits provide a level of safety comparable to the conservative criticality analysis methodology required by USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG–0800, June 1987, USNRC Spent Fuel Storage Facility Design Bases (for comment) Proposed Revision 2, 1981, Regulatory Guide 1.13, and ANSI/ANS–57.2–1983.

Therefore, the proposed changes will not result in a significant reduction in the plant's margin of safety.

Based on the above evaluations, Duke concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

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.

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard L. Emch, Jr.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: July 21, 2000, as supplemented by letter dated December 13, 2000.

Description of amendment request: The proposed amendment would reduce the limit for reactor coolant system (RCS) specific activity in technical specification (TS) 3/4.4.8. The dose equivalent iodine 131 (I–131) is proposed to be lowered from the current value of ≤ 0.35 micro Curies per gram (μCi/gram) to a value of ≤ 0.20 μCi/gram as specified in TS 3.4.8.a (and associated Actions and Table 4.4–12). This change will also lower the ''''Acceptable Operation'' line on Figure 3.4–1 from 21 μCi/gram to 12 μCi/gram Dose Equivalent I–131 for 80-percent to 100-percent power, and a commensurate reduction for power between 20-percent and 80-percent power.

In conjunction with the reduced TS limit for RCS specific activity, the

Beaver Valley Power Station (BVPS) Unit 1, control room and offsite dose consequences resulting from a postulated Main Steam Line Break have been re-analyzed to allow for higher primary-to-secondary leakage in accordance with methodology described in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes by Outside Diameter Stress Corrosion Cracking,'' and as previously approved in BVPS–1 license amendment number 205.

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

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

The proposed change, which lowers the Technical Specification limit for Dose Equivalent I–131, is conservative and will not adversely affect the current calculated dose values for BVPS Unit 1 Design Basis Accidents (DBAs) since a lower RCS specific activity will lower the calculated dose from any resultant steam generator tube leakage postulated during the DBA. The Standard Review Plan assumption for accident-induced steam generator tube leakage spike remains valid. Thus, the dose listed in the BVPS Unit 1 UFSAR [Updated Final Safety Analysis Report] from those DBAs which calculate and list a dose value in the BVPS Unit 1 UFSAR will remain bounding values, except for the Main Steam Line Break (MSLB) DBA.

The immediate effect upon receiving a revised lower primary coolant specific activity limit in Technical Specification 3.4.8.a would also result in a lower calculated MSLB dose value, if incorporated into the MSLB dose calculation without any other modifications. But the BVPS Unit 1 MSLB analysis is analyzed per GL 95–05 which states that a reduction [in] RCS iodine activity is an acceptable means for accepting higher projected leakage rates and still meeting the applicable limit of Title 10 of the Code of Federal Regulations Part 100 and GDC [General Design Criterion] 19 utilizing currently accepted licensing basis assumptions. Thus, pursuant to this GL 95–05 methodology, the reduced RCS specific activity limit for Technical Specification 3.4.8.a will be used to allow for higher projected leakage rates, while still meeting the applicable regulatory dose limits.

Thus, the current BVPS Unit 1 MSLB calculated dose value will not decrease with a new lower RCS specific activity value in order to allow for a higher projected leakage rates[sic]. However, the BVPS Unit 1 MSLB calculated dose values will remain within the limits specified in 10 CFR 50 Appendix A, GDC 19, and the radiological doses to the public will remain a small fraction of the regulatory limits specified in 10 CFR 100.11, using methodology previously accepted in BVPS Unit 1 License Amendment No. 205.

Therefore, this proposed change will not increase the probability of occurrence of a postulated accident or will not significantly increase the consequences of an accident previously evaluated since the change would continue to comply with the current BVPS Unit 1 and Unit 2 licensing basis as it relates to the dose limits of GDC 19 and 10 CFR Part 100.

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

The proposed license amendment to the primary coolant specific activity limit does not change the way the RCS is operated. The proposed changes only involve changes to the primary coolant specific activity limit where continued power operation may occur. This reduced limit is conservative and does not alter the RCS or steam generators' ability to perform their design bases [functions].

GL 95–05 states that any reduction of RCS specific activity less than 0.35 μCi/gram Dose Equivalent I–131 requires an evaluation of release rate data. This evaluation shows that BVPS Unit 1 RCS Dose Equivalent I–131 data fully supports lowering the Technical Specification RCS specific activity limit to 0.20 μCi/gram without compromising the Standard Review Plan assumption of a post-event iodine spike factor of 500.

Therefore, this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated accident since the RCS and steam generator will continue to operate in accordance with their design bases.

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

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not adversely affect the ability of systems, structures or components important to the mitigation and control of design bases accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling conditions for extended periods of time.

The proposed license amendment to the primary coolant specific activity limit does not adversely change the way the RCS or steam generators are operated. This modification does not alter these systems' ability to perform their design bases [functions]. The existing safety analyses remain bounding. Therefore, the margin of safety is not significantly reduced.

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

Attorney for Licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Marsha Gamberoni.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: December 4, 2000.

Description of amendment request: The proposed license amendment would revise the St. Lucie Unit 1 Updated Safety Analysis Report to reflect the new main steam line break (MSLB) analysis treatment of a hypothesized single failure of a main feedwater isolation valve (MFIV). The new analysis of the MSLB terminates feedwater addition to the faulted steam generator by crediting MFIV closure and tripping the main feedwater (MFW) and condensate pumps.

This proposed change to the Unit 1 licensing bases for the MSLB analysis is required to resolve an existing Generic Letter 91–18 degraded, but operable, condition regarding the postulated peak pressure during an MSLB inside containment. In December 1998 the draft results of a Unit 1 MSLB containment re-analysis indicated an unexpected higher peak containment pressure of 55.9 psig. The Unit 1 containment design pressure is 44 psig. The cause for the higher peak pressure in the re-analyzed MSLB event is that non-conservative assumptions were used in the original analysis of record. When these non-conservatisms were corrected and input to the MSLB licensing bases analysis, the containment peak pressure exceeded the containment design pressure. This condition was reported to the NRC via Licensee Event Report No. 50–335/1998–009.

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.

This amendment changes the licensing bases for the MSLB analysis to credit a trip of the non-safety MFW and condensate pumps as a backup method to terminate feedwater addition should a MFIV fail to close. This activity has no increase in the probability of a MSLB, as no physical changes are being made to the steam generators, main steam piping, and the normal operating temperatures and pressures for the main steam system remain unchanged. This activity also has no adverse effect on the consequences of an accident because the MSLB containment response is bounded by the new analysis. Main feedwater termination occurs during a postulated MSLB such that the containment design pressure is not exceeded. Although a circuit failure (short) in the MSIS [main steam isolation signal] backup trip of the MFW and condensate pump breakers would result in tripping the running MFW and condensate pumps, this is less probable due to the energized to actuate design than existing postulated failures in the MSIS circuitry that would also lead to a loss of feedwater event. Therefore, 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.

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.

This amendment changes the licensing bases for the MSLB analysis to credit a trip of the non-safety MFW and condensate pumps as a backup method to terminate feedwater addition should a MFIV fail to close. The physical modifications made to support the installation of the new pneumatic valve operators for the MFIVs and installation of the backup main steam isolation signal (MSIS) trip of the non-safety MFW and condensate pumps conform to all applicable design standards. Failure modes introduced by these changes are bounded by the original design, and no other physical changes were made to the plant. Therefore, 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.

This amendment changes the licensing bases for the MSLB inside containment response analysis to credit a trip of the non-safety MFW and condensate pumps as a backup method to terminate feedwater addition should a MFIV fail to close. This differs from the currently licensed analysis that credits the closure of redundant safety related valves for main feedwater termination single failure considerations. However, Sections 6.2.1.4 and 15.1.5 of the Standard Review Plan allows the use of a non-safety backup in response to a failure of safety related components with regards to mitigating the effects of the mass energy release of ruptured secondary piping inside containment. This change to the licensing bases is consistent with the guidance provided in the Standard Review Plan. In addition, a probabilistic safety assessment was performed to evaluate the change in main feedwater isolation reliability between crediting redundant safety related isolation valves or safety related isolation valves and trip of the non-safety MFW and condensate pumps. This assessment concluded that the change in reliability is not risk significant. Therefore, 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.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company, Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: November 28, 2000.

Description of amendment request: The proposed license amendment would revise the Updated Final Safety Analysis Report (UFSAR) with a revised post-trip steam line break (SLB) analysis. The design basis for the current analysis of record ensures that no fuel failure will occur for all post-trip SLB cases. The new analysis supports a change to the fuel failure criterion, to limit fuel failure to less than or equal to 2%. The change in allowed fuel failure fraction results in a shutdown margin benefit and provides additional flexibility in the core design. Limits for the physics parameters that most affect the post-trip SLB results will be established on a core-specific basis and included in the Core Operating Limits Report for each cycle. The revised analysis, with the limit of 2% fuel failure, continues to meet the 10 CFR part 100 dose criteria.

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 proposed amendment revises the post-trip SLB analysis of record to support a fuel failure limit of 2% as compared to the current criterion of no fuel failure. Post-trip SLB is a current design basis event for St. Lucie Unit 2 and is defined in the St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR). The revision to the analysis does not impact the event initiator and requires no change to any plant component or system. The plant configuration remains unchanged, and thus the probability of occurrence of previously analyzed accidents is not affected by the proposed change.

Radiological consequences for the return to power (RTP) SLB event for St. Lucie Unit 2 have been calculated to infer the allowed fuel failure fraction from the 2-hour and 8-hour 10 CFR 100 dose limits and are consistent with the results presented in License Amendment 105. Releases were calculated based on fuel that violates Centerline-Melt (CTM) criteria and produces fuel failure limits of 13.5% for inside containment SLB and 3.4% fuel failure for an outside containment SLB.

These fuel failure values represent an upper bound limit corresponding to the 10 CFR 100 dose criteria. A conservative value of 2% fuel failures (from violation of CTM and/or departure nucleate boiling ratio (DNBR) specified acceptable fuel design limits (SAFDL)) will be utilized as a cycle specific limit for post-trip SLB. The peak power density during the post-trip SLB also will be limited to less than or equal to 30 kW/ft. For each fuel cycle core design, these limits will be verified based on the calculated physics data for that cycle.

The limit of 2% fuel failures, in conjunction with the 30 kW/ft on peak power density, ensures a coolable geometry during and subsequent to the post-trip SLB RTP. This fuel failure limit, along with a conservative allowance for DNB propagation failures, remains well below the upper bound limits of 13.5% and 3.4% fuel failure for the inside and the outside containment breaks, respectively, corresponding to the 10 CFR 100 dose criteria.

Therefore, 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.

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

The proposed amendment is merely a revision to the post-trip SLB event analysis, which continues to meet the applicable limits of 10 CFR 100 dose criteria. There is no change to the plant configuration, systems, or components that would create new failure modes. The modes of operation of the plant remain unchanged.

Therefore, 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. Use of the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendment revises the post-trip SLB analysis and supports a change to the fuel failure acceptance criterion. The revised analysis, with the limit of 2% fuel failure, would continue to provide margin to the applicable limits of 10 CFR 100 dose criteria. The proposed change, including any core design variations, will have no adverse impact on other plant safety analysis. The plant operation would continue to remain within all design basis requirements, which would ensure that a safety margin to the acceptance criteria would continue to remain available during plant operation at all power levels.

Therefore, 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.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 6, 2000.

Description of amendment request: The proposed license amendments would revise the Turkey Point Units 3 and 4 Technical Specification (TS) Surveillance Requirement (SR) 4.6.1.3.c to allow performance of the required surveillance test of the air lock interlock at an interval of 24 months. Currently, SR 4.6.1.3.c requires the interlocks to be tested at least once every six month, and is, therefore, done with the plant online.

Each containment at Turkey Point has two air locks, commonly named the personnel air lock and the escape hatch. Each air lock has an inner and an outer door. Interlocks prevent both doors in the air lock from being opened at the same time, thereby preserving containment integrity, when required. These interlocks are completely mechanical, and contain no degradable components. Historically, the air lock interlock test frequency was chosen to coincide with that of the overall airlock leakage test. Turkey Point Units 3 and 4 TSs were amended in January 1997, to permit the extension of the overall airlock leakage test frequency up to a maximum of 30 months, based on acceptable test results. The licensee requested revision of SR 4.6.1.3.c. to require testing of the air lock interlock at an interval of 24 months, which would also allow a maximum interval of up to 30 months between tests. Therefore, the proposed amendments would realign the SR frequencies of the air lock interlock test and the overall leakage test with each other. In support of these amendments, the licensee stated that currently the SR test is being performed with the plant online, when the interlocks are required to be operable. If the proposed amendments are granted, the licensee expects to perform the test during refueling outages, when the plant is in a mode in which the interlock is not required to be operable. Also, the licensee stated that the proposed amendments are consistent with the as-low-as-reasonably-achievable principles, because they would preclude performance of the test with the plant online, which involves some risk of dose to workers.

Additionally, the licensee requested to amend TS 3.3.2, Table 3.3–2, Item 1.e, that addresses the requirements for the safety injection signal (SIS) generated by high steamline differential pressure, to change the asterisk following Modes 1, 2, 3, to a pound sign (i.e., #). The licensee stated that the existing asterisk refers to an incorrect note, in that it indicates that the SIS may be blocked below the Tavg—Low Interlock Setpoint, when in fact the Block Permissive for this SIS is pressurizer pressure below 2000 psi. The licensee stated that this is due to a typographical error, and that the change is requested to make the Mode Applicability consistent with the design of the protection logic.

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 proposed changes allow performance of the required surveillance at the same frequency as the performance of the air lock overall leakage surveillance. The proposed relaxation in surveillance frequency will not impact the initiating event for any previously evaluated accident. The correction of the typographical error has no impact on any accident analysis. The proposed changes do not affect any of the assumptions made or methodologies used for any accident analysis. Thus the proposed changes have no impact on any of the accident probabilities or consequences. Therefore, the proposed amendments do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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 previously evaluated.

The proposed changes do not alter the design, physical configuration, or modes of operation of the plant. No changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed Technical Specification changes do not impact any other plant systems. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not change the operation, function, or modes of plant or equipment operation. The proposed changes do not change the level of assurance of containment integrity. Plant processes and training preclude challenges to the air lock interlocks. The correction of the typographical error has no impact on any margin of safety. Therefore, operation of the facility in accordance with the proposed amendments 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.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: December 5, 2000.

Description of amendment request: The proposed amendment would implement programmatic controls for radiological effluent technical specifications (RETS) in the administrative section of the Technical Specifications (TSs) and relocate the procedural details of the RETS to the offsite dose calculation manual (ODCM), the process control program (PCP), or other new programs, consistent with the guidance of Standard TSs (STS) (NUREG–1433) and NRC Generic Letter 89–01.

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

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

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes are administrative in nature and alter only the format and location of programmatic controls and procedural details relative to radioactive effluents, radiological environmental monitoring, radioactive source leakage testing, solid radioactive wastes, and associated reporting requirements. Existing TS containing procedural details on radioactive effluents, radiological environmental monitoring, radioactive source leakage testing, explosive gas monitoring, storage tank radioactive content limits, solid radioactive wastes and associated reporting requirements are being relocated to the ODCM, PCP or other new programs as appropriate. Compliance with applicable regulatory requirements will continue to be maintained. In addition, the proposed changes do not alter the conditions or assumptions in any of the previous accident analyses. Since the previous accident analyses remain bounding, the radiological consequences previously evaluated are not adversely affected by the proposed changes.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes. Also, there will be no change in types or increase in the amounts of any effluents released offsite.

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

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

The proposed changes do not involve a significant reduction in a margin of safety. The proposed changes do not involve any actual change in the methodology used in the control of radioactive effluents, radioactive sources, solid radioactive wastes, or radiological environmental monitoring. These changes are considered administrative in nature and provide for the relocation of procedural details outside of the technical specifications but add appropriate administrative controls to provide continued assurance of compliance to applicable regulatory requirements. These proposed changes also comply with the guidance contained in Generic Letter 89–01 and the STS.

Therefore, it can be concluded a significant reduction in the 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Claudia M. Craig.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: January 10, 2001.

Description of amendment request: The proposed amendment would remove the standby liquid control pump flow surveillance requirement to recycle demineralized water to the test tank and change the testing frequency from monthly to quarterly.

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

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

The only significant consequence of these changes compared to present plant operation will be to change the test frequency of the MNGP [Monticello Nuclear Generating Plant] SLC [standby liquid control] pump capacity test to quarterly, which has been previously reviewed and approved by the NRC staff for similar boiling water reactors (BWRs). There are no changes to equipment performance or postulated failure modes. The change does not affect the assumptions or methods of accident mitigation previously evaluated. The proposed amendment will have no impact on the probability or consequences of an accident.

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

The only significant consequence of these changes compared to present plant operation will be to change the test frequency of the MNGP SLC capacity flow test to quarterly, which has been previously reviewed and approved by the NRC staff for similar BWRs. The change does not affect or introduce any new plant operating modes. The changes do not alter any existing system interaction and do not introduce any new failure modes. The proposed amendment will not create the possibility for any new or different accidents for those previously analyzed.

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

The only significant consequence of these changes compared to present plant operation will be to change the test frequency of the MNGP SLC pump capacity test to quarterly, which has been previously reviewed and approved by the NRC staff for similar BWRs. There is no change in the reliability or performance of the SLC system. Other surveillance requirements assure that SLC hydraulic conditions will not degrade between quarterly surveillances. The proposed changes have no effect on the mitigation of any postulated accident or event at MNGP. The proposed Technical Specification changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Claudia M. Craig.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: November 20, 2000.

Description of amendment request: The proposed amendments will implement changes to the Technical Specifications to increase the allowable deviation in individual rod position indication (IRPI). The portion of this amendment that pertains to control rod misalignment above 85 percent as a function of peaking factors will be reviewed by the NRC staff as a separate action.

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 Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

Based on the analyses documented in WCAP–15432, Revision 1, all pertinent licensing-basis acceptance criteria have been met and the margin of safety, as defined in the Technical Specification Bases, is not significantly reduced in any of the Point Beach licensing basis accident analyses based on the subject change. Therefore, the probability of an accident previously evaluated has not significantly increased. Because design limitations continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Neither rod position indication nor the limits on allowed rod position deviation is an accident initiator or precursor. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

Based on the analyses documented in WCAP–15432, Revision 1, all pertinent licensing-basis acceptance criteria have been met and the margin of safety, as defined in the Technical Specification Bases, is not significantly reduced in any of the Point Beach licensing basis accident analyses based on the subject change.

The possibility for a new or different type of accident from any accident previously evaluated is not created as a result of this amendment. The changes described in the amendment are supported by the analyses and evaluations described in Attachment 2 of this letter (safety evaluation) [licensee's application dated November 20, 2000]. The evaluation of the effects of the proposed changes indicate that all design standards and applicable safety criteria limits are met. These changes therefore do not cause the initiation of any new or different accident nor create any new failure mechanisms.

All equipment important to safety will continue to operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

Based on the analyses documented in WCAP–15432, Revision 1, all pertinent licensing-basis acceptance criteria have been met and the margin of safety, as defined in the Technical Specification Bases, is not significantly reduced in any of the Point Beach licensing basis accident analyses based on the subject changes to safety analyses input parameter values. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. Since the safety evaluation in Attachment 2 of this letter [licensee's application dated November 20, 2000] demonstrates that all applicable acceptance criteria continue to be met, the subject operating conditions will not involve a significant reduction in a margin of safety at Point Beach.

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.

Attorney for licensee: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Claudia M. Craig.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: December 20, 2000.

Description of amendment request: The proposed changes would revise the Technical Specifications and Technical Requirements Manual requirements applicable when actions direct suspension of operations involving positive reactivity changes, by removing the requirement not to make positive reactivity changes during certain plant conditions, and by limiting the amount of reactivity changes that are allowed to those that will continue to assure appropriate reactivity limits are met. Related changes to the Bases are also proposed. In addition, an administrative change is also proposed to remove a footnote that allowed an alternate onsite emergency power source to be substituted for one of the required diesel generators for 21 consecutive days for refueling outages 1RE05 and 2RE04 only.

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

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

The proposed change does not involve an increase in the probability or consequences of an accident previously evaluated. The proposed activities to be allowed during certain operating conditions are permitted at other times during routine operating conditions. The changes do not affect the limits on reactivity that are specified in other specifications. The proposed changes do not reduce restrictions on addition or flowpaths of unborated water that are in the existing specifications. The proposed change does not affect the limits on reactivity that are credited in the safety analysis. Therefore, no increase in the probability or consequences of any accident previously evaluated will occur.

In addition to the changes proposed to controls over reactivity changes, an administrative change is proposed to remove a footnote that is no longer applicable to the facility. Since the footnote no longer has meaning or relevance to the operation of the facility, its removal does not increase the probability or consequences of any accident previously evaluated.

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

The proposed changes merely permit the conduct of normal operating evolutions during limited periods when additional controls over reactivity margin are imposed by the Technical Specifications. The proposed change does not introduce any new equipment into the plant or significantly alter the manner in which existing equipment will be operated. The changes to operating allowances are minor and are only applicable during certain conditions. The operating allowances are consistent with those acceptable at other times. Since the proposed changes only allow activities that are presently approved and routinely conducted, no possibility exists for a new or different kind of accident from those previously evaluated.

In addition to the changes proposed to controls over reactivity changes, an administrative change is proposed to remove a footnote that is no longer applicable to the facility. Since the footnote no longer has meaning or relevance to the operation of the facility, its removal cannot create the possibility of a new or different kind of accident from those previously evaluated.

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

The proposed changes do not involve a significant reduction in a margin of safety because the ability to make the reactor subcritical and maintain it subcritical during all operating conditions and modes of operation will be maintained. The margin of safety is defined by the shutdown margin limits and the refueling boron concentration limit. The proposed changes do not affect these operating restrictions and the margin of safety which assures the ability to make and maintain the reactor subcritical is not affected.

In addition to the changes proposed to controls over reactivity changes, an administrative change is proposed to remove a footnote that is no longer applicable to the facility. Since the footnote no longer has meaning or relevance to the operation of the facility, its removal cannot result in a reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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: October 6, 2000.

Description of amendment request: The proposed amendment would revise each of the three units' Technical Specifications (TS) to provide action requirements and completion times for use under plant conditions involving one inoperable low pressure coolant injection (LPCI) pump in each of the two emergency core cooling system (ECCS) divisions. The new requirements are consistent with those currently specified for use under conditions of two inoperable LPCI pumps in the same ECCS division.

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

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

The LPCI system consists of two independent LPCI subsystems. Each of the two LPCI subsystems has two LPCI pumps. The current TS Limiting Conditions for Operation are based on ECCS analyses that postulate the failure of one entire subsystem. New analyses have been performed that postulate the failure of one LPCI pump in each subsystem (i.e., both subsystems operating at reduced capacity). The new analyses show that the total ECCS flow capacity provided by the entire LPCI system is greater when operating under the newly-analyzed conditions than for the previously analyzed conditions. Thus, the action requirements and completion times associated with inoperability of two LPCI pumps in on the same LPCI subsystem may be applied in cases when one LPCI pump is inoperable in each LPCI subsystem. Since ECCS performance is not adversely affected, there is no increase in the probability or consequences of any analyzed accident.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment or require any existing equipment to be operated in a manner different from the present design. The proposed change will not impose any new or eliminate any existing requirements.

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

The proposed change will not reduce a margin of safety because it has no adverse effect on any safety analyses assumptions. The proposed new Conditions involving one inoperable LPCI pump in each LPCI injection subsystem represent more reliable configurations than the existing LCOs which apply for two inoperable LPCI pumps in one ECCS subsystem.

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.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of application for amendments: January 22, 2001 (TS 00–01).

Brief description of amendments: The proposed amendments would change the Sequoyah Nuclear Plant Technical Specification surveillance requirements for assuring against ice condenser flow blockage.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), Tennessee Valley Authority (TVA), 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 only analyzed accidents of possible consideration in regards to changes potentially affecting the ice condenser are a loss-of-coolant accident (LOCA) and a high energy line break (HELB) inside containment. However, the ice condenser is not postulated as being the initiator of any LOCA or HELB. This is because it is designed to remain functional following a design basis earthquake, and the ice condenser does not interconnect or interact with any systems that interconnect or interact with the reactor coolant or main steam systems.

Neither the TS [Technical Specification] amendment nor the TS Bases changes can increase the probability of occurrence of any analyzed accident because they are not the result or cause of any physical modification to ice condenser structures, and for the current design of the ice condenser, there is no correlation between any credible failure of it and the initiation of any previously analyzed event.

Regarding the consequences of analyzed accidents, the ice condenser is an engineered safety feature designed, in part, to limit the containment subcompartment and steel containment vessel pressures immediately following the initiation of a LOCA or HELB. Conservative subcompartment pressure analysis shows this criteria will be met if the reduction in the flow area per bay provided for ice condenser air and or steam flow channels is less than or equal to 15 percent, or if the total flow area blocked within each lumped analysis section is less than or equal to the 15 percent as assumed in the safety analysis.

The proposed amendment also revises the flow area verification surveillance frequency from at least once per 12 months to at least once per 18 months such that it will coincide with refueling outages. Management of ice condenser maintenance activities has successfully limited activities, with the potential for significant flow channel degradation, to the refueling outage. Verifying an ice bed is left with less than or equal to 15 percent flow channel blockage at the conclusion of a refueling outage assures the ice bed will remain in an acceptable condition for the duration of the operating cycle. During the operating cycle, a certain amount of ice sublimates and reforms as frost on the colder surfaces in the ice condenser. However, frost does not degrade the flow channel flow area. The surveillance will effectively demonstrate operability for an allowed 18-month surveillance period. Therefore, increasing the surveillance interval does not affect the ice condenser operation or accident response. Limiting ice bed flow channel blockage to less than or equal to 15 percent ensures operation is consistent with the assumptions of the DBA analyses. Thus, the proposed amendment for flow blockage determination provides the necessary assurance that flow channel requirements are met without additional evaluations and thus will not increase the consequences of a LOCA or HELB.

In regard to [the] TS 3.6.5.3 Bases change, clarifying the action entry of Action b to not apply when personnel are standing on or opening doors for a short duration to perform surveillances or minor maintenance activities, such as ice removal, does not increase analyzed accident consequences. These are not new or additional actions compared to those performed previously, the probability of an accident versus the time to perform these actions is small, the number of personnel involved is small, and their duration is generally much less than the four-hour frequency of required Action b (monitor maximum ice condenser temperature). Therefore, these activities do not adversely affect ice bed sublimation, melting, or ice condenser flow channels. However, if during these activities any door is determined to be restrained, not fully closed from a previous activity, or otherwise not operable, then separate entry into Action b is required.

Thus, based on the above, the proposed changes do not involve a significant increase in 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.

Because the TS and [TS] Bases changes do not involve any physical changes to the ice condenser, or make any changes in the operational or maintenance aspects of the ice condenser as required by the TSs, there can be no new accidents created from those already identified and evaluated.

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

Design basis accident analysis have shown that with 85 percent of the total flow area available (uniformly distributed), the ice condenser will perform its intended function. Thus, the safety limit for ice condenser operability is a maximum 15 percent blockage of flow channels. Surveillance Requirement (SR) 4.6.5.1 currently applies the 15 percent flow blockage criteria to the total flow area of each bay which includes flow passages between the ice baskets, past lattice frames, through intermediate and top deck floor grating, or past lower plenum support structures and turning vanes. This application of the criteria does not have direct correlation to the safety limit for blockage of ice condenser flow channels (those areas that comprise the area between ice baskets, and past lattice frames and wall panels). Changing the TS to implement a surveillance program that uses acceptance criteria consistent with the transient mass distribution (TMD) analysis will not reduce the margin of safety.

Additionally, verifying an ice bed is left with less than or equal to 15 percent flow channel blockage at the end of a refueling outage assures the ice bed will remain in an acceptable condition for the duration of the operating cycle. During the operating cycle, a certain amount of ice sublimates and reforms as frost on the colder surfaces in the ice condenser. However, frost has been determined to not degrade the flow channel flow area. Thus, design limits for the continued safe function of containment subcompartment walls and the steel containment vessel are not exceeded due to this change.

The change made to TS 3.6.5.3 Bases does not affect the margin of safety as defined in any TS as it does not involve design specifications or acceptance criteria. This change only adds a clarifying note that entry into Action b is not required solely because of actions (standing on and opening intermediate/upper deck doors) necessary for the performance of required ice condenser surveillances, maintenance, or routine activities. This does not preclude entry into Action b during performance of these activities should an intermediate deck door or upper deck door otherwise be determined inoperable.

The NRC 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.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: February 28, 2000, as supplemented by letters dated May 12, May 24, June 1 and June 28, 2000.

Brief description of amendment: The amendments revise certain license conditions to reflect the change in ownership interest from PECO to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

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

Amendment No.: 137.

Facility Operating License No. NPF–62: The amendment revised the License.

*Date of initial notice in **Federal Register:*** April 11, 2000 (65 FR 19396). The May 12, May 24, June 1, and June 28, 2000, supplemental letters provided additional clarifying information and did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: October 6, 2000 (U–603332).

Brief description of amendment: The amendment removes from the Technical Specification surveillance requirements the minimum operating time specified for the containment/drywell hydrogen mixing system.

Date of issuance: January 25, 2001.

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

Amendment No.: 138.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 29, 2000 (65 FR 71132) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 25, 2001.

No significant hazards consideration comments recieved: No.

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: December 1, 1999, as supplemented on September 15, 2000.

Brief description of amendment: The proposed amendment revised the Technical Specifications to change the standard by which you test charcoal used in engineered safeguards features systems to American Society for Testing and Materials D3803–1989. These revisions are made in accordance with Generic Letter 99–02.

Date of Issuance: January 24, 2001.

Effective date: January 24, 2001 and shall be implemented within 30 days of issuance.

Amendment No.: 219.

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

*Date of initial notice in **Federal Register:*** May 17, 2000 (65 FR 31357)

The September 15, 2000, letter provided clarifying information within the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 24, 2001.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: December 20, 1999, as supplemented on January 14, March 10, March 23, March 29, and June 16, 2000.

Brief description of amendments: The amendments revise the licenses and technical specifications to reflect the transfer of the licenses from Commonwealth Edison Company to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

Effective date: Immediately to be implemented within 30 days.

Amendment Nos.: 109 & 115.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Licenses and Technical Specifications.

*Date of initial notice in **Federal Register:*** March 9, 2000 (65 FR 12583) and (65 FR 12584). The March 10, March 23, March 29, and June 16, 2000, supplemental letters provided additional clarifying information and did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. 50–10, 50–237, and 50–249, Dresden Nuclear Power Station, Units 1, 2, and 3, Grundy County, Illinois

Date of application for amendments: December 20, 1999, as supplemented January 14, March 10, March 23, March 29, and June 16 2000.

Brief description of amendments: The amendments revise the licenses to reflect the transfer of the licenses from Commonwealth Edison Company to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 40, 183, and 178.

Facility Operating License Nos. DPR–2, DPR–19 and DPR–25: The amendments revised the Licenses to reflect the transfer of the licenses from Commonwealth Edison Company to Exelon Generation Company, LLC.

*Date of initial notice in **Federal Register:*** March 9, 2000 (65 FR 12582).

The March 10, March 23, March 29 and June 16, 2000 letters are within the scope of the original notice and did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No

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

Date of application for amendments: December 20, 1999, as supplemented January 14, March 10, March 23, March 29, and June 16, 2000.

Brief description of amendments: The amendments revise the licenses and technical specifications to reflect the transfer of the license from Commonwealth Edison Company to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

Effective date: Immediately to be implemented within 30 days.

Amendment Nos.: 146 and 132.

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

*Date of initial notice in **Federal Register:*** March 9, 2000 (65 FR 12585). The March 10, March 23, March 29, and June 16, 2000, supplemental letters provided additional clarifying information and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 20, 1999, as supplemented January 14, March 10, March 23, March 29, and June 16, 2000.

Brief description of amendments: The amendments revise the license to reflect the transfer of the license from Commonwealth Edison Company to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

Effective date: January 12, 2001.

Amendment Nos.: 197 and 193.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Licenses.

*Date of initial notice in **Federal Register:*** March 9, 2000 (65 FR 12581). The March 10, March 23, March 29, and June 16, 2000, supplemental letters provided additional clarifying information and did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 20, 1999, as supplemented January 14, March 10, March 23, March 29, and June 16, 2000.

Brief description of amendments: The amendments revised the operating licenses to reflect the transfer of the licenses from Commonwealth Edison Company to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

Effective date: January 12, 2001, to be implemented within 30 days from the date of issuance.

Amendment Nos.: 181 and 168.

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Operating Licenses.

*Date of initial notice in **Federal Register:*** March 9, 2000 (65 FR 12586).

The March 10, March 23, March 29, and June 16, 2000, supplemental letters provided additional clarifying information and did not change the staff's original no significant hazard consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: November 17, 1999, as supplemented June 14, November 13 and December 4, 2000.

Brief description of amendment: Increased the allowed outage time to restore an inoperable emergency diesel generator set to operable status from 72 hours to 14 days.

Date of Issuance: January 19, 2001.

Effective Date: January 19, 2001.

Amendment No.: 170.

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

*Date of initial notice in **Federal Register:*** December 15, 1999 (65 FR 70089). The June 14, November 13, and December 4, 2000, supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 19, 2001.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: July 19, 2000.

Brief description of amendment: Revised the license: (1) to implement Siemens Power Corporation (SPC) high thermal performance fuel assembly design in Cycle 17, (2) relocate shutdown margin requirements in Modes 1 to 5 to the Core Operating Limits Report (COLR), (3) update the COLR methodologies listed in the Technical Specification (TS) Section 6.9.1.11, and (4) request relief from the SPC fuel assembly reconstitution restrictions for peripheral low power fuel assemblies. Additionally, administrative changes were made to the boron concentration specifications related to the boration requirements.

Date of Issuance: January 25, 2001.

Effective Date: January 25, 2001.

Amendment No.: 171.

Facility Operating License No. DPR–67: Amendment revised the TSs.

*Date of initial notice in **Federal Register:*** August 9, 2000 (65 FR 48748).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 25, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 19, 2000.

Brief description of amendment: The amendment changes Technical Specifications (TS) 3.8.4.1, ''Electrical Power System—Containment Penetration Conductor Overcurrent Protective Devices;'' 3.8.4.2.1, ''Electrical Power Systems—Motor-Operated Valves Thermal Overload Protections;'' and 3.8.4.2.2, ''Electrical Power Systems—Motor-Operated Valves Thermal Overload Protection Not Bypassed.'' The proposed changes would relocate the requirements for containment penetration conductor overcurrent and motor-operated valve thermal overload protective devices from the TS to the licensee's Technical Requirements Manual (TRM). The Bases for these TSs would also be relocated to the TRM.

Date of issuance: January 16, 2001.

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

Amendment No.: 192.

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

*Date of initial notice in **Federal Register:*** August 23, 2000 (65 FR 51360).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: September 19, 2000.

Brief description of amendment: The amendment revises the Standby Liquid Control boron solution requirements in TS Figure 3.1.7–1 to ensure a minimum boron concentration of 660 parts per million in the reactor.

Date of issuance: January 23, 2001.

Effective date: As of the date of issuance and shall be implemented before entering Mode 2 during Cycle 18.

Amendment No.: 236.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 1, 2000 (65 FR 65343).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 2001.

No significant hazards consideration comments received: No.

PECO Energy Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: October 14, 1999, as supplemented February 11, September 22, and October 18, 2000.

Brief description of amendment: This amendment revised TS Section 2.2, ''Safety Limits and Limiting Safety Systems Settings,'' and TS Section 3.0/4.0, ''Limiting Conditions for Operation and Surveillance Requirements.'' These revisions will support the installation of LGS Modification P00224 for Unit 2, which will install a Power Range Neutron Monitoring System and incorporate long-term thermal-hydraulic stability solution hardware.

Date of issuance: January 16, 2001.

Effective date: As of date of issuance and shall be implemented during the Limerick Unit 2 refueling outage scheduled to begin in the spring of 2001.

Amendment No.: 109.

Facility Operating License No. NPF–85. This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 1, 1999 (64 FR 67337). The February 11, September 22, and October 18, 2000, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2001.

No significant hazards consideration comments received: No.

PECO Energy Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: December 20, 1999, as supplemented January 3, February 14, March 10, March 23, March 30, and June 15, 2000.

Brief description of amendments: The amendments revised the licenses for Limerick Units 1 and 2 to reflect the transfer of PECO's ownership of these units to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

Effective date: As of date of issuance and shall be implemented within 30 days.

Amendment Nos.: 147 and 108.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the license.

Date of initial notice in ***Federal Register:*** March 9, 2000 (65 FR 12587).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: December 20, 1999, as supplemented December 22, 1999, January 3, February 14, March 10, March 23, March 30, and June 15, 2000.

Brief description of amendments: The amendments revise the licenses to reflect the transfer of PECO Energy Company's ownership interest in the Salem Nuclear Generating Station, Unit Nos. 1 and 2, to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

Effective date: January 12, 2001.

Amendment Nos.: 241 & 222.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Facility Operating Licenses.

Date of initial notice in ***Federal Register:*** March 9, 2000 (65 FR 12591). The December 22, 1999, January 3, February 14, March 10, March 23, March 30, and June 15, 2000, supplements did not expand the scope of the original application with respect to both the proposed transfer action and the proposed amendment action as initially noticed in the **Federal Register**. No hearing requests or comments were received. In addition, the submittal did not affect the applicability of the Commission's generic no significant hazards consideration determination set forth in 10 CFR 2.1315.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 30, 2000.

Brief description of amendments: The amendments revised main steam isolation valve surveillance testing requirements. Specifically, the amendments permit use of the minimum pathway leakage value for the ''as-found'' test limit.

Date of issuance: January 24, 2001.

Effective date: January 24, 2001.

Amendment Nos.: 267 and 227.

Facility Operating License Nos. DPR–52 and DPR–68: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 29, 2000.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 24, 2001.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: June 7, 2000, as supplemented June 23, August 24, September 26, October 6, October 27 and November 16, 2000.

Brief description of amendment: The amendment changes the Facility Operating License (FOL) and the Technical Specifications (TS) to reflect an increase in the full core power rating from 3411 to 3459 megawatts thermal.

Date of issuance: January 19, 2001.

Effective date: January 19, 2001.

Amendment No.: 31.

Facility Operating License No. NPF–90: Amendment revises the FOL and TS.

Date of initial notice in ***Federal Register:*** September 7, 2000 (65 FR 54322).

The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated November 21, 2000 and in a Safety Evaluation dated January 19, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 25, 2000.

Brief description of amendment: The amendment makes editorial and administrative changes to the Technical Specifications (TSs). These changes correct spelling and grammatical errors, correct references, eliminate excessive detail related to specifying a job title, revise position titles, consolidate pages and generalize statements allowing U.S. Nuclear Regulatory Commission (NRC) approved alternatives to specified requirements.

Date of Issuance: January 23, 2001.

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

Amendment No.: 196.

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

Date of initial notice in ***Federal Register:*** November 29, 2000 (65 FR 71140).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 23, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 31st day of January 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–3028 Filed 2–6–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Consolidated Guidance About Materials Licenses: Guidance About Administrative Licensing Procedures

AGENCY: Nuclear Regulatory Commission (NRC).

ACTION: Notice of Availability of final NUREG.

SUMMARY: The NRC is announcing the availability of the final NUREG–1556, Volume 20, ''Consolidated Guidance about Materials Licenses: Guidance about Administrative Licensing Procedures,'' dated December 2000.

The NRC is using Business Process Redesign techniques to redesign its materials licensing process, as described in NUREG–1539, ''Methodology and Findings of the NRC's Materials Licensing Process Redesign.'' A critical element of the new process is consolidating and updating numerous guidance documents into a NUREG-series of reports. This final NUREG report is the 20th guidance document developed for the new process.

This guidance is intended for use by the NRC staff, and will also be available to Agreement States, applicants, and licensees. This document combines and updates the guidance for NRC license reviewers and licensing assistants previously found in the documents listed in Appendix A of the NUREG. NRC licensing staff will use these administrative procedures to process license applications and prepare licenses.

A free single copy of final NUREG–1556, Volume 20, may be requested by writing to the U.S. Nuclear Regulatory Commission, ATTN: Mrs. Carrie Brown, Mail Stop TWFN 9–F–31, Washington, DC. 20555–0001. Alternatively, submit requests through the Internet by