NRC Form 540 and 540A: 13,400
NRC Form 541 and 541A: 13,400
NRC Form 542 and 542A: 756

7. *The number of annual respondents:*

NRC Form 540 and 540A: 2,500 licensees
NRC Form 541 and 541A: 2,500 licensees
NRC Form 542 and 542A: 22 licensees

8. *An estimate of the total number of hours needed annually to complete the requirement or request:*

NRC Form 540 and 540A: 2,238 (.2 hours per response)
NRC Form 541 and 541A: 2,238 (.2 hours per response)
NRC Form 542 and 542A: 126 (.2 hours per response)

9. *An indication of whether section 3507(d), Pub. L. 104–13 applies:* Not applicable.

10. *Abstract:* NRC Forms 540, 541, and 542, together with their continuation pages, designated by the ''A'' suffix, provide a set of standardized forms to meet Department of Transportation (DOT), NRC, and State requirements. The forms were developed by NRC at the request of low-level waste industry groups. The forms provide uniformity and efficiency in the collection of information contained in manifests which are required to control transfers of low-level radioactive waste intended for disposal at a land disposal facility. NRC Form 540 contains information needed to satisfy DOT shipping paper requirements in 49 CFR Part 172 and the waste tracking requirements of NRC in 10 CFR Part 20. NRC Form 541 contains information needed by disposal site facilities to safely dispose of low-level waste and information to meet NRC and State requirements regulating these activities. NRC Form 542, completed by waste collectors or processors, contains information which facilitates tracking the identity of the waste generator. That tracking becomes more complicated when the waste forms, dimensions, or packagings are changed by the waste processor. Each container of waste shipped from a waste processor may contain waste from several different generators. The information provided on NRC Form 542 permits the States and Compacts to know the original generators of low-level waste, as authorized by the Low-Level Radioactive Waste Policy Amendments Act of 1985, so they can ensure that waste is disposed of in the appropriate Compact.

A copy of the final supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O–1F23, Rockville, MD 20852. OMB clearance requests are available at the NRC worldwide web site: http://www.nrc.gov/NRC/PUBLIC/OMB/index.html). The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Comments and questions should be directed to the OMB reviewer by May 4, 2001: Amy Farrell, Office of Information and Regulatory Affairs (3150–0164, 0165, & 0166), NEOB–10202, Office of Management and Budget, Washington, DC 20503.

Comments can also be submitted by telephone at (202) 395–7318.

The NRC Clearance Officer is Brenda Jo. Shelton, (301) 415–7233.

Dated at Rockville, Maryland, this 28th day of March 2001.

For the Nuclear Regulatory Commission.

Brenda Jo. Shelton,

NRC Clearance Officer, Office of the Chief Information Officer.

[FR Doc. 01–8235 Filed 4–3–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment; Notice of Meeting

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment will hold a meeting on April 17, 2001, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Tuesday, April 17, 2001–8:30 a.m. Until the Conclusion of Business

The Subcommittee will discuss the results of the staff's Phase 1 development of risk-based performance indicators for reactors, and related matters. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Mr. Michael T. Markley (telephone 301/415–6885) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: March 29, 2001.

James E. Lyons,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 01–8234 Filed 4–3–01; 8:45 am]

BILLING CODE 7590–01–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 March 12 through March 23, 2001. The last biweekly notice was published on March 21, 2001 (66 FR 15915).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity For a Hearing

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

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

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

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of 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 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 May 4, 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: Rulemaking and Adjudications 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 and 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: March 1, 2001.

Description of amendment request: The proposed amendment would increase the reactor core isolation cooling system surveillance test upper pressure limit from 1020 psig to 1045 psig.

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

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

The Reactor Core Isolation Cooling (RCIC) System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. The RCIC System is also designed to provide core cooling for a wide range of reactor pressures, from 150 pounds per square inch gage (psig) to 1200 psig. The proposed change to the Technical Specifications (TS) Section 3.5.3, ''RCIC System,'' Surveillance Requirement (SR) 3.5.3.3 to allow the RCIC system high pressure test to be performed at a higher reactor pressure (i.e., less than or equal to 1045 psig) is consistent with the current design and licensing basis for the RCIC system. The change to the upper pressure limit for the conduct of this SR will not adversely impact the performance characteristics of any structure, system, or component that is assumed to initiate a previously evaluated accident. Therefore, the proposed change will not result in an increase in the probability of an accident previously evaluated.

The proposed change to the TS SR will not result in reduced performance or effectiveness of the reactor coolant pressure boundary and therefore will not have an adverse impact on any barriers. As such, the RCIC System will still be capable of performing its transient and accident mitigation function as assumed in the accident analysis. On this basis, the consequences of any accident previously evaluated are not affected by the proposed change.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences on any accident previously evaluated.

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

(2) The proposed change to the TS SR to allow the RCIC System high pressure test to be performed at a higher reactor pressure (i.e., less than or equal to 1045 psig) is consistent with the current design and licensing basis for the RCIC system. The proposed change will not change the method for performing the test and the revised test pressure is within the current operating design basis of the plant. Since the proposed test pressure is within the design basis for the reactor and the RCIC System, performing the SR at the new pressure will not prevent the RCIC System from performing its required function or result in a failure of the reactor coolant pressure boundary. As such, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed change will not involve a significant reduction in the margin of safety.

The RCIC System is designed to operate either automatically or manually following RPV isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. The RCIC System is also designed to provide core cooling for a wide range of reactor pressures, from 150 psig to 1200 psig. The proposed change to TS SR 3.5.3.3 to allow the RCIC System high pressure test to be performed at a higher pressure (i.e., less than or equal to 1045 psig) is consistent with the current design and licensing basis for the RCIC system. Since the test at the higher reactor pressure will continue to provide reasonable assurance that the RCIC System will perform its intended safety function when called upon during an accident or transient, the proposed change will 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: 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, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: December 29, 2000.

Description of amendment request: The proposed amendment would revise the offsite power sources identified in Technical Specification (TS) 3.7.A.3 to remove one listed source and add a different source. In addition, the bases would be revised to reflect the availability of the offsite sources and also be revised administratively for minor changes.

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

The proposed amendment does not:

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

The proposed change to the Technical Specifications involves the removal of the 230 kV line from TS 3.7.A.3.a and addition of the 69 kV Sands Point line in its place. This ensures that two active offsite power sources are connected to the plant to support plant operation. Two 230 kV lines (considered one active power source), 34.5 kV line Q121 and the 69 kV Sands Point line will be normally maintained as active sources. Utility system operators will connect the Z52 line under certain grid conditions, which provides a backup to the normally available sources and improves offsite power reliability.

Since the number of normally available active offsite power sources is maintained at three, the probability of occurrence of a loss of offsite power is not adversely impacted and, therefore, the proposed change does not significantly increase the probability of occurrence of an accident previously evaluated.

Voltage analyses and voltage regulation studies have reviewed various degraded grid and plant operating scenarios. Degraded grid studies have considered single contingency events, such as loss of transmission lines or transformers, and various plant outages, minimum Technical Specification conditions and local system blackouts. Voltage regulation studies have considered various plant operating modes such as normal operation, accident motor starting and loading and shutdown conditions. The studies have concluded that adequate voltage will be available to safety-related electrical loads under all of the plant operating modes. Therefore, the consequences of any accident will not change since the operation of safety-related systems are not affected by the change in offsite power sources. For a complete loss of offsite power, the standby diesel generators are relied upon to provide electrical power to safety systems. Since the proposed change to Technical Specification 3.7.A.3 does not affect the operability of the standby diesel generators, the consequences of a loss of offsite power are unchanged. Therefore, there is no significant increase in the consequences of an accident previously evaluated.

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

The proposed change to Technical Specification 3.7.A.3 adds a new offsite power source (69 kV Sands Point line) and removes the second 230 kV line as an additional source since it is not separate from the other 230 kV line. Therefore, the proposed change involves the availability of offsite power connections to the plant. A potential loss of offsite power is already evaluated and the standby diesel generators are relied upon to provide power to accident mitigation and safe shutdown equipment in the event all offsite power is lost. Therefore, the proposed change to available offsite power sources does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Technical Specification 3.7.A.3 requires two offsite power lines to be fully operational for plant start-up and operation. The proposed change to the Technical Specifications involves the removal of the 230 kV line from TS 3.7.A.3.a and addition of the 69 kV Sands Point line in its place. This ensures that two active offsite power sources are connected to the plant to support plant operation. Two 230 kV lines (considered one active power source), 34.5 kV line Q121 and the 69 kV Sands Point line will be normally maintained as active sources. The removal of the second 230 kV line from TS 3.7.A.3.a has no impact on available active sources since the [sic] both 230 kV lines are routed on the same towers and are therefore considered as a single active source. Analyses have concluded that sufficient capacity and capability of offsite power is available when TS 3.7.A.2 and 3.7.A.3 are met. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: February 27, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3, ''Engineered Safety Features,'' and TS 3.7, ''Auxiliary Electrical Systems,'' to change the mode applicability for certain systems from the point of time when the reactor is made critical to when the average reactor coolant temperature is heated above 350 °F. The amendment would also change the associated action that must be taken when the TS conditions cannot be met to require a plant cooldown to below 350 °F. In addition, the associated TS statements that incorrectly refer to ''power operation'' and ''normal reactor operation'' for these TSs are proposed to be corrected. The proposed amendment would also revise the applicable TS Bases sections and make some minor formatting and editorial changes.

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

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

The proposed changes consist of revisions to the TS requirements for certain safeguards equipment and associated auxiliary electrical equipment to reflect the requirements of the steam line break analyses. The result of these changes will be that these safeguards systems will be required to be operable for additional plant conditions (with average reactor coolant temperature above 350 °F and the reactor not critical). These operability requirements for the safeguards equipment meet the assumptions utilized in the IP2 [Indian Point Unit 2] safety analyses and, therefore, will not result in a change in the consequences of the accident analyses.

Additionally, the affected safeguards equipment is not an initiator for any accident previously analyzed for IP2. The proposed changes do not result in a change to the design or operation of the safeguards equipment but extends the plant conditions under which this equipment will be required to be operable.

Therefore, there is no increase in the probability or in the consequences of an accident previously evaluated.

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

The proposed changes involve revising the TS applicability for certain safeguards equipment and associated auxiliary electrical systems to require this equipment to be operable with average reactor coolant temperature above 350 °F. The proposed changes do not involve a change to the design or operation of any plant system or equipment. The result of the proposed change is an increased range of operating conditions under which the safeguards equipment will be required to be operable. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes reflect the assumptions for safeguards equipment operability assumed in the steam line break accident analyses. These changes ensure that the affected TS reflect the assumptions of the safety analyses but do not result in a change to any of the safety analyses or any margin of safety.

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

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: January 30, 2001.

Description of amendment request: The proposed amendment would make two changes to reporting requirements in Facility Operating License DPR–20. First, the requirement in Section 2.C.(3)b that ''All changes in the approved [Fire Protection] program shall be reported annually, along with the FSAR [Final Safety Analysis Report] revision * * *'' would be changed to state ''All changes to the approved program shall be reported along with the FSAR revision as required by 10 CFR 50.71(e) * * *'' Secondly, a change would be made to Section 2.F, which currently states:

Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 50.73(b), (c), and (e).

The revised Section 2.F would state:

The licensee shall report any violations of Section 2.C(1) of this license within 24 hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 60 days in accordance with 10 CFR 50.73(b), (c), and (e).

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

The following evaluation supports the finding that operation of the facility in accordance with the proposed changes would not:

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

Two changes are proposed; both deal solely with clarification of reporting requirements contained in the Facility Operating License. Since these changes have no effect on the physical plant or its operation, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated. Therefore, operation of the facility in accordance with the proposed changes to the Facility Operating License would not involve a significant increase in the probability or consequences of an accident previously evaluated.

b. create the possibility of a new or different kind of accident from any previously evaluated.

Two changes are proposed; both deal solely with clarification of reporting requirements contained in the Facility Operating License. Since these changes have no effect on the physical plant or its operation, they cannot create the possibility of a new or different kind of accident from any previously evaluated.

Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications [sic, Facility Operating License] would not create the possibility of a new or different kind of accident from any previously evaluated.

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

Two changes are proposed; both deal solely with clarification of reporting requirements contained in the Facility Operating License. Since these changes have no effect on the physical plant or its operation, they cannot involve a significant reduction in a margin of safety.

Therefore, the proposed change to the Technical Specifications [sic, Facility Operating License] 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 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: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: February 20, 2001.

Description of amendment request: The licensee is proposing to add a new chapter to the Columbia Generating Station Physical Security Plan pertaining to the Independent Spent Fuel Storage Installation (ISFSI) security requirements. Specifically, the licensee proposes the following changes regarding the ISFSI: (1) Illumination will be sufficient to permit adequate assessment of unauthorized penetrations or activities within the protected area, (2) personnel access will be controlled by a key and lock system administered by the security force, (3) personnel identification will be by visual identification using plant access picture badges and an ISFSI authorization list, (4) no vehicle barrier around the perimeter of the ISFSI, (5) response time for valid alarms that only needs to be sufficient to assess the situation and the further need for corrective actions, and (6) secondary power supply for alarm annunciator equipment and non-portable communications equipment will have secondary power from an uninterruptible power supply not in the vital area.

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

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

The probability of an evaluated accident is derived from the probabilities of individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of facility systems and the ability of plant personnel to mitigate those consequences.

Confinement of all radioactive materials at the Energy Northwest ISFSI is provided by the required use of certified spent fuel storage casks in accordance with 10 CFR 72.214. The design objective of NRC certified spent fuel storage casks is to provide a confinement boundary that ensures there are no credible design basis events resulting in unacceptable radiological releases to the environment. In addition, these spent fuel storage casks are to be located within the confines of the Energy Northwest ISFSI which is designed as a protected area.

Since the design objective of the spent fuel storage cask has not been altered, there is no increase in individual precursors of an accident and the probability of an evaluated accident is not increased. The spent fuel casks stored at the Energy Northwest ISFSI will be inside a new fenced protected area with access requirements, detection aids, alarm devices, communication requirements, and observational capabilities commensurate with the activity of passive dry cask spent fuel storage that meet or exceed the criteria specified under 10 CFR 73.51. Since the Energy Northwest ISFSI physical security program will provide a high degree of assurance that activities involving spent nuclear fuel do not constitute an unreasonable risk, the consequences of an accident previously evaluated are not expected to increase.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of facility configuration, including changes in allowable modes of operation or the potential for new or different personnel errors.

The proposed license amendment does not alter the design objective of NRC certified spent fuel storage casks. This license amendment request does not involve any modifications of the spent fuel storage casks or allowable modes of operation and no potential exists for the creation of personnel errors that might be new accident precursors. Thus, no new precursors of an accident are created and there is not a possibility of a new or different kind of accident.

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

Confinement of all radioactive materials and substantial physical protection of the spent nuclear fuel is accomplished by the required use of an NRC certified spent fuel storage cask as provided by Certificate of Compliance listed under 10 CFR 72.214. The spent fuel casks stored in the Energy

Northwest ISFSI will be inside a new protected area with access requirements, detection aids, alarm devices, communication requirements, and observational capabilities that meet or exceed the criteria specified in 10 CFR 73.51 for spent fuel stored under a specific license.

Since the Energy Northwest ISFSI will provide a high degree of assurance that activities involving spent nuclear fuel do not constitute an unreasonable risk, there is not a significant reduction in the margin of safety.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005–3502.

NRC Section Chief: Stephen Dembek.

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: March 12, 2001.

Description of amendment request: The proposed amendment would modify Technical Specification 3.1.3.4a to reduce the minimum requirement for average reactor coolant temperature during the rod cluster control assembly (RCCA) drop test from greater than or equal to 541°F to greater than or equal to 500°F. RCCA drop tests are required prior to reactor criticality: (1) For all rods, following each removal of the reactor vessel head, (2) for specifically affected individual rods, following maintenance work which could affect the drop times of those specific rods, and (3) at least every 18 months.

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

(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 probability of occurrence of an accident previously evaluated for Turkey Point is not altered by the proposed amendments to the Technical Specifications. The proposed changes do not impact the integrity of the reactor coolant system pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore do not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not make any physical changes to the facility design, material, or construction standards. The probability of any design basis accident (DBA) is not affected by these changes, nor are the consequences of any DBA affected by these changes. The proposed changes are not considered to be an initiator or contributor to any accident currently evaluated in the Turkey Point Updated Final Safety Analysis Report (UFSAR). Based on the above, Florida Power and Light Company concludes that 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 Rod Cluster Control Assembly (RCCA) drop test is routinely performed each refueling. Decreasing the test temperature will not create the possibility of a new or different accident. The proposed test conditions remain bounded by the analysis of record since the RCCA drop time assumption in the UFSAR accident analysis will not be changed. Since no new failure modes are associated with the proposed changes, the proposed amendments 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.

These Technical Specification changes do not involve a significant reduction in margin since the acceptance criterion for RCCA drop time will not change. The proposed changes will reduce the minimum RCCA rod drop test temperature from greater than or equal to 541°F to greater than or equal to 500°F. This will slightly increase the test drop time, but will be well within the current Technical Specifications limit of 2.4 seconds. Therefore, the margin to safety as defined by Technical Specifications acceptance criterion is not impacted by the proposed amendments.

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 Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of amendment request: March 7, 2001.

Description of amendment request: The licensee requests allowing a one-time interval extension for the Crystal River Unit 3 (CR–3) Type A, Integrated Leakage Rate Test (ILRT) for no more than 6 years.

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

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

The proposed revision to the CR–3 [Improved Technical Specifications] ITS adds a one-time extension to the current interval for Type A testing. The current test interval of 10 years, would be extended on a one-time basis to 16 years from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A testing extension is not a modification to plant systems, nor a change to plant operation that could initiate an accident. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493 found that, generically, very few potential containment leakage paths fail to be identified by Type B and C tests. In fact, an analysis of 144 ILRT results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A (ILRT) testing frequency to one per twenty years would lead to an imperceptible increase in risk. CR–3 provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. Inspections required by the Maintenance Rule and American Society of Mechanical Engineers (ASME) code are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing required by the CR–3 ITS will identify any containment opening, such as valves, that would otherwise be detected by the Type A tests. These factors show that a CR–3 Type A test extension will not represent a significant increase in the consequences of an accident.

2. Does not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant. There are no changes to the operation of the plant that could introduce a new failure mode creating the possibility of a new or different kind of accident.

3. Does not involve a significant reduction in the margin of safety.

The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20 year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes a very small amount to the individual risk, and that the decrease in Type A testing frequency would have a minimal affect on this risk since most potential leakage paths are detected by Type C testing.

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: R. Alexander Glenn, Associate General Counsel (MAC–BT15A), Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Richard P. Correia.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: March 6, 2001.

Description of amendment request: The licensee proposed to amend the unit's Technical Specifications (TSs), Section 3.4.4, ''Emergency Ventilation System [EVS],'' and Section 3.4.5, ''Control Room Air Treatment [CRAT] System,'' to require testing consistent with American Society for Testing and Materials (ASTM) Standard D3803–1989 (currently the American National Standards Institute (ANSI) standard N510–1980 is specified). Concurrently, the licensee proposed to change the charcoal bed testing efficiency of the EVS and CRAT from 90 percent to 95 percent, and requiring the pressure drop across the CRAT System high efficiency particulate air (HEPA) filters and charcoal adsorber banks to be demonstrated to be less than 1.5 inches of water. The licensee's application for amendment is a response to the NRC's Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The associated licensee-controlled TS Bases document would also be changed to reflect these TS changes.

The staff had previously published notices (65 FR 9009, February 23, 2000, and 65 FR 56955, September 20, 2000) for the licensee's November 30, 1999, and August 15, 2000, submittals. The licensee's March 6, 2001, submittal supersedes the original submittals in their entirety. Hence this notice also supersedes the previous two notices.

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 reviewed the licensee's analysis against the three standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed TS changes will require that the charcoal filter beds be tested in accordance with an NRC-approved standard (i.e., ASTM D3803–1989), and to improved acceptance criteria. The CRAT and EVS do not involve initiators or precursors to an accident previously evaluated, as these systems perform only mitigative functions in response to an accident. Failure of these systems would result in inability or decreased ability to perform their mitigative functions, but would not increase the probability of an accident. The proposed testing requirements would improve the performance of these systems, and would not have any effect in reducing their design functions. Therefore, the probability and consequences of an accident previously evaluated will not be increased by the proposed TS changes.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed TS change will only revise the testing requirements. These changes will not involve placing the systems in new configurations or operating the systems in different manners. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since no design, operation procedure, or analysis methodology is changed, proposed TS changes will not adversely affect the performance characteristics of the CRAT or EVS, nor will they affect the ability of the systems to perform their intended functions. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment requests: June 8, 2000, as supplemented by letter dated January 4, 2001.

Description of amendment requests: The proposed license amendments would revise Section 3.5.5, ''Emergency Core Cooling Systems—Seal Injection Flow,'' of the improved Technical Specifications to replace the description of the seal injection flow with a description consistent with the method used to establish and verify reactor coolant pump seal injection flow limits and the method used to calculate the seal injection flow in the safety analyses for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2.

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

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

The emergency core cooling system (ECCS) analyses model the reactor coolant pump (RCP) seal injection flow path as a hydraulic flow resistance. This proposed change clarifies that RCP seal flow is a function of system conditions rather than specifying an actual flow rate. The seal flow rate can vary during operation, but the hydraulic flow resistance is fixed by positioning the manual seal injection throttle valves. The resistance does not change if the valve adjustments are not changed. Thus, RCP seal flow variation due to changing reactor coolant system (RCS) back pressure following a loss of coolant accident (LOCA) is explicitly determined as a result of modeling the RCP seal injection flow path resistance.

The proposed improved Technical Specification change is only a clarification and does not impact the way the RCP seal flow is established and thus cannot affect RCP seal integrity. The seal flow resistance otherwise only affects ECCS flow. Since ECCS flow occurs after an accident the proposed change cannot impact the probability of an accident.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change continues to ensure that the assumed ECCS flow is available. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. Since the change continues to ensure that the assumed ECCS flow is available, no new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. Since the change continues to ensure the assumed ECCS flow is available, there will be no impact 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(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: February 20, 2001. This application supersedes the June 19, 2000, application and supplement dated September 12, 2000 (published in the **Federal Register** on October 4, 2000 [65 FR59223]).

Description of amendment requests: The proposed license amendments would revise Sections 5.5.9, ''Steam Generator (SG) Tube Surveillance Program'' and 5.6.10, ''SG Tube Inspection Report,'' of the Diablo Canyon Power Plant, Unit Nos. 1 and 2 Technical Specifications (TS), to add new surveillance and reporting requirements associated with SG tube inspection and repair. The new requirements establish alternate repair criteria for axial primary water stress corrosion cracking at dented tube support plate intersections.

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

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

Examination of crack morphology for primary water stress corrosion cracking (PWSCC) at dented intersections has been found to show one or two microcracks well aligned with only a few uncorroded ligaments and little or no other inside diameter axial cracking at the intersection. This relatively simple morphology is conducive to obtaining good accuracy in nondestructive examination (NDE) sizing of these indications. Accordingly, alternate repair criteria (ARC) are established based on crack length and average and maximum depth within the thickness of the tube support plate (TSP).

The application of the ARC requires a Monte Carlo condition monitoring assessment to determine the as-found condition of the tubing. The condition monitoring analysis described in WCAP–15573, Revision 0, is consistent with NRC Generic Letter 95–05 requirements.

The application of the ARC requires a Monte Carlo operational assessment to determine the need for tube repair. The repair bases are obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile using Monte Carlo analysis techniques described in WCAP–15573, Revision 0. The burst pressure and leakage are compared to the requirements in WCAP–15573, Revision 0. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected end of cycle (EOC) requirements are satisfied, the tube will be left in service.

A steam generator (SG) tube rupture event is one of a number of design basis accidents that are analyzed as part of a plant's licensing basis. A single or multiple tube rupture event would not be expected in a SG in which the ARC has been applied. The ARC requires repair of any indication having a maximum crack depth greater than or equal to 40 percent outside the TSP, thus limiting the potential length of a deep crack outside the TSP at EOC conditions and providing margin against burst and leakage for free span indications.

For other design basis accidents such as a main steam line break, main feed line break, control rod ejection, and locked reactor coolant pump motor, the tubes are assumed to retain their structural integrity.

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

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

Implementation of the proposed SG tube ARC does not introduce any significant changes to the plant design basis. A single or multiple tube rupture event would not be expected in a SG in which the ARC has been applied. Both condition monitoring and operational assessments are completed as part of the implementation of ARC to determine that structural and leakage margin exists prior to returning SGs to service following inspections. If the condition monitoring requirements are not satisfied for burst or leakage, the causal factors for EOC indications exceeding the expected values will be evaluated. The methodology and application of this ARC will continue to ensure that tube integrity is maintained during all plant conditions consistent with the requirements of Regulatory Guide (RG) 1.121 and Revision 1 of RG 1.83.

In the analysis of a SG tube rupture event, a bounding primary-to-secondary leakage rate equal to the operational leakage limits in the Technical Specifications (TS), plus the leak rate associated with the double-ended rupture of a single tube, is assumed. For other design basis accidents, the tubes are assumed to retain their structural integrity and exhibit primary-to-secondary leakage within the limits assumed in the current licensing basis accident analyses. Steam line break leakage rates from the proposed PWSCC ARC are combined with leakage rates from other approved ARC (i.e., voltage-based ARC and W* ARC). The combined leakage rates will not exceed the limits assumed in the current licensing basis accident analyses.

The 40 percent maximum depth repair limit for free span indications provides a very low likelihood of free span leakage under design basis or severe accident conditions. Leakage from indications inside the TSP is limited by the constraint of the TSP even under severe accident conditions, and leakage behavior in a severe accident would be similar to that found acceptable by the NRC under approved ARC for axial outside diameter stress corrosion cracking (ODSCC) at TSP intersections. Therefore, even under severe accident conditions, it is concluded that application of the proposed ARC for PWSCC at dented TSP locations results in a negligible difference in risk of a tube rupture or large leakage event, when compared to current 40 percent repair limits or previously approved ARC.

Diablo Canyon Power Plant (DCPP) continues to implement a maximum operating condition leak rate limit of 150 gallons per day per SG to preclude the potential for excessive leakage during all plant conditions.

The possibility of a new or different kind of accident from any previously evaluated is not created because SG tube integrity is maintained by inservice inspection, condition monitoring, operational assessment, tube repair, and primary-to-secondary leakage monitoring.

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

Tube repair limits provide reasonable assurance that tubes accepted for continued service without repair will exhibit adequate tube structural and leakage integrity during subsequent plant operation. The implementation of the proposed ARC is demonstrated to maintain SG tube integrity consistent with the criteria of draft NRC Regulatory Guide 1.121. The guidelines of RG 1.121 describe a method acceptable to the NRC staff for meeting General Design Criteria

(GDC) 2, 4, 14, 15, 31, and 32 by ensuring the probability or the consequences of SG tube rupture remain within acceptable limits. This is accomplished by determining the limiting conditions of degradation of SG tubing, for which tubes with unacceptable cracking should be removed from service.

Upon implementation of the proposed ARC, even under the worst-case conditions, the occurrence of PWSCC at the tube support plate elevations is not expected to lead to a SG tube rupture event during normal or faulted plant conditions. The ARC involves a computational assessment to be completed for each indication left in service ensuring that performance criteria for tube integrity and leak tightness are met until the next scheduled outage.

As discussed below, certain tubes are excluded from application of ARC. Existing tube integrity requirements apply to these tubes, and the margin of safety is not reduced.

In addressing the combined loading effects of a loss-of-coolant (LOCA) and safe shutdown earthquake (SSE) on the SGs (as required by GDC 2), the potential exists for yielding of the TSP in the vicinity of the wedge groups, accompanied by deformation of tubes and a subsequent postulated in-leakage. Tube deformation could lead to opening of pre-existing tight through wall cracks, resulting in secondary to primary in-leakage following the event, which could have an adverse affect on the Final Safety Analysis Report (FSAR) results. Based on a DCPP analysis of LOCA and SSE, SG tubes located in wedge region exclusion zones are susceptible to deformation, and are excluded from application of ARC.

A DCPP tube stress analysis for feed line break (FLB)/steam line break (SLB) plus SSE loading determined that high bending stresses occur in certain SG tubes at the seventh TSP, because the stresses exceed the maximum imposed bending stress for existing test data (equal to approximately the lower tolerance limit yield stress). These tubes are located in rows 11 to 15 and 36 to 46, and are excluded from application of ARC.

Tube intersections that contain TSP ligament cracking are also excluded from application of ARC.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to the plant safety analyses as defined in the FSAR or TS.

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

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

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

NRC Section Chief: Stephen Dembek.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: March 6, 2001.

Description of amendment request: The proposed amendment revises Section 5.0, ''Administrative Controls,'' of the Trojan Nuclear Plant (TNP or Trojan) Technical Specifications. The first change is associated with modification of the TNP organizational structure. Specifically, the position of Senior Vice President, Power Supply, will be eliminated and the position Trojan Site Executive and Plant General Manager will be divided into two separate positions: (1) Trojan Site Executive, and (2) General Manager, Trojan. The second change is associated with revising language used in the TNP Technical Specifications to conform with the language of the revised 10 CFR 50.59. Phrases which included the wording ''unreviewed safety question'' and ''safety evaluation'' will be replaced with wording that will continue to conform to the requirements of the revised 10 CFR 50.59.

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

As described above [in the licensee's amendment request], the changes in management titles and reporting relationships are administrative in nature, and as concluded by the NRC in the discussion accompanying the final rule, the changes made for consistency with the new 10 CFR 50.59 are viewed as editorial in nature. As such, these proposed changes do not alter the intent of the Possession Only License, and do not modify the present plant systems or administrative controls necessary to preserve and protect the integrity of the nuclear fuel at the TNP. Since no plant systems or administrative controls are changed, the probability or consequences of accidents previously evaluated are unaffected. The General Manager, Trojan will be located at the site and will provide management attention to each of the functional areas in the TNP organization during decommissioning of the facility.

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

As described above [in the licensee's amendment request], the changes in management titles and reporting relationships are administrative in nature, and as concluded by the NRC in the discussion accompanying the final rule, the changes made for consistency with the new 10 CFR 50.59 are viewed as editorial in nature. As such, these changes do not affect the manner in which systems and components are operated or maintained, and do not alter the intent of the Possession Only License. There are no new accident scenarios or failure modes created by the requested administrative/editorial changes. Therefore, the requested changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

As described above [in the licensee's amendment request], the changes in management titles and reporting relationships are administrative in nature, and as concluded by the NRC in the discussion accompanying the final rule, the changes made for consistency with the new 10 CFR 50.59 are viewed as editorial in nature. As such, these changes do not affect the manner in which systems and components are operated or maintained, do not alter the intent of the Possession Only License, and do not adversely impact previously accepted margins of safety. Therefore, the requested amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Douglas R. Nichols, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204.

NRC Section Chief: Robert A. Gramm.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendments: March 9, 2001 (TS 01–01).

Brief description of amendments: The proposed amendment would change the Sequoyah Nuclear Plant (SQN) Technical Specification section on reactor core design (Section 5.3) by adding a provision for including a limited number of lead test assemblies in the core.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), Tennessee Valley Authority, 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 lead test assemblies (LTAs) are identical to the other Mark-BW fuel assemblies with the exception of the initial uranium isotopic composition change. This composition change does not effect the chemical properties or affect the thermal-hydraulic performance of the fuel. The change in composition does change the neutronic response of the fuel. However the operational behavior of the fuel is accurately predicted by the NRC approved methodologies used for reload core design and analysis as demonstrated in the Topical Report [Framatome Cogema Fuels Topical Report BAW–2328], and the successful operation during SQN Unit 2 Cycle 10. Therefore, the LTAs do not significantly increase the probability of accidents while in the reactor.

A preliminary reload design analysis performed, based upon the tentative use of the LTAs in SQN Unit 1 operating Cycle 12 fuel load pattern, shows that the LTAs will not become the most limiting fuel assemblies in the core during the cycle. Additionally, the peak pin criteria will be analyzed for each reload pattern to ensure that the LTAs do not become the most limiting peak pin at any time during their residence in the core.

The potential effects of the LTAs on plant operation and safety are evaluated for each reload core design. The key core safety analysis parameters are examined each cycle to ensure each parameter remains bounded by the more limiting values used in the safety analysis of record and that there is no increase in the probability of occurrence for any design basis accident described in the Final Safety Analysis Report (FSAR).

The impacts of the LTAs on the radiological consequences for all postulated events have been evaluated. The total calculated source term and the source-term activity of isotopes, which significantly contribute to operator and off-site accident exposure levels, were shown to be less than standard fuel assemblies with the same burnup, therefore, it will not increase the consequences of any accident previously evaluated.

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

The fuel assembly design for the LTAs is identical to the standard fuel assemblies. The main difference between the LTAs and the production fuel is that the initial concentration of the U^{234} and U^{236} isotopes will be higher in the LTA fuel pellets than that typically found in standard fuel. These isotopic differences will not affect the chemical, mechanical, or thermal properties of the fuel pellet.

The LTAs meet the same design criteria and licensing basis criteria as the standard fuel assemblies and were manufactured with the same processes. The LTA skeleton is identical to the standard skeleton, which ensures that the loadings associated with normal operation, seismic events, loss-of-coolant accident (LOCA) events, and shipping and handling are not affected.

Pressure and temperature safety limits will be maintained the same as those for the current operating cycle, thus ensuring that the fuel will be maintained within the same range of safety parameters that form the basis for previous accident evaluations. No new performance requirements are being imposed on any system or component that exceed design criteria or cause the core to operate in excess of design basis operating limits. No credible scenario has been identified, which could jeopardize equipment that could cause or intensify an accident sequence or mitigate events. Therefore, the LTAs will not create the possibility of accidents or equipment malfunctions of a different type than previously evaluated while in the reactor.

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

The LTAs will not adversely affect reactor neutronic or thermal-hydraulic performance. The LOCA acceptance criteria with LTAs installed in the core will continue to be met. The acceptance criteria for departure from nucleate boiling (DNB) events with the LTAs installed in the core will also continue to be met. Other acceptance criteria have also been demonstrated to remain within acceptable limits. The total calculated source-term activity and the source-term activity of isotopes, which significantly contribute to operator and off-site accident exposure levels of the LTAs, was determined to be less than that for the standard fuel assembly with the same burnup. All previously evaluated events remain bounding and valid. For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

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.

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

Date of amendment request: March 2, 2001.

Description of amendment request: The proposed amendment would revise Watts Bar Nuclear Plant (WBN) Unit 1 Technical Specifications (TS) Section 5.6, ''TS Bases Control Program,'' to adopt NRC-approved Technical Specification Task Force (TSTF) item TSTF–364, Revision 0. TSTF–364 revises the Industry Standard TS consistent with the recent revision to 10 CFR 50.59.

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

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

The proposed change is an administrative modification of existing TS requirements for the TS Bases Control Program to reference changes pursuant to 10 CFR 50.59 rather than ''unreviewed safety question.'' This change has no affect on the current review and approval process for changes to the Final Safety Analyses Report [FSAR] and Bases. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As such, there is no effect on initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed amendment does 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.

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

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

The proposed change is an administrative modification of existing TS requirements for the TS Bases Control Program to reference changes pursuant to 10 CFR 50.59 rather than ''unreviewed safety question''. This change has no affect on the current review process for changes to the FSAR and Bases, and will not reduce a margin of safety because it has no effect on any safety analyses assumptions. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. For these reasons, the proposed amendment does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of application request: February 16, 2001 (ULNRC–04390).

Description of amendment request: The amendment would add the word ''Senior'' to the title ''Vice President and Chief Nuclear Officer'' in paragraph c to Technical Specification 5.2.1, ''Onsite and Offsite Organizations.'' The new title would be ''Senior Vice President and Chief Nuclear Officer.''

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

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

The proposed change revises Callaway Plant management organization by changing the title, Vice President and Chief Nuclear Officer to Senior Vice President and Chief Nuclear Officer; creating Vice President-Nuclear, to add another corporate level of oversight for plant site activities and nuclear staff supervision; and centralizing the Operations, Operations Support, and Engineering functions under the Vice President-Nuclear. These are administrative changes. [The proposed change does not change any plant safety limit, plant operations, or the plant design related to any accident previously evaluated.]

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

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

The proposed change revises Callaway Plant management organization by changing the title, Vice President and Chief Nuclear Officer to Senior Vice President and Chief Nuclear Officer; creating the title Vice President-Nuclear, to add another corporate level of oversight for plant site activities and nuclear staff supervision; and centralizing the Operations, Operations Support, and Engineering functions under the Vice President-Nuclear. These are administrative changes. [The proposed change does not involve an initiator of an accident.]

Therefore, the proposed revision will not create a new or different kind of accident from any accident previously evaluated.

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

The proposed change revises Callaway Plant management organization by changing the title, Vice President and Chief Nuclear Officer to Senior Vice President and Chief Nuclear Officer; creating the title Vice President-Nuclear, to add another corporate level of oversight for plant site activities and nuclear staff supervision; and centralizing the Operations, Operations Support, and Engineering functions under the Vice President-Nuclear. These are administrative changes.

Therefore, the proposed change to the Technical Specifications 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: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

Yankee Atomic Electric Co., Docket No. 50–29, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts

Date of amendment request: November 22, 2000.

Description of amendment request: The requested amendment would relocate certain administrative requirements from the YNPS Defueled Technical Specifications to the YNPS Decommissioning Quality Assurance Program (YDQAP). Additional editorial changes to titles and designations are also proposed.

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

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

The administrative nature of the changes will not affect any important to safety systems or components or their mode of operation. Relocation of TS administrative Sections 6.5, 6.7 and 6.9 to the YDQAP does not result in changes to either system design or operating strategies. Relocation of these administrative requirements to the YDQAP has no affect on accident initiators or mitigation. Therefore, the proposed administrative changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not modify plant operation, systems, or components. Relocation of TS administrative Sections 6.5, 6.7 and 6.9 to the YDQAP does not affect any of the parameters or conditions that could contribute to the initiation of any accident. No new accident scenarios are created as a result of relocating the aforementioned administrative requirements to the YDQAP. In addition, no important to safety equipment or functions are altered as a result of this proposed change. Therefore, the proposed administrative changes will not create the possibility of a new or different accident from any previously evaluated.

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

The changes are administrative in nature involving the relocation of administrative requirements from one licensing document to another licensing document currently containing related requirements. Relocation of TS administrative Sections 6.5, 6.7 and 6.9 to the YDQAP does not affect plant operation, systems, or components. The proposed administrative changes do not represent a change in initial conditions, system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. Therefore, the proposed administrative changes will 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: Thomas Dignan, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Section Chief: Stephen Dembek.

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

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

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

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

Date of amendment request: April 14, 2000, as supplemented by letters dated June 2, July 28, and December 1, 2000, and January 31, 2001.

Brief description of amendment request: The proposed amendment would change the surveillance requirements for laboratory testing of the charcoal adsorbers for the control room, the spent fuel pool storage area and the safety injection pump rooms. In addition, the amendment would delete the laboratory testing requirements for the containment charcoal adsorbers. The changes comply with the guidance of Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of publication of individual notice in ***Federal Register:*** March 5, 2001 (66 FR 13355).

Expiration date of individual notice: April 4, 2001.

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 and 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. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: August 9, 2000, as supplemented February 22, 2001. The February 22, 2001, supplement provided additional clarifying information and did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice.

Brief description of amendment: The amendment approved a revision to the Updated Final Safety Analysis Report (UFSAR) to reflect a revised steam generator tube failure accident analysis which includes the dose resulting from the postulated post-accident steam release through the main steam safety valves. The existing radiological dose calculations described in the UFSAR do not account for this release.

Date of issuance: March 9, 2001.

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

Amendment No.: 230.

Facility Operating License No. DPR–50. Amendment authorized UFSAR revision.

Date of initial notice in ***Federal Register:*** October 18, 2000 (65 FR 62382).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 9, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 8, 2000, as supplemented by the letters of January 3 and March 13, 2001.

Brief description of amendments: The amendments revise Technical Specification (TS) 5.6.5, ''Core Operating Limits Report,'' to add a methodology using the CASMO–4 and SIMULATE–3 Codes to the list of analytical methods used to determine core operating limits contained in TS 5.6.5.b. The amendments allow the use of the CASMO–4 and SIMULATE–3 methodology to perform nuclear design calculations; however, as stated in the supplemental letter of January 3, 2001, the licensee agreed that the introduction of significantly different or new fuel designs will require further validation of the physics methods in CASMO–4/SIMULATE–3 for application to Palo Verde Units 1, 2, and 3, and will require review by the NRC staff.

Date of issuance: March 20, 2001.

Effective date: March 20, 2001,and shall be implemented within 45 days of the date of issuance, including putting the condition mentioned above on the use of the new methodology, that was given in the licensee's letter of March 13, 2001, in the Updated Final Safety Analysis Report for Palo Verde.

Amendment Nos.: Unit 1–132, Unit 2–132, Unit 3–132.

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

Date of initial notice in ***Federal Register:*** October 4, 2000 (65 FR 59219).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 20, 2001.

No significant hazards consideration comments received: No.

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 application for amendments: January 27, 2000, as supplemented on June 15, 2000, and November 21, 2000.

Brief description of amendments: The amendments revise Technical Specifications 3.9.3 and 3.9.4 by modifying the conditions of containment closure during core alterations, fuel handling and the loss of shutdown cooling. The amendments also revise the way the personnel air lock and the containment purge system are operated during maintenance activities on the shutdown cooling system.

Date of issuance: March 12, 2001.

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

Amendment Nos.: 242 and 216.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 8, 2000 (65 FR 12288).

The June 15, 2000, and November 21, 2000, submittals provided clarifying information that did not change the original proposed no significant hazards consideration determination.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 12, 2001.

No significant hazards consideration comments received: No.

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 application for amendments: September 14, 2000.

Brief description of amendments: The amendments revise the reactor coolant heatup and cooldown curves in the Technical Specifications.

Date of issuance: March 15, 2001.

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

Amendment Nos.: 243 and 217.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 18, 2000 (65 FR 62382).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 15, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 10, 2000.

Brief description of amendment: This amendment revises Required Actions suspending operations involving reactivity additions and revises various Limiting Condition for Operation Notes precluding reduction in boron concentration.

Date of issuance: March 14, 2001.

Effective date: March 14, 2001.

Amendment No. 190.

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

*Date of initial notice in **Federal Register:*** September 6, 2000 (65 FR 54084).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 8, 2000.

Brief description of amendment: The amendment deletes Technical Specification Section 5.5.3, ''Post Accident Sampling Program,'' for Palisades and thereby eliminates the requirements to have and maintain the post-accident sampling system for the plant.

Date of issuance: March 7, 2001.

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

Amendment No.: 193.

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

*Date of initial notice in **Federal Register:*** January 24, 2001 (66 FR 7679).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 28, 2000.

Brief description of amendment: The amendment changes the Arkansas Nuclear One, Unit 1 technical specifications to revise the safety-related 4160 Volt (V) bus loss-of-voltage and 480 V bus degraded voltage relay allowable values.

Date of issuance: March 12, 2001.

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

Amendment No.: 211.

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

*Date of initial notice in **Federal Register:*** December 13, 2000 (65 FR 77918).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 21, 2000, as supplemented by letters dated June 29, September 1, October 26, and December 22, 2000, and February 22, 2001.

Brief description of amendment: The amendment provides for a full-scope implementation of the alternative source term, as described in NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,'' and 10 CFR 50.67, ''Accident source term.''

Date of issuance: March 14, 2001.

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

Amendment No: 145.

Facility Operating License No. NPF–29: The amendment revises the Facility Operating License and Technical Specifications.

*Date of initial notice in **Federal Register:*** March 22, 2000 (65 FR 15380).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: February 29, 2000, as supplemented by letter dated January 11, 2001.

Brief description of amendments: The amendments reduced the number of safety valves required for overpressure protection at Dresden, Unit 2, by removing from Technical Specifications (TS) Section 3.6.E, the safety valve function and setpoint of the Target Rock safety/relief valve (SRV). The amendments also moved the remaining safety valve lift pressure setpoints from TS Section 3.6.E to TS Section 4.6.E, changed the number of required safety valves from nine to eight, and removed footnote ''c'' of Unit 3 TS Section 4.6.E.

Date of issuance: March 23, 2001.

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

Amendment Nos.: 184 and 179.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 21, 2001 (66 FR 11055).

The January 11, 2001, letter is 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 March 23, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 21, 2000, as supplemented by letters dated December 1, and December 13, 2000, and January 29, 2001.

Brief description of amendment: This amendment approves revisions to the Main Steam Line Break (MSLB) design-basis accident dose consequence analysis as documented in the Updated Final Safety Analysis Report (UFSAR) and a technical specification (TS) change. The changes to the MSLB accident dose consequence analysis include revisions to input parameter values and assumptions. The TS change reduces the limit on reactor coolant system specific activity in technical specification 3/4.4.8. The revisions are in accordance with the methodology described in Nuclear Regulatory Commission Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator tubes by Outside Diameter Stress Corrosion Cracking.''

Date of issuance: March 12, 2001.

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

Amendment No: 236.

Facility Operating License No. DPR–66: Amendment revised the Technical Specifications and approved changes to the UFSAR.

Date of initial notice in ***Federal Register:*** February 7, 2001 (66 FR 9382).

Information from the July 21, and December 13, 2000, letters was used for the staff's initial proposal to determine that the amendment request involves a no significant hazards consideration determination. The December 1, 2000, and January 29, 2001, letters provided supplemental information applicable to this amendment request but did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the original notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2001.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of application for amendment: December 11, 2000, as supplemented by letter dated February 15, 2001.

Brief description of amendment: This amendment revised the existing Minimum Critical Power Ratio (MCPR) Safety Limit contained in Technical Specification 2.1.1.2 by increasing the limit for two recirculation loop operation from 1.09 to 1.10.

Date of issuance: March 12, 2001.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment No.: 119.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2013). The supplemental letter contained clarifying information that was within the scope of the original **Federal Register** notice and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 28, 2000, as supplemented January 17, 2001, and February 15, 2001.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to permit, as an alternative to the current dedicated Shift Technical Advisor (STA), a single, qualified individual to simultaneously serve as an STA and a Senior Reactor Operator, and either option would be permitted on a shift-by-shift basis.

Date of Issuance: March 14, 2001.

Effective Date: March 14, 2001.

Amendment Nos.: 113 and 173.

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

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81922). The letters dated January 17, 2001, and February 15, 2001, contained clarifying information that did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 14, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 15, 1999, as supplemented by letter dated November 14, 2000.

Brief description of amendment: The amendment authorized revision of the Updated Safety Analysis Report (USAR) to allow the use of the service water system to directly supply cooling water to the reactor equipment cooling system during a loss-of-coolant accident event.

Date of issuance: March 13, 2001.

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

Amendment No.: 185.

Facility Operating License No. DPR–46: Amendment authorized revision to the USAR.

Date of initial notice in ***Federal Register:*** July 14, 1999 (64 FR 38030). The November 14, 2000, supplemental letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 20, 2000.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to (1) include the automatic reactor water cleanup (RWCU) system isolation feature, (2) restore the dose equivalent iodine-131 limit to 2 microcuries per gram, (3) change the RWCU reactor water level automatic isolation signal from Low to Low-Low reactor water level and add TSs for the high pressure coolant injection (HPCI) and reactor core isolation cooling low steam line pressure isolation instrumentation, (4) delete the HPCI 150,000 lb/hr low range high flow isolation instrumentation and adds a time delay to the 300,000 lb/hr upper range high flow isolation instrumentation, and (5) change the suppression chamber water allowable water level from volume units to level units.

Date of issuance: March 7, 2001.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 117.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51361).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 10, 2001.

Brief description of amendment: The amendment removes the standby liquid control (SLC) pump flow surveillance requirement to recycle demineralized water to the test tank and changes the testing frequency of the SLC pump capacity test from monthly to quarterly.

Date of issuance: March 8, 2001.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 118.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 7, 2001 (66 FR 9386).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 5, 2000, as supplemented by letters dated September 28, 2000, December 1, 2000, and December 11, 2000.

Brief description of amendment: The amendment revised Sections 1.1, 1.3, 2.10, 3.10, and 5.9 and associated Bases of the Fort Calhoun Station, Unit No. 1 (FCS) technical specifications. The amendment allows use of NRC-approved Siemens Power Corporation (SPC) methodologies for determining reactor core operating limits in conjunction with use of SPC fabricated nuclear fuel. Additionally, the revised SPC fuel assembly growth model for FCS Cycle 20 core reload was reviewed and approved.

Date of issuance: March 14, 2001

Effective date: March 14, 2001, to be implemented within 30 days from the date of issuance.

Amendment No.: 196.

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

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81925).

The September 28, December 1 and 11, 2000, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2001.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: July 31, 2000.

Brief description of amendments: The amendments revised the main steam isolation valve leakage rate surveillance requirements.

Date of issuance: March 9, 2001.

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

Amendment Nos.: 190 and 165.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 18, 2000 (65 FR 62390).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 9, 2001.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: November 16, 2000.

Brief description of amendments: The amendments eliminated response time testing requirements for certain reactor protection system and isolation actuation system instrumentation.

Date of issuance: March 12, 2001.

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

Amendment Nos.: 191 and 166.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2022).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 12, 2001.

No significant hazards consideration comments received: No.

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 application for amendments: October 6, 2000.

Description of amendment request: The amendments revise the Technical Specifications (TS) to specify required actions and completion times applicable to conditions when two low-pressure coolant injection pumps, each in a different subsystem, are inoperable.

Date of issuance: March 12, 2001.

Effective date: March 12, 2001.

Amendment Nos: 240, 269, 229.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68. Amendments revise the TS..

Date of initial notice in the ***Federal Register:*** November 15, 2000 (65 FR 69066) and re-noticed February 7, 2001 (66 FR 9387).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2001.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–260, Browns Ferry Nuclear Plant, Unit 2, Limestone County, Alabama

Date of application for amendment: November 21, 2000 as supplemented by a February 9, 2001 reply to a request for additional information.

Brief description of amendment: It revises the minimum critical power ratio safety limits specified in the facility Technical Specifications (TS) for two-loop and single-loop operation.

Date of issuance: March 13, 2001.

Effective date: March 13, 2001.

Amendment No.: 270

Facility Operating License No. DPR–52: Amendment revises the TS.

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77927). The letter dated February 9, 2001, contained clarifying information that did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 22, 2000

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.14, ''Technical Specifications (TS) Bases Control Program'' to reflect the changes made to 10 CFR 50.59 as published in the **Federal Register** on October 4, 1999 (Volume 64, Number 191, ''Changes, Tests, and Experiments,'' pages 53582 through 53617). A conforming change is made to TS 5.5.14 to replace the word ''involves'' with the word ''requires,'' as it applies to changes to the TS Bases without prior NRC approval.

Date of issuance: March 15, 2001

Effective date: March 15, 2001, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 142.

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

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81931). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 15, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: June 22, 2000, as supplemented November 15, 2000.

Brief description of amendment: These amendments revise Technical Specification (TS) 3.1.2.7, TS 3.1.2.8, TS 3.5.1, TS 3.5.5, TS 3.6.2.2, and TS 3.9.1 to increase the boron concentration limits in the refueling water storage tank, casing cooling tank, safety injection accumulators, and the reactor coolant system during refueling.

Date of issuance: March 20, 2001.

Effective date: As of the date of issuance and shall be implemented at the end of the Fall 2001 refueling outage for Unit 1, and at the end of the Fall 2002 refueling outage for Unit 2.

Amendment Nos.: 225 and 206

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

Date of initial notice in ***Federal Register:*** July 26, 2000 (65 FR 46018). The November 15, 2000, supplement contained clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 20, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: December 19, 2000.

Brief Description of amendments: These amendments revise Table 3.7–4, item 7, and Technical Specification 3.6.B. The changes revise the range of allowable values for the 4160-volt bus loss-of-voltage and degraded voltage relay settings.

Date of issuance: March 12, 2001.

Effective date: March 12, 2001.

Amendment Nos.: 224 and 224.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2025).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 12, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland this 27th day of March 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

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SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–44106; File No. 4–429]

Joint Industry Plan; Notice of Filing of Amendment to the Options Intermarket Linkage Plan To Conform the Options Intermarket Linkage Plan to the Requirements of Securities Exchange Act Rule 11Ac1–7

Pursuant to section 11A(a)(3) of the Securities Exchange Act of 1934 (''Act'')[1] and Rule 11Aa3–2, thereunder,[2] notice is hereby given that on March 13, 2001, the American Stock Exchange LLC (''Amex''), Chicago Board Options Exchange, Inc. (''CBOE''), International Securities Exchange LLC (''ISE''), Pacific Exchange, Inc. (''PCX''), and Philadelphia Stock Exchange, Inc. (''Phlx'') (collectively the ''Participants'') submitted to the Securities and Exchange Commission (''SEC'' or ''Commission'') an amendment to the Options Intermarket Linkage Plan.[3] The amendment proposes to conform the Linkage Plan to the requirements of the recently-adopted Exchange Act Rule 11Ac1-7, the Trade-Through Disclosure Rule.[4] The Commission is publishing this notice to solicit comments from interested persons on the proposed Linkage Plan amendment.

I. Description and Purpose of the Amendment

On November 17, 2000, the Commission adopted Rule 11Ac1-7 to require a broker-dealer to disclose to its customer when the customer's order for listed options is executed at a price inferior to a better published quote (''intermarket trade-through''), and to disclose the better published quote available at that time. However, a broker-dealer is not required to disclose to its customer an intermarket trade-through if the broker-dealer effects the transaction on an exchange that participates in an approved linkage plan that includes provisions reasonably designed to limit customers' orders from being executed at prices that trade through a better published quote. The purpose of the proposed amendment to the Linkage Plan is to add provisions to the Linkage Plan that are reasonably designed to limit intermarket trade-throughs.

The proposed amendment would change the definitions of ''National Best Bid or Offer'' (''NBBO'') and ''Trade-Throughs'' so that the terms would apply to unlinked, as well as linked, exchanges. The Participants represent that the proposed changes would extend the requirement in the Linkage Plan that, absent reasonable justification and during normal market conditions, members should not effect trade-throughs, to unlinked markets, as well as linked markets.

Next, the proposed amendment would require that Participants establish procedures for conducting surveillance for trade-throughs, both with respect to trading through linked and unlinked markets. It also would require that Participants adopt uniform rules that

[1] 15 U.S.C. 78k-1(a)(3).

[2] 17 CFR 240,11Aa3-2.

[3] On July 28, 2000, the Commission approved a national market system plan for the purpose of creating and operating an intermarket options market linkage (''Linkage Plan'') proposed by the Amex, CBOE, and ISE. *See* Securities Exchange Act Release No. 43086 (July 28, 2000), 65 FR 48023 (August 4, 2000). Subsequently, upon request by the Phlx and PCX, the Commission issued orders to permit these exchanges to participate in the Linkage Plan. *See* Securities Exchange Act Release Nos. 43573 (November 16, 2000), 65 FR 70850 (November 28, 2000) and 43574 (November 16, 2000), 65 FR 70851 (November 28, 2000).

[4] 17 CFR 240.11Ac1-7. *See* Securities Exchange Act Release No. 43591 (November 17, 2000), 65 FR 75439 (December 1, 2000) (''Adopting Release''). Specifically, in the Adopting Release, the Commission noted that to conform to the regulations of the Trade Through Disclosure Rule, a linkage plan must, at a minimum: (1) Limit participants from trading through, not only the quotes of other linkage plan participants, but also, the quotes of exchanges that are *not* participants in an approved linkage plan; (2) require plan participants to actively surveil their markets for trades executed at prices inferior to those publicly quoted on other exchanges; and (3) make clear that the failure of a market with a better quote to complain within a specified period of time that its quote was traded-through may affect potential liability, but does not signify that a trade-through has not occurred.