(DEIS) for Yucca Mountain (tentative) (Open)—The Committee will receive an information briefing from the NRC staff on their plans to review the DOE DEIS for the proposed HLW repository at Yucca Mountain, Nevada.

I. *2:00–3 p.m.: Break and Preparation of Draft ACNW Reports* (Open)—Cognizant ACNW members will prepare draft reports, as needed, for consideration by the full Committee.

J. *3:00–5 p.m.: Discussion of Proposed ACNW Reports* (Open)—The Committee will continue its discussion of proposed ACNW reports.

Thursday, May 17, 2001

K. *8:30–8:35 a.m.: Opening Remarks by the ACNW Chairman* (Open)—The ACNW Chairman will make opening remarks regarding the conduct of the meeting.

L. *8:35–10 a.m.: Meeting Reports* (Open)—The Committee will hear reports from the members and staff on meetings attended since the 125th ACNW Meeting, including the National Research Council Meeting on their report on long-term institutional control, the 9th International HLW Conference and the Nuclear Waste Technical Review Board Spring Meeting.

M. *10:15–12 Noon: Discussion of Proposed ACNW Reports* (Open)—The Committee will continue its discussion of proposed ACNW reports.

N. *1:00–1:30 p.m.: Miscellaneous* (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACNW meetings were published in the **Federal Register** on October 11, 2000 (65 FR 60475). In accordance with these procedures, oral or written statements may be presented by members of the public, 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 Committee, its consultants, and staff. Persons desiring to make oral statements should notify Howard J. Larson, ACNW, as far in advance as practicable so that appropriate arrangements can be made to schedule the necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting will be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for taking pictures may be obtained by contacting the ACNW office, prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should notify Mr. Larson as to their particular needs.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore can be obtained by contacting Mr. Howard J. Larson, ACNW (Telephone 301/415–6805), between 8 A.M. and 5 P.M. EDT.

ACNW meeting notices, meeting transcripts, and letter reports are now available for downloading or viewing on the internet at http://www.nrc.gov/ACRSACNW.

Videoteleconferencing service is available for observing open sessions of ACNW meetings. Those wishing to use this service for observing ACNW meetings should contact Mr. Theron Brown, ACNW Audiovisual Technician (301/415–8066), between 7:30 a.m. and 3:45 p.m. EDT at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: April 26, 2001.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 01–10964 Filed 5–1–01; 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 April 9, 2001, through April 20, 2001. The last biweekly notice was published on April 18, 2001 (66 FR 19998).

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 June 1, 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).

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

Date of amendments request: April 1, 2001 (102–04552).

Description of amendments request: The amendments would revise the requirements on the following programs in the administrative controls section of the technical specifications (TSs): (1) Section 5.5.13, ''Diesel Fuel Oil Testing Program,'' (2) Section 5.5.14, ''TS Bases Control Program,'' (3) Section 5.5.15, ''Safety Functions Determination

Program (SFDP),'' and (4) Section 5.6.5, ''Core Operating Limits Report (COLR).'' The proposed changes clarify the program requirements in Section 5.5.13 without changing testing methods or limits, revise the program in Section 5.5.14 based on changes to 10 CFR 50.59 in the regulations, clarify the program requirements in Section 5.5.15 including changing the program name to the plant-specific name for the program, and add the CENTS code to the list of analytical methods used, including the use of CENTS for control element assembly ejection analyses, to determine core operating limits and revise the list of referenced topical reports in the COLR in Section 5.6.5.

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.

Technical Specification (TS) 5.5.13, Diesel Generator Fuel Oil Program. TS 5.5.13.a.3 currently states, ''Water and sediment are within the limits of ASTM D1796,'' for the acceptability of new diesel fuel oil. This is an incorrect reference for the limits of water and sediment of new fuel oil. The water and sediment limits for new fuel oil are contained within the Technical Specification Bases. ASTM D1796 contains testing methods used for analysis of new fuel oil for water and sediment. This proposed amendment changes the wording of TS 5.5.13.a.3 to state, ''Water and sediment within limits when tested in accordance with ASTM D1796.'' This proposed change is an administrative change and will have no affect on plant design, operation, or maintenance. Additionally, this proposed change does not result in any hardware changes or affect plant operating practices. The water and sediment testing methods and limits are not affected by this change. Thus, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.5.14, TS Bases Control Program, requires a program for processing changes to the Bases of the TS [...]

In the initial sentence to TS 5.5.14.b, the word ''involve'' will be replaced with ''require.'' Additionally, the second allowance for changing TS Bases as described in TS 5.5.14.b will be revised to state, ''A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.'' This change is based on the changes to 10 CFR 50.59 published in the **Federal Register** (Volume 64, Number 191) dated October 4, 1999. This change is consistent with NRC approved Technical Specifications Task Force (TSTF) traveler number 364-revision 0.

This change will also numerically format the two options listed in TS 5.5.14.b. This is consistent with other listings contained in Section 5.0 of the TS.

This proposed change deletes the reference to ''unreviewed safety question'' as previously used in 10 CFR 50.59[, before the rule change published October 4, 1999, in the **Federal Register**.] Deletion of this definition was approved by the NRC with the revision to 10 CFR 50.59.

[These] proposed change[s to TS 5.5.14 are] administrative change[s] and will have no affect on plant design, operation, or maintenance. Additionally, [these] change[s do] not result in any hardware changes or affect plant operating practices. Therefore, [the] proposed change[s to TS 5.5.14 do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.5.15, Safety Functions Determination Program (SFDP). Clarification is being added to TS 5.5.15. The second paragraph of TS 5.5.15 will be changed to read: ''A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and * * *''

An additional paragraph will be added to the end of TS 5.5.15 stating, ''When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.''

Additionally, clarification will be added to limiting conditions for operation (LCO) 3.0.6 Bases of the ''appropriate LCO for loss of safety function.'' The Bases will also clarify the requirement for the SFDP that consideration does not have to be made for a loss of power in determining loss of function. This change is consistent with NRC approved TSTF traveler number 273-revision 2, as amended by editorial change WOG–ED–23.

In addition, an editorial change to remove the ''*s*'' from the word ''Function*s*'' in the title for TS 5.5.15 will occur. The change reflects the plant specific name for this program.

[These] proposed change[s to TS Section 5.5.15 are] administrative change[s] and will have no affect on plant design, operation, or maintenance. The change[s] clarif[y] the requirements for determining loss of safety function and the correct LCO to enter for loss of safety function. The proposed change[s do] not result in any hardware changes or affect plant operating practices. The program will still determine when a safety function has been lost and will direct the appropriate action. Therefore, [the] proposed change[s to TS 5.5.15 do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.6.5, Core Operating Limits Report (COLR) is being revised to add the option to use the CENTS computer code in licensing analysis by adding CENTS to the list of approved core operating limit analytical methods contained in TS 5.6.5.b. The CENTS computer code has been generally approved for the calculation of transient behavior in Pressurized Water reactors (PWRs) designed by Combustion Engineering (CE). PVNGS intends to qualify CENTS for use in future Palo Verde licensing analyses by following the guidelines prescribed in Generic Letter (GL) 83–11, Supplement 1.

CENTS is a best-estimate code designed to provide realistic simulation of Nuclear Steam Supply System (NSSS) behavior during normal and transient conditions. The CENTS Safety Evaluation (SE) documents the generic NRC approval of the CENTS code for use in the licensing analyses for PWRs designed by CE. The CENTS SE is described in letter, ''Acceptance for Referencing of Licensing Topical Report CE–NPD 282–P, ''Technical Manual for the CENTS Code'' dated March 17, 1994, from USNRC to S. A. Toelle, ABB Combustion Engineering.

The proposed change does not immediately alter any methodology used in [an] reload analysis. It only provides the option to replace the CESEC transient simulation code with an alternate NRC approved code. Providing the option to substitute the NRC approved CESEC code with another NRC approved code (CENTS) will not alter the physical characteristics of any component involved in the initiation or mitigation of an accident. The actual implementation of the CENTS code will be performed by following the guidance provided in Generic Letter (GL) 83–11, Supplement 1. This proposed change does not result in any hardware changes or affect plant operating practices. Thus, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.6.5, core operating limits report (COLR) which identifies the methodology report(s) by number, title, date, and NRC staff approval document, will be revised to allow the reports to be identified by number and title only. A note will be added to TS 5.6.5.b to specify that a complete citation be included in the COLR for each report, including the report number, title, revision, date, and any supplements.

This change has previously been reviewed and accepted by the NRC in letter, ''Acceptance for Siemens References to Approved Topical Reports in Technical Specifications'' from S.A. Richards, NRC to J.F. Mallay, Siemens Power Corporation dated December 15, 1999. This change is also consistent with NRC accepted TSTF 363-revision 0.

Additionally, TS 5.6.5.b.6 and 5.6.5.b.7 both list the same topical report (Calculative Methods for the CE Small Break LOCA Evaluation Model, CENPD–137). TS 5.6.5.b.7 is the supplement to the topical report listed in [TS] 5.6.5.b.6. TS 5.6.5.b.7 will be deleted and the ''Calculative Methods for the CE Small Break LOCA Evaluation Model, CENPD–137'' topical report (along with its supplement) will be listed in full text within the COLR.

[The] proposed change[s related to the listing of topical reports in TS 5.6.5.b are] administrative change[s] and will have no affect on plant design, operation, or maintenance. Thus, [these] proposed change[s 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.

TS 5.5.13, Diesel Generator Fuel Oil Program. The proposed change is an administrative change. This change would have no affect on the physical plant. Consequently, plant configuration and the operational characteristics remain unchanged and the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.5.14, TS Bases Control Program. The proposed changes associated with TS 5.5.14.b do not involve any physical changes. These changes allow PVNGS to be in compliance with NRC approved changes to 10 CFR 50.59. This change is an administrative change. Plant configuration and the operational characteristics remain unchanged and thus, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.5.15, SFDP. The proposed change to TS 5.5.15 does not involve any physical changes to the plant[s]. This change is an administrative change. The loss of function of the specific component is addressed in its specific TS LCO and plant configuration will be governed by the required actions of those LCOs. Since this proposed change is a clarification that does not degrade the availability or capability of safety related equipment, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.6.5, COLR is being revised to add the option to use the CENTS computer code in licensing analysis by adding CENTS to the list of approved core operating limit analytical methods contained in TS 5.6.5.b. The proposed change will not affect reload analysis other than providing an option to replace the CESEC transient simulation code with an equivalent code. Providing this option in and of itself will not alter the physical characteristics of any component in the plant. Since providing the option to use the CENTS code will not alter the physical characteristics of any component in the plant, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.6.5, Core Operating Limits Report (COLR) which identifies the methodology report(s) by number, title, date, and NRC staff approval document, will be revised to allow the reports to be identified by number and title only. This is an administrative change. This change has no affect on the physical plant. Plant configuration and the operational characteristics remain unchanged and thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

TS 5.5.13, Diesel Generator Fuel Oil Program. The proposed change to TS 5.5.13.a.3 is an administrative change. This change would have no affect on the physical plant and has no effect on any safety analyses assumptions. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.5.14, TS Bases Control Program. The proposed changes associated with TS 5.5.14.b will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph (c)(2) of 10 CFR 50.59 will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature and is based on NRC reviewed and approved changes to 10 CFR 50.59. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.5.15, SFDP. The proposed change to TS 5.5.15 are clarifications only. No changes are made in the LCO, the time required for the TS required actions to be completed, or the out of service time for the components involved. The NRC has approved the proposed administrative changes (TSTF 273-revision 2, as amended by editorial change WOG–ED–23). Safety-related equipment controlled by the TS will still perform as credited in the safety analysis. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.6.5, COLR is being revised to add the option to use the CENTS computer code in licensing analysis by adding CENTS to the list of approved core operating limit analytical methods. The proposed change will allow running existing analyses with a different method that has been reviewed and approved by NRC. The actual implementation of the CENTS code will be performed by following the guidance provided in Generic Letter (GL) 83–11, Supplement 1. Thus, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.6.5, Core Operating Limits Report (COLR) which identifies the methodology report(s) by number, title, date, and NRC staff approval document, will be revised to allow the reports to be identified by number and title only. This is an administrative change. This change has no affect on the physical plant. Plant configuration and the operational characteristics remain unchanged. Therefore, change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: April 4, 2001 (102–04554).

Description of amendments request: The amendments would revise Specification 3.3.12, ''Boron Dilution Alarm System (BDAS),'' and Specification 3.9.2, ''Refueling Operations—Nuclear Instrumentation'' of the technical specification (TSs). Specification 3.9.2 applies to the required operability of startup range monitors (SRMs). The applicability modes for limiting condition for operation (LCO) 3.3.12 would be extended to Mode 6, refueling. A note to ''Enter applicable Conditions and Required Actions of LCO 3.3.12, ''Boron Dilution Alarm System (BDAS),'' for BDAS made inoperable by SRMs'' would be added to the Actions for LCO 3.9.2 and the Required Action B.2 on performing surveillance requirement (SR) 3.9.1.1, and associated completion time, for LCO 3.9.2 would be deleted.

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

No. The proposed changes to Technical Specifications 3.3.12 and 3.9.2 do not significantly increase the probability or consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR) [for Palo Verde Nuclear Generating Station]. The proposed amendment[s] would add MODE 6 Applicability to TS 3.3.12 for the BDAS. In addition, the proposed amendment[s] would add a note to the Actions of TS 3.9.2 which directs the operator to enter the applicable Conditions and Required Actions of TS 3.3.12 in the event that the BDAS is made inoperable by inoperable startup range monitors (SRMs). Finally, the proposed amendment[s] would delete the TS 3.9.2 Required Action B.2.

The boron dilution alarm system (BDAS) and chemical monitoring of the reactor coolant system (RCS) boron concentration are established in the MODE 6 inadvertent deboration analysis in UFSAR Section 15.4.6 to alert the operator of a boron dilution event at least 30 minutes prior to a loss of subcriticality. The BDAS and RCS boron monitoring are not accident initiators. The proposed changes will ensure that the assumptions of UFSAR Section 15.4.6, for mitigating an inadvertent deboration event, are met. In addition, the proposed changes do not alter the design or configuration of the plant but establish requirements for operating the plant as analyzed and designed. The amendment[s do] not physically affect the operability or availability of the boron dilution alarm system (BDAS), but ensures it is available as required or that sufficient actions are taken if it becomes inoperable. Furthermore, the inadvertent deboration event analysis does not involve dose consequences since the acceptance criteria is to provide operator notification at least 30 minutes prior to the loss of subcriticality such that the operator may terminate the event before subcriticality is achieved [and exceeded,] and the RCS and fuel clad boundaries are challenged. Therefore, the proposed amendment[s] to TS 3.3.12 and TS 3.9.2 [do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment[s] to Technical Specifications 3.3.12 and 3.9.2 [do] not create the possibility of an accident of a new or different kind from any accident previously evaluated. The proposed amendment[s] would add MODE 6 Applicability to TS 3.3.12 for the BDAS. In addition, the proposed amendment[s] would add a note to the Actions of TS 3.9.2 which directs the operator to enter the applicable Conditions and Required Actions of TS 3.3.12 in the event that the BDAS is made inoperable by inoperable startup range monitors (SRMs). Finally, the proposed amendment[s] would delete the TS 3.9.2 Required Action B.2. The proposed changes do not alter the design or configuration of the plant but establish requirements for operating the plant as analyzed and designed.

In MODE 6, the BDAS and the startup range monitors (SRM) are the primary means to monitor reactivity changes during core alterations and to alert the operator of a boron dilution event in time to prevent a loss of subcriticality. Chemical sampling to monitor RCS boron concentration is used when the BDAS is unavailable. Accidents involving reactivity anomalies are evaluated in UFSAR Section 15.4, Reactivity and Power Distribution Anomalies. Inadvertent deboration is described in UFSAR Section 15.4.6 as requiring the BDAS or chemical monitoring of the RCS boron concentration to alert the operator at least 30 minutes prior to the loss of subcriticality in MODE 6. The proposed changes to TS 3.3.12 and 3.9.2 will require the BDAS to be OPERABLE in MODE 6 or perform RCS boron concentration monitoring if the BDAS is inoperable.

The BDAS and RCS boron concentration monitoring are means to detect a boron dilution event. The proposed changes ensure this detection occurs as required. The proposed amendment[s do] not physically affect the response or operation of the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed changes to Technical Specifications 3.3.12 and 3.9.2 do not involve a significant reduction in a margin of safety. The proposed amendment[s] would add MODE 6 Applicability to TS 3.3.12 for the BDAS. In addition, the proposed amendment[s] would add a note to the Actions of TS 3.9.2 which directs the operator to enter the applicable Conditions and Required Actions of TS 3.3.12 in the event that the BDAS is made inoperable by inoperable startup range monitors (SRMs). Finally, the proposed amendment[s] would delete the TS 3.9.2 Required Action B.2. These changes ensure the adequate detection of a boron dilution event.

The current Technical Specifications 3.3.12 satisfies the inadvertent deboration safety analysis requirements to have the BDAS OPERABLE in MODES 3, 4,and 5. In accordance with UFSAR Section 15.4.6, Inadvertent Deboration, the same requirements and actions apply for MODE 6. Therefore, it is proposed that MODE 6 Applicability for the BDAS be added to TS 3.3.12. In addition, the Action section of TS 3.9.2 would be modified with a note to ensure the safety analysis assumptions are satisfied in MODE 6, since the SRM must be OPERABLE for the corresponding BDAS channel to be OPERABLE. Technical Specification Bases 3.3.12 and UFSAR Section 15.4.6 indicate that the BDAS is necessary to alert the operator of an inadvertent deboration event at least 15 minutes before the reactor loses subcriticality in MODES 3, 4, and 5. UFSAR Section 15.4.6 also indicates that 30 minutes is required in MODE 6. These criteria are in agreement with the guidance of NUREG 0800, [NRC's] Standard Review Plan. Therefore, the margin of safety being considered for [these] proposed amendment[s] is the 30 minutes before the loss of subcriticality that the operator must be notified [...] in the event of a boron dilution event. The proposed changes to TS 3.3.12 and TS 3.9.2 will require the BDAS to be OPERABLE in MODE 6 and, if the BDAS is inoperable, will require that the RSC boron concentration be monitored at pre-analyzed frequencies via chemical sampling in order to satisfy the 30 minute acceptance criteria. Finally, the proposed change[s] also serve to clarify that an inoperable SRM will cause the corresponding BDAS channel to be inoperable, thus requiring action in accordance with TS 3.3.12, in addition to TS 3.9.2. [The proposed changes add a requirement to TS 3.3.12 and account for the BDAS being inoperable because of inoperable SRMS] Therefore, the proposed change[s do] not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

Consumers Energy Company, Docket No. 50–155, Big Rock Point Plant, Charlevoix, County, Michigan

Date of amendment request: October 26, 2000, as supplemented by letters dated February 9, February 28, March 14, March 15, and March 23, 2001.

Description of amendment request: The proposed amendment reflects the replacement of the original 75-ton reactor building gantry crane by an upgraded single-failure proof 125-ton crane designed to meet Crane Manufacturers Association of America (CMAA) Specification 70 and American Society of Mechanical Engineers (ASME) B30.2. The proposed amendment to the Technical Specifications (TSs) would revise (1) Definition 1.8 on fuel handling, (2) the applicability of TS 3/4.2.1 on fuel handling support system requirements, and (3) Section 3.2.2.d of the limiting conditions for operation for TS 3/4.2.2 on fuel handling general requirements, and would delete TS 3/4.3.1 on control of heavy loads. The licensee also submitted revisions to the bases for TSs 3/4.2.2 and 3/4.3.1. The crane has a Design Rated Load (DRL) of 125 tons; however, it has been analyzed to safely retain a load of 105 tons under the site-specific earthquake and the Maximum Critical Load (MCL) for the crane is 105 tons.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided its analysis in its letters dated October 26, 2000, and March 14, 2001, which address the issue of no significant hazards consideration, and is presented below:

The proposed [amendment] does not:

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

A significant increase in the probability of an accident is not created because:

- The replacement crane will not be utilized for a greater number of fuel handling evolutions than was the case for the existing 75-ton crane. The existing crane was utilized for each transfer of fuel assemblies between the reactor and the Spent Fuel Pool; in the case of full-core offloads, which was the normal practice during refueling outages at Big Rock Point [Plant], the existing crane would make 84 transfers of irradiated fuel from the reactor to the Spent Fuel Pool, and a nominal 62 transfers of irradiated fuel from the Spent Fuel Pool back to the reactor. The replacement crane will handle fuel only after it has been placed into the W100 Transfer Cask. It is anticipated that the W100 Fuel Transfer Cask will be handled 14 times while it contains fuel (one movement from the

Spent Fuel Pool to a staging area in Room 444, and one movement from Room 444 to a W150 Storage Cask), during loading of seven W150 Storage Casks. Additional moves of the W100 Transfer Cask when it is loaded with fuel would be required only if an off-normal condition required a loaded cask to be returned to the Spent Fuel Pool.

• The replacement crane has been analyzed to safely handle the 105-ton W100 Fuel Transfer Cask under seismic conditions that include the Big Rock Point [Plant] site-specific safe shutdown earthquake of 0.104g. The UFHSR [Updated Final Hazards Summary Report] is being revised to limit the weight of loads being moved over the Spent Fuel Pool to 105 tons.

• The existing crane has been used to lift the properly rigged 24-ton fuel transfer cask over fuel; the probability of dropping the 24-ton fuel transfer cask was minimized by the proper rigging that consisted of attaching a safety catch device to the transfer cask. In the case of the replacement crane, loads will be prevented from dropping by the design of the single-failure proof Ederer X–SAM hoist, which prevents loads from dropping more than 18 inches in the event of any single failure. Administrative controls will be instituted on the use of the replacement crane to require lifts of any heavy loads over fuel or over structures, the failure of which would jeopardize safe storage of fuel, to be done at a height of greater than 18 inches. Administrative controls will be instituted to prohibit use of the replacement crane for movement of any cask over fuel; these controls will be specified in the Big Rock Point [Plant] UFHSR. Administrative controls that apply to our [the licensee's] current 75-ton crane will be maintained, and strengthened, as appropriate, to provide greater assurance that heavy loads transported over fuel will be safely transported. Strengthened administrative controls include limiting the number of crane operators to approximately 12 individuals, and requiring that they receive Operator Engineer training in the use of the upgraded crane.

• The existing crane met single-failure proof criteria only when it was used to handle the properly-rigged 24-ton fuel transfer cask; the 105-ton single-failure proof crane will be single-failure proof for all lifts of loads which are 105 tons or less.

• The replacement of the existing crane with a 105-ton single-failure proof crane is being performed as a safety-related modification, and 10 CFR [Part] 50 Appendix B [Quality Assurance] criteria are being applied to all critical elements of design, purchasing, installation and testing. Therefore, the replacement crane and trolley can be expected to perform in accordance with their design specifications. As a result, the probability of a trolley failure on the replacement crane is considered to be no greater than the probability of a failure of the safety catch device which was employed with the existing crane when it was used to handle the 24-ton fuel transfer cask.

A significant increase in the consequences of an accident is not created because:

• This change affects fuel handling, and fuel handling accidents have already been analyzed and bound all other categories of accidents at the Big Rock Point Plant. Analysis indicates that the dose from the bounding fuel accident (a 24-ton fuel transfer cask drop), assuming a free release path without isolation of ventilation from containment, falls below the Protective Action Guidelines (PAGs) of Environmental Protection Agency–400 (EPA–400) 68 days following plant shutdown (the reactor was shutdown [as] of 8/29/1997). The analysis assumed a total of 500 damaged assemblies in the Spent Fuel Pool, with 84 of them being freshly discharged from the reactor. The Spent Fuel Pool contains 441 fuel assemblies. With more than three years of radiological and heat decay since the plant was shutdown, the potential source terms for gaseous and volatile radionuclides associated with the remaining design basis accidents has continued to decrease; therefore, the doses at the site boundary associated with a postulated accident involving any number of the available fuel assemblies have also decreased. The design of the Ederer X–SAM trolley and hoist is such that upon a single failure of the trolley that would allow the suspended load to free-fall, the load could fall for a maximum of 18 inches before the drum brake mechanism would engage to stop the downward travel. An 18-inch drop of the 105-ton dry fuel storage system fuel transfer cask has been analyzed and has been determined not to result in failure of the floors of the Spent Fuel Pool, Room 444 or the laydown area at the 599-foot 5-inch elevation of containment. These are the only floors in containment over which the cask will [be] moved with the 105-ton single-failure proof crane at a height of less than 18 inches. The 105-ton W100 Transfer Cask is the largest load that will be handled over the Spent Fuel Pool, when fuel is being stored in the Pool. For other floors/structures, (*i.e.,* the Reactor Deck at elevation 632′ 6″ [632 feet 6 inches]), administrative controls will be imposed to require the 105-ton cask to be suspended at least 18 inches above the floor/structure.

[The proposed amendment reflects the replacement of the original 75-ton non single-failure proof crane, with a single-failure proof crane. The replacement crane addresses malfunctions (e.g., dropping loads under single-failure conditions) that were possible with the original crane.]

Based on this discussion, it is concluded that this proposed change to the Defueled Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is requested to reflect the removal of the original 75-ton reactor building non single-failure proof semi-gantry crane and its replacement with a single-failure proof 105-ton crane, which will be designed to meet the applicable criteria and guidelines of NUREG–0554 and NUREG–0612. The change results from installation of a crane that replaces another crane. The general functions performed by the replacement crane (cask handling and movement of heavy loads) do not differ from those performed by the original crane. Therefore, new or different accidents will not be created by elimination of restrictions associated with the original 75-ton crane, since the design of the replacement crane addresses malfunctions (for example, dropping loads under single-failure conditions) that were possible with the original crane.

Based on this discussion, it is concluded that this proposed change to the Defueled Technical Specifications does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

To prevent failure of the Spent Fuel Pool structure when handling loads over the Pool with the existing 75-ton crane, loads were limited to 24 tons, and cask handling evolutions were limited to the southwest corner of the Spent Fuel Pool. These measures ensured that the Spent Fuel pool would not fail as a result of a load being dropped into it. The replacement crane has been designed such that a load will not drop more than 18 inches if a single failure should occur in its trolley. A drop of the 105-ton W100 Fuel Transfer Cask from a height of 18 inches to the Spent Fuel Pool floor has been determined not to result in failure of the Spent Fuel Pool; loads handled by the replacement crane will be restricted to 105 tons to ensure that the structural integrity of the Spent Fuel Pool will not be compromised by a postulated drop of the 105-ton W100 Fuel Transfer Cask.

The existing crane is designed to handle loads up to 75 tons Because the existing crane was not designed as a single-failure proof crane, restrictions were placed on load paths, load weights, and the configuration of the 24-ton fuel transfer cask (the cask was required to have a safety catch device attached between the cask and the crane structure to prevent dropping the transfer cask in the event of a trolley failure) to ensure that a margin of safety existed with respect to dropping heavy loads on spent fuel and to prevent a dropped load from causing structural failure of the Spent Fuel Pool. The replacement crane is designed to withstand the Big Rock Point [Plant] site-specific safe shutdown earthquake of 0.104g while safely retaining a load equal to 105 tons. Therefore, handling the 105-ton W100 transfer cask with this crane provides equivalent margins with respect to crane failure as the current restriction that limits loads being handled over the Spent Fuel Pool to 24 tons. The UFHSR will restrict handling of loads over the Spent Fuel Pool to 105 tons whenever fuel is stored in the Pool. The trolley and hoist for the replacement crane are designed to be single-failure proof, and provide a margin of safety for dropping a suspended load equivalent to the safety catch employed with 24-ton transfer cask safety catch.

[The proposed amendment reflects the replacement of the original 75-ton non single-failure proof crane, with a single-failure proof 125-ton crane having an MCL of 105 tons.. The replacement crane is designed to meet the applicable criteria of NUREG–0554 and

NUREG–0612, CMAA Specification 70, and ASME B30.2.]

Based on this discussion, it is concluded that this proposed change to the Defueled Technical Specifications does not involve a significant decrease in the margin of safety.

The NRC staff has reviewed the licensee's analysis in both letters of October 26, 2000, and March 14, 2001, 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: David A. Mikelonis, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 5, 2001, as revised March 30, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.12, ''Technical Specifications (TS) Bases Control Program,'' to be consistent with the changes to 10 CFR 50.59 published in the **Federal Register** on October 4, 1999 (64 FR 53582), as reflected in the Nuclear Energy Institute's Technical Specification Task Force (TSTF) Standard TS Change Traveler, TSTF–364, ''Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59.'' Specifically, Palisades TS 5.5.12b currently states, in part, that licensees may make changes to Bases without prior NRC approval provided the changes do not ''involve * * * [a] change to the updated FSAR [Final Safety Analysis Report] or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.'' The proposed amendment would change this quoted portion of TS 5.5.12b to state ''require * * * [a] change to the updated FSAR or Bases that requires NRC approval pursuant 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:

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.

The proposed change deletes the reference to unreviewed safety question as defined in 10 CFR 50.59. Deletion of the definition of unreviewed safety question was approved by the NRC with the revision of 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does 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.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph 10 CFR 50.59(c)(2) will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the revision to 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: April 2, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) by removing all requirements for, and references to, the ''Assembly Radial Peaking Factor,'' (F_R^A). Consequently, in TS Section 1.0, the definition of Assembly Radial Peaking Factor would be deleted and the definition of the Total Radial Peaking Factor (F_R^T) would be corrected to read: ''F_R^T shall be the maximum ratio of the individual fuel pin power to the core average pin power integrated over the total core height, including tilt.'' In Limiting Condition for Operation (LCO) 3.2.2, the title would be changed to ''TOTAL RADIAL PEAKING FACTOR (F_R^T);'' the wording would state ''F_R^T shall be within the limits specified in the [Core Operating Limits Report] COLR;'' Condition A would state ''F_R^T not within limits specified in the COLR;'' Required Action A.1 would state ''Restore F_R^T to within limits;'' and Surveillance Requirement (SR) 3.2.2.1 would state ''Verify F_R^T is within limits specified in the COLR.'' In LCO 3.2.3, Required Action A.1 would state: Verify F_R^T is within the limits of LCO 3.2.2, ''Total Radial Peaking Factor (F_R^T)'.'' Associated changes would be made to the TS Bases and table of contents.

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

There are no changes in plant systems, plant control operating procedures or instrument alarm or trip settings associated with this [TS Change Request] TSCR. Because neither physical equipment, nor operating methods for that equipment change, the probability of accident initiation would not change. Therefore, the proposed technical specification change would not involve a significant increase in the probability of an accident previously evaluated.

The assembly radial peaking (F_R^A) has been used in the past safety analyses and radiological consequence analyses. These analyses utilized the assumption that F_R^A would remain within the Technical Specifications limit during plant operations. These analyses verify, for Anticipated Operational Occurrences (AOOs) and Postulated Accidents (PAs), that:

(1) The Departure from Nucleate Boiling Ratio (DNBR) remains above the appropriate Technical Specifications Safety Limit, and

(2) The calculated offsite doses and control room dose for the affected events remained within the guidelines of 10 CFR 100, Section 11, ''Determination of exclusion area, low population zone and population center distance,'' and 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, ''Control room.''

Improved DNB correlations and better spacer grid design have allowed the safety analysis calculations to be performed using only the total radial peaking factor (F_R^T) limit (which remains unchanged), without exceeding the specified Safety Limits. The radiological consequence events that previously used the F_R^A limit have been re-analyzed using the slightly higher F_R^T limit to determine the source strength. The revised calculated offsite dose and control room dose for the affected events remained within the guidelines of 10 CFR 100 and GDC 19.

Because the results of the transient analyses, which were performed without F_R^A

assumptions, continue to meet the Safety Limits, and because the dose consequences of all analyzed events, which were also performed without F_R^A assumptions, continue to be within the guidelines of 10 CFR 100 and GDC 19, the proposed technical specification change would not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, operation of the plant in accordance with the proposed Technical Specifications would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Operation of the plant in accordance with the proposed Technical Specifications would not add any new equipment, settings, or alter any plant operating practices. The only change is the deletion of all Technical Specifications references to the Assembly Radial Peaking Factor, F_R^A, (a peaking factor no longer used in core design or safety analyses). Since there will be no change in operating plant equipment, settings, or normal operating practices, operation in accordance with the proposed Technical Specifications would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The disposition of the [Standard Review Plan] SRP Chapter 15 events, the setpoint verification, the [fuel centerline melt] FCM and the [minimum departure from nucleate boiling ratio] MDNBR analyses documented in Siemens report EMF–2259 Revision 1, ''Palisades Cycle 15 Safety Analysis Report'' dated August 1999 considered the impact of several changes in fuel design and plant operations for Cycle 15. A detailed and simplified XCOBRA–IIIC model that incorporated limiting radial and axial power distributions, as well as the removal of the F_R^A peaking limit, were developed for Cycle 15. This model was applied to all DNB event analyses for Cycle 15 and the MDNBR values for limiting AOOs and PAs were evaluated with the [High Thermal Performance] HTP DNB correlation. The limiting MDNBR is calculated for SRP event 15.3.3 Reactor Coolant Pump Rotor Seizure and the limiting FCM is calculated for SRP event 15.4.3 Single Rod Withdrawal. The calculated results for the limiting events meet the Safety Limits specified in TS LCO 2.1.

The SRP events were dispositioned in accordance with Siemens approved methodologies listed in Palisades TS Section 5.6.5, Amendment 189. The completed safety analysis supports Palisades plant operation at 2530 Mwt.

The results of the transient analyses, which were performed without F_R^A assumptions, continue to meet the Safety Limits, and the dose consequence of all analyzed events, which were also performed without F_R^A assumptions, continue to be within the guidelines of 10 CFR 100 and GDC 19 * * * [Therefore] operation of the Facility in accordance with the proposed technical specification change would not involve a significant reduction in the margin of safety.

Therefore, operation of the plant in accordance with the proposed Technical Specifications 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.

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

Date of amendment request: March 1, 2001.

Description of amendment request: The amendments would allow implementation of 10 CFR Part 50, Appendix J, Option B, which governs performance-based containment leakage testing requirements for Types B and C testing. Catawba has previously implemented 10 CFR Part 50, Appendix J, Option B requirements for Type A testing. In addition to the changes associated with the adoption of 10 CFR Part 50, Appendix J, Option B, the licensee is also proposing the following two changes: (1) Technical Specification (TS) 3.6.3 will be modified to delete the requirement for conducting soap bubble tests of welded penetrations during Type A tests which are not individually Type B or Type C testable, and (2) the Bases for TS 3.6.2 will be modified to clarify that for the purpose of certain TS 3.6.2 Required Actions, the air lock door bulkhead is considered to be part of the door.

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 discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or
3. Involve a significant reduction in a margin of safety.

First Standard

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within acceptance limits as delineated in 10 CFR [Part] 50, Appendix J, Option B. The changes are consistent with current safety analyses. Although some of the proposed changes represent minor relaxation to existing TS requirements, they are consistent with the requirements specified by Option B of 10 CFR [Part] 50, Appendix J. The systems affecting containment integrity related to this proposed amendment request are not assumed in any safety analyses to initiate any accident sequence. Therefore, the probability of any accident previously evaluated is not increased by this proposed amendment. The proposed changes maintain an equivalent level of reliability and availability for all affected systems. In addition, maintaining leakage within analyzed limits assumed in accident analyses does not adversely affect either onsite or offsite dose consequences. Therefore, the proposed amendment does not increase the consequences of any accident previously evaluated.

Second Standard

The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No changes are being proposed which will introduce any physical changes to the existing plant design. The proposed changes are consistent with the current safety analyses. Some of the changes may involve revision in the testing of components; however, these are in accordance with the Catawba current safety analyses and provide for appropriate testing or surveillance that is consistent with 10 CFR [Part] 50, Appendix J, Option B. The proposed changes will not introduce new failure mechanisms beyond those already considered in the current safety analyses. No new modes of operation are introduced by the proposed changes. The proposed changes maintain, at minimum, the present level of operability of any system that affects containment integrity.

Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. The provisions specified in Option B of 10 CFR [Part] 50, Appendix J allow changes to Type B and Type C test intervals based upon the performance of past leak rate tests. 10 CFR [Part] 50, Appendix J, Option B allows longer intervals between leakage tests based on performance trends, but does not relax the leakage acceptance criteria. Changing test intervals from those currently provided in the TS to those provided in 10 CFR [Part] 50, Appendix J, Option B does not increase any risks above and beyond those that the NRC has deemed acceptable for the performance based option. In addition, there are risk reduction benefits associated with reduction in component cycling, stress, and wear associated with increased test intervals. The proposed changes provide continued assurance of leakage integrity of containment without adversely affecting the public health and safety and will not significantly reduce existing safety margins. Similar proposed changes have been previously reviewed and approved by the NRC, and they are applicable to Catawba.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard L. Emch, Jr..

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

Date of amendment request: March 29, 2001.

Description of amendment request: The proposed amendments would revise the Keowee Hydro Unit (KHU) Technical Specifications Surveillance Requirements (SRs) to address concerns related to voltage and frequency overshoot during surveillance testing. This would be accomplished by removing the note that had been implemented by Amendment Nos. 316, 316, and 316 (October 4, 2000) for Oconee Nuclear Station, Units 1, 2, and 3, respectively, to temporarily waive the upper limits specified in Surveillance Requirement 3.8.1.9, thereby reinstalling the original SR. In addition, as a result of an upgrade of the KHU governors, the proposed amendments would reduce the time delay specified in Technical Specification 3.8.1 and SR 3.8.1.17 from 12 seconds ±1 second to 5 seconds ±1 second. In addition, related Bases changes have been 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

No. The License Amendment Request (LAR) removes a Note to Surveillance Requirement (SR) 3.8.1.9 that temporarily waived the surveillance requirements associated with the upper limits for Keowee Hydro Unit (KHU) voltage and frequency. The waiver of these requirements allowed Duke to avoid an unplanned forced shutdown of all three Oconee units, and the potential safety consequences and operational risks associated with that action.

This LAR also changes the arming time delay associated with the out-of-tolerance logic that had been approved for installation in Amendment Nos. 312, 312, and 312. This change lowers the allowed time delay, thereby resulting in the activation of the out-of-tolerance logic more quickly after KHU startup.

Since this LAR assures that each KHU reaches its required operating band within the required time, and that if maloperation of a unit occurs, the KHU will be taken off line, the probability or consequences of an accident previously evaluated is not significantly increased.

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

No. The LAR involves removing a Note that temporarily waived SR 3.8.1.9.a associated with the KHUs. This LAR also changes the time delay associated with the activation of out-of-tolerance logic that had been approved for installation in Amendment Nos. 312, 312, and 312. This change lowers the allowed time delay, thereby resulting in the activation of the out-of-tolerance logic more quickly after KHU startup.

Since this LAR restores Technical Specification SR 3.8.1.9 to the condition prior to Amendment Nos. 316, 316, and 316 and provides a shortened arming delay for the out-of-tolerance logic that was approved in Amendment Nos. 312, 312, and 312, no new failure mechanism or accident sequence is introduced. Therefore, the possibility of a new or different kind of accident from any kind of accident previously evaluated is not created.

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

No. The LAR involves removing a Note that allowed temporary waiver of the requirements to meet SR 3.8.1.9.a and shortens the arming time delay associated with the activation of out-of-tolerance logic that had been approved for installation in Amendment Nos. 312, 312, and 312.

This LAR, therefore, improves the margin of safety by assuring that SR 3.8.1.9.a can be implemented. The change to a shorter arming time delay for the out-of-tolerance circuit activation also improves the margin of safety by limiting the time that a KHU would be carrying safety loads in an out-of-tolerance condition.

Therefore, this request 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 13, 2001.

Description of amendment request: The proposed change relaxes the allowable cooldown rate in the Reactor Coolant System (RCS) Technical Specifications (TS) 3.4.8.1, ''Pressure / Temperature Limits.'' Specifically, the change eliminates the limitation of a 10 °F per hour cooldown rate when the RCS temperature is below 135 °F. The proposed limitations permit a 100 °F per hour cooldown rate to continue down to an RCS temperature of 110 °F, at which point the rate is reduced to 30 °F per hour.

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

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

Response:

Limitations have been imposed on cooldown of the Reactor Coolant System (RCS) to assure compliance with the minimum temperature requirements of 10 CFR [Part] 50, Appendix G. The proposed changes revise the allowable cooldown limits in a way such that operation remains consistent with the design assumptions and satisfies the stress limits for cyclic operation. By ensuring operation remains within the bounds of the existing design basis and assumptions, the probability of a brittle fracture of the reactor vessel has not been increased.

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

2. Will the operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed changes will not create the possibility of a new or different kind of accident from any previously analyzed since they do not introduce new systems, failure modes, or other plant perturbations. The proposed changes revise the cooldown limitations based on the fact the conservatively estimated peak pressure that can occur when the RCS cold leg temperature is below 200 °F is less than the proposed pressure limit. The limits assure that operation remains consistent with the design assumptions and satisfies the stress limits for cyclic operation.

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

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

Response:

The margin of safety provided by Technical Specification 3.4.8.1 is based on assuring that the maximum cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. The proposed changes will not involve a significant reduction in the margin of safety since equivalent pressure and temperature limit requirements for reactor operation will be applied. The proposed changes were derived in accordance with approved NRC methodology which was developed to assure the reactor coolant system pressure boundary is designed with sufficient margin to withstand any condition during normal operation including anticipated operational occurrences and system in-service leak and hydrostatic tests.

These requirements were revised in accordance with 10 CFR [Part] 50, Appendix G utilizing the latest NRC guidance in Regulatory Guide 1.99, Revision 2 relative to estimating neutron irradiation damage to the reactor vessel. In addition, the 16 EFPY [effective full power year] basis for these pressure/temperature limits has been found to include sufficient margin to account for the limits of uncertainty described in Draft Regulatory Guide DG–1053.

Therefore, the proposed change will not involve in 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: N.S. Reynolds, Esquire, Winston & Strawn, 1400 L Street NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 9, 2001.

Description of amendment request: The proposed amendment (one-time change) revises the Steam Generator (SG) inspection frequency requirements in TS 5.5.9.d.2, ''Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies,'' for the Braidwood Station, Unit 1, Fall 2001 refueling outage, to allow a 40 month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C–1 classification. This one-time change is proposed to eliminate unnecessary SG inspections during the upcoming Unit 1, Fall 2001 refueling outage, thus, resulting in significant dose, schedule, and cost savings.

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

The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in Technical Specifications (TS) 5.5.9.d.2, ''Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies,'' for the Braidwood Station, Unit 1, Fall 2001 refueling outage, to allow a 40 month inspection frequency after one inspection, rather than after two consecutive inspections results that are within the C–1 category. C–1 category is defined as ''<5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.''

The proposed one-time extension of the Unit 1, SG tube inservice inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the Byron/Braidwood Stations' Updated Final Safety Analysis Report (UFSAR). As such, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would not go undetected. Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last Braidwood Station, Unit 1, refueling outage (i.e., the first refueling outage following SG replacement) exceeded the TS requirements for the first two refueling outages after SG replacement. That is, more tubes were inspected than were required by TS. Currently, Braidwood Station, Unit 1, does not have an active SG damage mechanism, and will meet the current industry examination guidelines without performing SG inspections during the next refueling outage. Additionally, as part of our SG Tube Surveillance Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines,'' performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the Braidwood Station, Unit 1, Spring 2000 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with the improved SG design and reliability of Babcock and Wilcox International (BWI) replacement SGs, extending the SG tube inspection frequency does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

The proposed change revises the SG inspection frequency requirements in TS 5.5.9.d.2, ''Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies,'' for the Braidwood Station, Unit 1, Fall 2001 refueling outage, to allow a 40 month inspection interval after one inspection, rather than after two consecutive inspections with inspection results within the C–1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during the last Braidwood Station, Unit 1, refueling outage (i.e., the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary to secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection frequency, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components.

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

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

The SG tubes are an integral part of the Reactor Coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system.

The safety function of the SGs is maintained by ensuring the integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for the same model of replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

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

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

Attorney for licensee: Mr. Robert Helfrich, Senior Counsel, Nuclear, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, Illinois 60515.

NRC Section Chief: Anthony J. Mendiola.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: April 6, 2001.

Description of amendment request: The proposed amendment would change the Kewaunee Nuclear Power Plant Technical Specification Section 6.2, ''Organization,'' and Section 6.13, ''High Radiation Area'' to reflect the title change from Shift Supervisor to Shift Manager.

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

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

The proposed change will not alter the intent of the TS. Changing the title from Shift Supervisor to Shift Manager is administrative in nature. It has no impact on accident initiators or plant equipment, and thus, does not affect the probability or consequences of an accident.

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

The proposed change does not involve a change to the physical plant or operations. Since this is an administrative change it does not contribute to accident initiation. Therefore, it does not produce a new accident scenario or produce a new type of equipment malfunction.

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

Since this is an administrative change, it does not involve a significant reduction in the margin of safety. The proposed change does not affect plant equipment or operation. Safety limits and limiting safety system settings are not affected by this change.

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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: February 7, 2001.

Description of amendment request: The proposed amendment would revise the technical specifications to replace the accident source term used in all design basis site boundary and control room dose analysis with the alternate source term. Additionally, the proposed amendment would implement regulatory guidance provided in Regulatory Guide (RG) 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' regarding the licensing basis source term for design basis events.

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

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

The proposed changes to FCS [Fort Calhoun Station] TS [Technical Specifications] modify requirements to: place the control room ventilation system in operation and in filtered air mode during refueling operations in the containment or spent fuel pool, place a spent fuel pool area radiation monitor in operation during refueling operations at the spent fuel pool, delete a specification that requires a ventilation isolation actuation signal (VIAS) and two radiation monitors to be operable, increase the volume of trisodium phosphate (TSP) in the reactor containment building, include both internal and external leakage for the residual heat removal (RHR) system leakage test, perform an internal leakage test on the RHR system, and credit the alternative source term (AST) for the design basis site boundary and control room dose analyses. These TS changes do not impact operation of other equipment or systems important to safety. The proposed TS changes reflect the parameters used in the radiological consequences calculations described in Attachment E [to the licensee's February 7, 2001, letter].

The current TS 3.16 limits RHR system leakage to 1243 cc/hour from external sources and does not provide a limit for leakage from internal sources due to valve seat back leakage to the safety injection refueling water tank (SIRWT) or require an internal leakage test to be performed. The re-analysis for LOCA [loss-of-coolant accident] assumed a total leakage from all RHR sources of 3800 cc/hour. The internal leakage would leak back into the water remaining in the SIRWT. While it appears the allowable leakage is being increased, the limit is more inclusive, and therefore, more conservative than the current leakage limit. The internal leakage test performed on the RHR system will measure and quantify the back leakage into the SIRWT.

The proposed changes to TSs 2.3 and 3.6 are necessary to ensure the post-LOCA pH of the recirculation water is equal to or greater than 7.0. Radiation levels in containment following a LOCA may cause the generation of hydrochloric and nitric acids from radiolysis of cable insulation and sump water. TSP will neutralize these acids. The radiolysis analysis performed demonstrates that the sump pH will be greater than or equal to 7.0 post design basis accident (DBA), which meets the intent of RG 1.183 regarding iodine revolatization. Therefore, there is no increase in the probability or consequences of an accident previously evaluated due to radiolysis concerns.

The proposed change to TS 2.8.2(4) requires the control room ventilation system to be in operation and in the Filtered Air mode. This is a conservative action to reduce control room operator exposure. This action is credited in the fuel handling accident analysis. 10 CFR 50.36 requires, in part, that if an operating restriction is an initial condition of a DBA, then a Limiting Condition for Operation (LCO) should be established. Therefore, this action, which will reduce operator exposure, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 2.8.3(5) will delete the requirement for the ventilation isolation actuation signal (VIAS) to be operable with two radiation monitors operable, and require the control room ventilation system to be in operation and in the Filtered Air mode and a spent fuel pool area radiation monitor to be in operation during refueling operations in the spent fuel pool. The current basis for TS 2.8.3(5) is to ensure the control room ventilation system is operated in Filtered Air mode upon receipt of a VIAS. The proposed change will require the control room ventilation system placed in the Filtered Air mode during refueling operations, thereby eliminating the need for the VIAS to be operable. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes proposed do not affect the precursors for accidents or transients analyzed in Chapter 14 of the FCS USAR. Therefore, there is no increase in the probability of accidents previously evaluated. The probability remains the same since the accident analyses performed and discussed in the basis for the TS changes, involve no change to a system, component or structure that affects initiating events for any USAR Chapter 14 accident evaluated. A re-analysis of USAR Chapter 14 events was conducted with respect to radiological consequences. This re-analysis was performed in accordance with current accepted methodology, and consequences were expressed in terms of TEDE [total effective dose equivalent] dose. The current methodology is no longer exactly comparable to the previous methods used for dose consequences. The previous dose calculations analyzed the dose consequences to thyroid and whole body as a result of postulated DBA events. The previous dose calculations were shown to be well below the regulatory limits of 10 CFR 100.11 (25 percent) with respect to thyroid and whole body dose. The current accepted NRC methodology, as described in 10 CFR 50.67, specifies new dose acceptance criteria in terms of TEDE dose. The revised analyses for all evaluated DBA events meet the applicable TEDE dose acceptance criteria (specified also in RG 1.183) for alternative source term implementation. The most current analyses do not credit several engineered safeguards features (ESF) filtration systems as the previous analyses did, and hence, are more conservative in that aspect. If a comparison is performed between the previous calculations (thyroid and whole body dose) and revised analyses TEDE results (per method shown in footnote 7 of RG 1.183), a slight increase in dose consequences is exhibited but is not significant, and the TEDE results are below regulatory acceptance criteria.

The changes proposed do not increase the probability of an accident previously evaluated. Because of the new regulatory requirements related to AST implementation, the dose consequences, if compared to previous ones, are only slightly increased (using guidance in footnote 7 of RG 1.183). However, the dose consequences of the revised analyses are below the AST regulatory acceptance criteria.

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

The implementation of the proposed changes does not create the possibility of an accident of a different type than was previously evaluated in the USAR. The proposed changes to FCS TS modify requirements to: place the control room ventilation system in operation and in filtered air mode during refueling operations in the containment or spent fuel pool, place a spent fuel pool area radiation monitor in operation during refueling operations at the spent fuel pool, delete a specification that requires a ventilation isolation actuation signal (VIAS) and two radiation monitors to be operable, increase the volume of trisodium phosphate (TSP) in the reactor containment building, include both internal and external leakage for the residual heat removal (RHR) system leakage test, perform an internal leakage test on the RHR system, and credit the alternative source term (AST) for the design basis site boundary and control room dose analyses[.]

The changes proposed do not change how DBA events were postulated nor do the changes themselves initiate a new kind of accident with a unique set of conditions. The changes proposed were based on a complete re-analysis of offsite and control room operator doses, where the system requirements being revised were not credited in the calculations. The revised analyses are consistent with the regulatory guidance established in RG 1.183. The revised analyses utilize the most current understanding of source term timing and chemical forms as a more appropriate mitigation technique. Not crediting filtration systems and only crediting natural forces is conservative from the aspect of dose consequences. Through this re-analysis, no new accident initiator or failure mode was identified.

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

The implementation of the proposed changes does not reduce the margin of safety. The radiological analyses results, with the proposed changes, remain within the regulatory acceptance criteria (10 CFR 50 Appendix A, 10 CFR 50.67) utilizing the TEDE dose acceptance criteria directed in RG 1.183. These criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting these limits demonstrates adequate protection of public health and safety. An acceptable margin of safety is inherent in these licensing limits. Therefore, there is no significant reduction in the margin of safety as a result of the proposed changes.

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: April 6, 2001.

Description of amendment requests: The amendment application proposes to revise the Facility Operating License No. NPF–10, and Facility Operating License No. NPF–15 for San Onofre Nuclear Generating Station, Units 2 and 3, respectively. The licensee proposed to add annotations to technical specification Surveillance Requirements 3.8.1.2, 3.8.1.3, 3.8.1.9, 3.8.1.10 and 3.8.1.19 that provide guidance to ensure a diesel generator sub-component, an automatic voltage regulator (AVR), is operable and regularly tested. The proposed annotations clarify that only one AVR is required for the associated diesel generator to be operable and only one AVR can be in service at any one time.

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

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

Response: No.

The diesel generators provide emergency power to accident mitigation equipment in the event of a loss of offsite power. They cannot cause an accident. The San Onofre Nuclear Generating Station (SONGS) emergency diesel generators (EDG) each have two automatic voltage regulators (AVRs) that are 100% redundant to each other. Maintaining both AVRs for each diesel in a high state of readiness, while minimizing unnecessary testing on the diesels, optimizes the overall availability of the diesel generator systems to perform their function if required.

This change allows testing the two AVRs for each diesel on a staggered monthly basis. In addition, it clarifies that each AVR only needs to be subjected to a dynamically challenging test once every 24 months provided that its dynamic performance is measured and determined to be acceptable. These testing requirements demonstrate a high level of assurance that each AVR will be capable of performing its design function while minimizing unnecessary wear on the diesels. The reliability of the diesel generators to provide emergency power will not be degraded as a result of this change. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Response: No.

The AVRs are a subcomponent of the EDGs. This change to the surveillance test frequency does not physically change the use, function, or design of the EDG or its subcomponent, the AVR.

This change ensures both 100% capacity AVRs are adequately tested to ensure operability without increasing the number of test starts of the EDGs.

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

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

Response: No.

This change allows testing the two AVRs for each diesel on a staggered monthly basis. In addition, it clarifies that each AVR only needs to be subjected to a dynamically challenging test once every 24 months provided that its dynamic performance is measured and determined to be acceptable. These testing requirements demonstrate a high level of assurance that each AVR will be capable of performing its design function while minimizing unnecessary wear on the diesels. This proposed change does not involve an alteration of the SONGS 2 and 3 design. The reliability of the diesel generators to provide emergency power will not be degraded as a result of this change.

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

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

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: January 11, 2001.

Description of amendment request: The proposed amendments would revise Technical Specifications 5.5.17, ''Containment Leakage Rate Testing Program,'' to add an exception to Regulatory Guide (RG) 1.163, ''Performance-Based Containment Leak-Testing Program.'' Specifically, the licensee proposes to use America Society of Mechanical Engineering, Subsections IWL and IWE to meet the intent of RG 1.163. The proposed change will affect the frequency of containment concrete visual examinations and allow the examinations to be preformed during power operation instead of exclusively during refueling outages.

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

No. The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 containments for the purpose of the Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The frequency of visual examinations of the concrete surfaces of the containments and the mode of operation during which those examinations are performed has no relationship to or adverse impact on the probability of any of the initiating events assumed for the accident analyses. Therefore, the proposed change does not involve a significant increase in the probability of any accident previously evaluated. The proposed change would allow visual examinations that are performed pursuant to NRC-approved [American Society of Mechanical Engineering] ASME Section XI Code requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations required by Regulatory Guide 1.163, without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the Code-required visual examinations. Therefore, the safety function of the VEGP containments as a fission product barrier will be maintained, and there will not be a significant increase in the consequences of an accident previously evaluated.

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

No. The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 containments for the purpose of the Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The proposed change does not adversely affect or otherwise alter plant operation. No new equipment is introduced, and no new limiting single failures are created. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 containments for the purpose of the Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The proposed change would allow visual examinations that are performed pursuant to NRC-approved ASME Section XI Code requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations required by Regulatory Guide 1.163, without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the Code-required visual examinations. Therefore, the safety function of the VEGP containments as a fission product barrier will be maintained, and there will not be 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: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of application for amendments: April 12, 2001 (TS 00–02).

Brief description of amendments: The proposed amendment would change the Sequoyah Nuclear Plant Technical Specification (TS) surveillance requirements for the ice condenser. The request would change the method and frequency for determining boron concentration and pH of the ice and proposes an additional test requirement for ice that is to be added to the ice condenser.

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

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

The only analyzed accidents of possible consideration in regards to changes potentially affecting the ice condenser are a loss of coolant accident (LOCA) and a main steam line break (MSLB) inside containment. However, the ice condenser is not postulated as being the initiator of any LOCA or MSLB. This is because it is designed to remain functional following a design basis earthquake, and the ice condenser does not interconnect or interact with any systems that interconnect or interact with the reactor coolant or main steam systems. Since the proposed changes to the TS and TS bases are solely to revise and provide clarification of the ice sampling and chemical analysis requirements, and are not the result of or require any physical change to the ice condenser, there can be no change in the probability of an accident previously evaluated in the Safety Analysis Report.

In order for the consequences of any previously evaluated event to be changed, there would have to be a change in the ice condenser's physical operation during a LOCA or MSLB, or in the chemical composition of the stored ice. The proposed changes do not alter either from existing requirements, except to add an upper limit on boron concentration, which is the bounding value for the hot leg switchover timing calculation. Though the frequency of the existing surveillance requirement (SR) for sampling the stored ice is changed from once every 18 months to once every 54 months, the sampling requirements are strengthened overall with: (1) the requirement to obtain one randomly selected sample from each ice condenser bay (24 total samples) rather than 9 ''representative'' samples, and (2) the addition of a new SR to verify each addition of ice meets the existing requirements for boron concentration and pH value. The only other change is to clarify that each sample of stored ice is individually analyzed for boron concentration and pH, but that the acceptance criteria for each parameter is based on the average values obtained for the 24 samples. This is consistent with the bases for the boron concentration of the ice, which is to ensure the accident analysis assumptions for containment sump pH and boron concentration are not altered following complete melting of the ice condenser. Historically, chemical analysis of the stored ice has had a very limited number of instances where an individual sample did not meet the boron or pH requirements, with all subsequent evaluations (follow-up sampling) showing the ice condenser as a whole was well within these requirements. Requiring chemical analysis of each sample is provided to preclude the practice of melting all samples together before performing the analysis, and to ensure the licensee is alerted to any localized anomalies for investigation and resolution without the burden of entering a 24-hour action, provided the averaged results are acceptable. Thus, based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The ice condenser TSs ensure that during a LOCA or MSLB the ice condenser will initially pass sufficient air and steam mass to preclude over pressurizing lower containment, that it will absorb sufficient heat energy initially and over a prescribed time period to assist in precluding containment vessel failure, and that it will not alter the bulk containment sump pH and boron concentration assumed in the accident analysis. Since the proposed changes do not physically alter the ice condenser, but rather only serve to strengthen and clarify ice sampling and analysis requirements, the only area of potential concern is the effect these changes could have on bulk containment sump pH and boron concentration following ice melt. However, this is not affected because there is no change in the existing requirements for pH and boron concentration, except to add an upper limit on boron concentration. This upper limit is the bounding value for the hot leg switchover timing calculation. Averaging the pH and boron values obtained from analysis of the individual samples taken is not a new practice, just one that was not consistently used by all ice condenser plants. Using the averaged values provides an equivalent bulk value for the ice condenser, which is consistent with the accident analysis for the bulk pH and boron concentration of the containment sump following ice melt. Changing the performance frequency for sampling the stored ice does not reduce any margin of safety because: (1) the newly proposed surveillance (SR 4.6.5.1.f) ensures ice additions meet the existing boron concentration and pH requirements, (2) there are no normal operating mechanisms, including sublimation, that reduce the ice condenser bulk pH and boron concentration, and (3) the number of required samples has been increased from 9 to 24 (1 randomly selected ice basket per bay), which is approximately the same number of samples that would have been taken in the same time period under the existing requirements. Thus, it can be concluded that the proposed TS and TS bases changes do not involve a significant reduction in the 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.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: April 3, 2001.

Brief description of amendments: The proposed change would revise Technical Specification (TS) 3.3.6, ''Containment Ventilation Isolation Instrumentation'' to modify the Note for Required Action B.1 such that it applies only to ''Required Action and associated Completion Time of Condition A not met.'' The proposed change is the result of the discovery of an error which occurred when the TS was converted to the ''improved TS format'' with issuance of License Amendment Nos. 64 and 64, for Comanche Peak Steam Electric Station, Units 1 and 2 on February 26, 1999.

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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change removes an allowance to open containment pressure relief valves under administrative controls when one train of Automatic Actuation Logic and Actuation Relays is inoperable. The proposed change corrects a non-conservative technical specification and thus makes the technical specifications consistent with the previously evaluated accident analyses. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change makes the technical specifications consistent with the previously evaluated accident analyses. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

The proposed change makes the technical specifications consistent with the previously evaluated accident analyses. Therefore the proposed change does not involve a reduction in a margin of safety.

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 22, 2001 (ET 01–0012).

Description of amendment request: The amendment would (1) decrease the allowable values for Function 8, pressurizer pressure-low, pressurizer pressure-high, in Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' and (2) decrease the allowable value for pressurizer pressure-low for safety injection in Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation.'' The changes are needed because the licensee will be replacing the existing Tobar pressurizer pressure transmitters with Rosemount transmitters in the next refueling outage.

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

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

The existing safety related pressurizer pressure transmitters are being replaced with ones of similar characteristics and functions, and without changing the design or functional basis of the system, structure, or components associated with the pressure transmitters.

The protection system performance will remain within the bounds of the previously performed accident analysis. The Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation will continue to function in a manner consistent with the plant design basis. The replacement of the pressurizer pressure transmitters and proposed changes to the affected Allowable Values will not affect any of the analysis assumptions for any of the accidents previously evaluated, since the changes are consistent with the setpoint methodology and ensure adequate margin to the Safety Analysis Limit. The proposed changes will not affect any event initiators nor will the proposed changes affect the ability of any safety related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation capabilities.

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.

A review of the failure modes and effects in Updated Safety Analysis Report Section 7.7.2 found that failure of the replacement pressure transmitters will be the same as for the existing pressure transmitters. As such, the effects of such failures on [the safety] functions of the other equipment are concluded to be similar to those previously evaluated.

There are no changes in the method by which any safety related plant system performs its safety function. The normal manner of plant operation remains unchanged. The increase [or decrease] in the pressurizer pressure functions Allowable Values still provides acceptable margin between the nominal Trip Setpoint and Allowable Value. The changes in Allowable Value does not impact the systems capability to provide both control and protection functions. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

Therefore, the proposed changes do 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 changes do not affect the acceptance criteria for any analyzed event nor is there a change in any Safety Analysis Limit. There will be no effect on the manner in which safety limits or RTS and ESFAS settings are determined nor will there be any affect on those plant systems necessary to assure the accomplishment of protection functions. [The proposed changes to the pressurizer pressure Allowable Values will maintain the accident analyses in the Updated Safety Analysis Report.]

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: March 23, 2001 (CO 01–0013).

Description of amendment request: The amendment proposes to (1) delete certain license conditions from the operating license, and (2) revise Table 5.5.9–2, ''Steam Generator Tube Inspection,'' in Section 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' of the technical specifications (TSs). License Conditions 2.C.(4) and 2.C.(6) through 2.C.(14) of the facility operating license are considered to have been completed and obsolete, or to duplicate other license requirements, and are proposed to be deleted. Attachments 2 and 3 to the facility operating license are also proposed to be deleted. Section 2.F of the facility operating license is considered to duplicate the reporting requirements in 10 CFR 50.72 and 50.73 and is proposed to be deleted. The reporting requirements in two ''Action Required'' columns of TS Table 5.5.9–2 are also considered to duplicate the reporting requirements in 10 CFR 50.72 and 50.73 and are proposed to be deleted. The list of the attachments and appendices to the facility operating license would also be revised to reflect the proposed deletion of Attachments 2 and 3.

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

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

This request involves administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed changes. 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.

This request involves administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, the proposed changes do 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.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves administrative changes only [and does not change these barriers].

No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety system settings, or will not relax the bases for any limiting conditions of operation [in the TSs]. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: April 3, 2001 (ET 01–0008).

Description of amendment request: The amendment would make the following changes to the technical specifications (TSs):

(1) Revise Safety Limit 2.1.1 by replacing Figure 2.1.1–1, ''Reactor Core Safety Limits,'' with a reference to limits being specified in the Core Operating Limits Report (COLR) and by adding two reactor core safety limits on departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature.

(2) Revise Note 1 on the over temperature ΔT in Table 3.3.1–1 of TS 3.3.1, ''Reactor Trip System Instrumentation,'' by replacing values of parameters with a reference to the values being specified in the COLR and correcting the expression for one term in the inequality for over temperature ΔT.

(3) Revise Note 2 on the overpower ΔT in Table 3.3.1–1 by replacing values of parameters with a reference to the values being specified in the COLR.

(4) Replace the limits for the reactor coolant system (RCS) pressure and average temperature with a reference to the limits being specified in the COLR for Limiting Condition for Operation (LCO) 3.4.1 and Surveillance Requirements (SRs) 3.4.1.1 and 3.4.1.2.

(5) Add the phrase ''and greater than or equal to the limit specified in the COLR'' to the RCS total flow rate in LCO 3.4.1 and SRs 3.4.1.3 and 3.4.1.4.

(6) Move items a. and b. to the left in the Note to the applicability in LCO 3.4.1.

(7) Revise TS Section 5.6.5 by adding TS 3.3.1 on over temperature and overpower ''T trip setpoints and TS 3.4.1 on RCS pressure, temperature, and flow limits to the existing list of core operating limits for each reload cycle that are documented in the COLR and revising the list of topical reports in the COLR that represent the analytical methods approved by the Commission to determine core operating limits.

The proposed changes remove cycle-specific parameter limits and relocate them to the COLR, but they (1) do not change any of the limits, (2) add more specific requirements regarding DNBR limit and peak fuel centerline temperature limit to the TSs, (3) revise the list of topical reports in the list of NRC-approved analytical methods, (4) correct one term of an expression, and (5) move terms in a Note to the mode applicability for an LCO.

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

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

The proposed changes are programmatic and administrative in nature which do not physically alter safety related systems, nor affect the way in which safety related systems perform their functions. More specific requirements regarding the safety limits (i.e., DNBR limit and peak fuel centerline temperature limit) are being imposed in TS 2.1.1, ''Reactor Core Safety Limits,'' which replace the Reactor Core Safety Limits figure and are consistent with the values stated in the USAR [Updated Safety Analysis Report]. The proposed changes remove the cycle-specific parameter limits from TS 3.4.1 and relocate them to the COLR which do not change plant design or affect system operating parameters. In addition, the minimum limit for RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used. The proposed changes do not, by themselves, alter any of the parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. The existing TS Section 5.6.5b, COLR Reporting Requirements, continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies. The existing TS Section 5.6.5c, COLR Reporting Requirements, continues to ensure that applicable limits of the safety analyses are met.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the COLR without having to submit an amendment to [the TS of] the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

Although the relocation of the cycle-specific parameter limits to the COLR would allow revision of the affected parameter limits without prior NRC approval, there is no significant effect on the probability or consequences of an accident previously evaluated. Future changes to the COLR parameter limits could result in event consequences which are either slightly less or slightly more severe than the consequences for the same event using the present parameter limits. The differences would not be significant and would be bounded by the existing requirement of TS Section 5.6.5c to meet the applicable limits of the safety analyses.

The cycle-specific parameter limits being transferred from the TS to the COLR will continue to be controlled under existing programs and procedures. The USAR accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies, ensuring that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, the proposed changes do not allow for an increase in plant power levels, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not change the types or increase the amounts of any effluents released offsite.

[The proposed changes to the expression of the $f_1(\Delta I)$ term, which is in the over temperature ΔT inequality, clarifies and corrects the term. Moving the terms in a Note to the LCO mode applicability is an administrative action.]

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

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

[The proposed changes are programmatic and administrative in nature which do not physically alter safety related systems, nor affect the way in which safety related systems perform their functions.]

The proposed changes that retain the minimum limit for RCS total flow rate in the TS, and that relocate certain cycle-specific parameter limits from the TS to the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters, do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no changes being made to the parameters within which the plant is operated, other than their relocation to the COLR. There are no setpoints affected by the proposed changes at which protective or mitigative actions are initiated. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analytical limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the COLR without having to submit an amendment to [the TS of] the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

Relocation of cycle-specific parameter limits has no influence or impact on, nor does it contribute in any way to the possibility of a new or different kind of accident. The relocated cycle-specific parameter limits will continue to be calculated using the NRC reviewed and approved methodology. The proposed changes do not alter assumptions made in the safety analysis and operation within the core operating limits will continue.

[The proposed changes to the expression of the $f_1(\Delta I)$ term, which is in the over temperature ΔT inequality, clarifies and corrects the term.]

Therefore, the proposed changes do 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 margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes [are programmatic and administrative in nature and] do not physically alter safety related systems, nor does it [a]ffect the way in which safety-related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As the proposed changes to relocate cycle-specific parameter limits to the COLR will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameter, and the plant will continue to operate within prescribed limits.

The development of cycle-specific parameter limits for future reload designs will continue to conform to NRC reviewed and approved methodologies, and will be performed pursuant to 10 CFR 50.59 to assure that plant operation [is] within cycle-specific parameter limits.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of [the] current Topical Reports to support limits in the COLR without having to submit an amendment to [the TS of] the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

[The proposed changes to the expression of the $f_1(\Delta I)$ term, which is in the over temperature ΔT inequality, clarifies and corrects the term. Moving the terms in a Note to the LCO mode applicability is an administrative action.]

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

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

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

NRC Section Chief: Stephen Dembek.

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).

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: February 28, 2001.

Brief description of amendments: The amendments revise the definitions of engineered safety feature response time and reactor protection system response time in Technical Specification (TS) 1.1, ''Definitions,'' to add the following statement: ''In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the [Nuclear Regulatory Commission] NRC.'' Approval of the amendments will allow either an allocated sensor response time or a measured sensor response time for the identified Reactor Protection System and Engineered Safety Features Actuation System pressure sensors when performing response time testing.

Date of issuance: April 19, 2001.

Effective date: April 19, 2001, and shall be implemented within 45 days of the date of issuance.

Amendment Nos.: Unit 1–135, Unit 2–135, Unit 3–135.

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

Date of initial notice in ***Federal Register:*** March 20, 2001 (66 FR 15766).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 19, 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: December 21, 2000, as supplemented on February 12, 2001, and March 5, 2001.

Brief description of amendments: The amendments revise Technical Specification 5.2.2.e by removing the reference to the Nuclear Regulatory Commission Policy Statement on working hours.

Date of issuance: April 5, 2001.

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

Amendment Nos.: 245 and 219.

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

Date of initial notice in ***Federal Register:*** February 7, 2001 (66 FR 9380).

The February 12, 2001, and March 5, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated April 5, 2001.

No significant hazards consideration comments received: No.

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

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

Brief description of amendment: The amendment permits operation of Calvert Cliffs Unit 2 with a core containing a lead fuel (test) assembly that includes fuel rods with advanced zirconium alloy cladding.

Date of issuance: April 5, 2001.

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

Amendment No.: 220.

Renewed License No. DPR–69: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2012).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 5, 2001.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 19, 2000, as supplemented by letters dated March 16, 2001, and April 4, 2001.

Brief description of amendments: The proposed amendments revised the technical specifications to remove their applicability related to the Boron Dilution Protection System (BDPS) after the next refueling outage for each unit. During the refueling outages, modifications are scheduled to be made which will permit mitigation of a boron dilution event without the use of the BDPS.

Date of issuance: April 6, 2001.

Effective date: Immediately, to be implemented upon completion of the modifications scheduled to be completed after cycle 9 for Byron, Unit 2, and Braidwood, Units 1 and 2, and after cycle 11 for Byron, Unit 1.

Amendment Nos.: 117 and 111.

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

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

Since the proposed additional changes provided in this supplement are more restrictive than the originally proposed changes, it does not change the previous determination of no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 6, 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: February 12, 2001.

Brief description of amendment: The amendment changes Technical Specification Section 5.6.5b, ''Reporting Requirements—Core Operating Limits Report (COLR),'' to add a report pertaining to statistical setpoint methodology to the list of approved methodology references.

Date of issuance: April 9, 2001.

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

Amendment No.: 195.

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

Date of initial notice in ***Federal Register:*** March 7, 2001 (66 FR 13801).

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

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: August 22, 2000, as supplemented by letter dated November 7, 2000.

Brief description of amendments: The amendments revise the Technical Specifications (TS) of each unit to restore a time limit for an allowable condition for the occurrence of an inoperable refueling water storage tank level transmitter in TS 3.3.2.

Date of issuance: April 12, 2001.

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

Amendment Nos.: 198 and 179.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** *November 1, 2000 (65 FR 65341).*

The supplement dated November 7, 2000, provided clarifying information that did not change the scope of the August 22, 2000, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: September 6, 2000, as supplemented on January 18, and April 2, 2001.

Brief description of amendment: The amendment revises Technical Specification 5.5.15 to allow a one time change in the 10 CFR part 50, Appendix J, Type A test interval from the required 10 years to a test interval of 15 years.

Date of issuance: April 17, 2001.

Effective date: April 17, 2001.

Amendment No.: 206.

Facility Operating License No. DPR–64:

Date of initial notice in ***Federal Register:*** January 24, 2000 (66 FR 7665).

The January 18, and April 2, 2001, submittals contained clarifying information only, 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 April 17, 2001.

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: February 16, 2001.

Brief description of amendment: This amendment substitutes a surveillance interval of ''Once/Operating Cycle'' for the current surveillance interval of ''Each Refueling Outage,'' for the following instruments in Technical Specification Table 4.2.F: Containment High Radiation Monitor, Reactor Building Vent Radiation Monitor, Main Stack Vent Radiation Monitor, and Turbine Building Vent Radiation Monitor.

Date of issuance: April 9, 2001.

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

Amendment No.: 189.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 7, 2001 (66 FR 13802).

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

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: November 22, 2000, as supplemented on January 30 and February 2, 2001.

Brief description of amendment: This amendment changes the pressure-temperature limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 of the Technical Specifications (TS) over operation between 20, 32, and 48 Effective Full Power Years. However, these curves will only apply for the remainder of operating cycles 13 and 14. The Bases section has been modified to reflect these TS changes.

Date of issuance: April 13, 2001.

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

Amendment No.: 190.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81915).

The January 30 and February 2, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: August 10, 2000, as supplemented by letter dated March 22, 2001.

Brief description of amendment: The amendment revised Technical Specification 3/4.9.4, ''Refueling Operations, Containment Building Penetrations,'' by deleting the requirements for the containment purge and exhaust system and by revising the closure requirements for containment building penetrations to require that containment penetrations are capable of being closed during the handling of irradiated fuel within the containment.

Date of issuance: April 18, 2001.

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

Amendment No.: 230.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 20, 2000 (65 FR 56950).

The March 22, 2001, 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 April 18, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: February 1, 2001, as supplemented March 6 and 23, 2001.

Brief description of amendment: This amendment revises the minimum critical power ratio safety limits for operating cycle 7.

Date of issuance: April 12, 2001.

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

Amendment No.: 114.

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

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11061).

The March 6 and 23, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 4, 2000, as supplemented February 9, 2001.

Brief description of amendment: This amendment changes the licensing bases to incorporate a revised analysis of the Main Steam Line Break inside containment.

Date of Issuance: April 20, 2001.

Effective Date: April 20, 2001.

Amendment No.: 175.

Facility Operating License No. DPR–67: Amendment revised the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** February 7, 2001 (66 FR 9383).

The February 9, 2001, Supplement did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: December 18, 2000.

Description of amendment request: The amendment deletes Technical Specifications Section 6.7.6.e, ''Post-Accident Sampling,'' for Seabrook Station, Unit No. 1 and thereby eliminates the requirements to have and maintain the post-accident sampling system.

Date of issuance: April 17, 2001.

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

Amendment No.: 78.

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

Date of initial notice in ***Federal Register:*** January 24, 2000 (66 FR 7683).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 17, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 19, 2000, as supplemented November 16, 2000, and April 9, 2001, and as limited in scope by letter dated March 23, 2001.

Brief description of amendment: The amendment revises the Technical Specifications regarding operability requirements during core alterations and while moving irradiated fuel assemblies within the secondary containment. The amendment also provides for a change in design and licensing bases for a selective application of the alternate radiological source term in accordance with 10 CFR 50.67, ''Accident Source Term,'' and revised meteorology dispersion values, both being limited to a design-basis fuel handling accident.

Date of issuance: April 16, 2001.

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

Amendment No.: 237.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications and the licensing and design bases regarding a design-basis fuel handling accident.

*Date of initial notice in **Federal Register:*** March 6, 2001 (66 FR 13598).

NMC's letters dated March 23 and April 9, 2001, are within the scope of the changes proposed in NMC's letter of October 19, 2000, that was noticed in the **Federal Register** on March 6, 2001.

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

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: April 17, 2000, as supplemented February 2, 2001.

Brief description of amendments: The amendments change the Technical Specifications (TSs) for removal of boric acid storage tanks from the safety injection (SI) system. These changes accomplish two objectives: (1) Eliminate high concentration boric acid from the SI system and (2) align this specific TS section with the standard TSs.

Date of issuance: April 16, 2001.

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

Amendment Nos.: 156 and 147.

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

*Date of initial notice in **Federal Register:*** March 7, 2001 (66 FR 13806).

The February 2, 2001, supplement provided clarifying information that was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 16, 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: October 4, 2000, as supplemented March 12, April 2, and April 5, 2001.

Brief description of amendments: The amendments revise the surveillance test requirements for excess flow check valves (EFCVs) to allow testing of a representative sample at 24-month intervals such that each EFCV is tested at least once every 10 years.

Date of issuance: April 11, 2001.

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

Amendment Nos.: 193 and 168.

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 2021).

The March 12, April 2, and April 5, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 11, 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: January 18, 2001, as supplemented by letter dated February 21, 2001.

Brief description of amendment: The amendment authorizes the installation of new engineered safety feature transformers as an improvement. This amendment will allow the installation and use of the new transformers equipped with automatic load tap changers and an update to the Final Safety Analysis Report (FSAR) to reflect their installation.

Date of issuance: April 6, 2001.

Effective date: April 6, 2001, and shall be implemented in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e).

Amendment No.: 143.

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

*Date of initial notice in **Federal Register:*** February 21, 2001 (66 FR 11063).

The February 21, 2001, supplemental letter 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 April 6, 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: January 18, 2001.

Brief description of amendment: The amendment deletes Section 5.5.3, ''Post Accident Sampling,'' of the Technical Specifications for the Callaway Plant and thereby eliminates the requirements to have and maintain the post-accident sampling system (PASS).

Date of issuance: April 6, 2001.

Effective date: April 6, 2001, to be implemented within 60 days of the date of issuance.

Amendment No.: 144.

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

*Date of initial notice in **Federal Register:*** March 7, 2001 (66 FR 13808).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 6, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland this 25th day of April 2001.

For The Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–10822 Filed 5–1–01; 8:45 am]

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