United States of America as represented by the Administrator of the National Aeronautics and Space Administration. Written objections to the prospective grant of a license should be sent to Ames Research Center.

DATES: Responses to this notice must be received by September 10, 2001.

FOR FURTHER INFORMATION CONTACT: Robert Padilla, Patent Counsel, NASA Ames Research Center, Mail Stop 202A–3, Moffett Field, CA 94035–1000, telephone (650) 604–5104.

Dated: July 2, 2001.

Edward A. Frankle,

General Counsel

[FR Doc. 01–17292 Filed 7–10–01; 8:45 am]

BILLING CODE 7510–01–P

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice 01–187]

Notice of Prospective Patent License

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of prospective patent license.

SUMMARY: NASA hereby gives notice that Wessex, Inc., of Blacksburg, VA has applied for an exclusive license to practice the invention disclosed in U.S. Patent No. 5,269,288, entitled ‘‘Protective Coating for Ceramic Materials,’’ which is assigned to the United States of America as represented by the Administrator of the National Aeronautics and Space Administration. Written objections to the prospective grant of a license should be sent to Ames Research Center.

DATES: Responses to this notice must be received by September 10, 2001.

FOR FURTHER INFORMATION CONTACT: Robert Padilla, Patent Counsel, NASA Ames Research Center, Mail Stop 202A–3, Moffett Field, CA 94035–1000, telephone (650) 604–5104.

Edward A. Frankle,

General Counsel.

[FR Doc. 01–17294 Filed 7–10–01; 8:45 am]

BILLING CODE 7510–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of July 9, 16, 23, 30, August 6, 13, 2001.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of July 9, 2001

There are no meetings scheduled for the Week of July 9, 2001.

Week of July 16, 2001—Tentative

Thursday, July 19, 2001

9:25 a.m.—Affirmation Session (Public Meeting) (If needed)
9:30 a.m.—Briefing on Results of Agency Action Review Meeting—Reactors (Public Meeting) (Contact: Ron Frahm, 301–415–2986)
1:30 p.m.—Briefing on Readiness for New Plant Applications and Construction (Public Meeting) (Contact: Nanette Gilles, 301–415–1180)

Friday, July 20, 2001

9:30 a.m.—Briefing on Results of Reactor Oversight Process Initial Implementation (Public Meeting) (Contact: Tim Frye, 301–415–1287)
1:00 p.m.—Briefing on Risk-Informing Special Treatment Requirements (Public Meeting) (Contact: John Nakoski, 301–415–1278)

Week of July 23, 2001—Tentative

There are no meetings scheduled for the Week of July 23, 2001.

Week of July 30, 2001—Tentative

Tuesday, July 31, 2001

1:25 p.m.—Affirmation Session (Public Meeting) (If needed)

Week of August 6, 2001—Tentative

There are no meetings scheduled for the Week of August 6, 2001.

Week of August 13, 2001—Tentative

Tuesday, August 14, 2001

9:30 a.m.—Briefing on NRC International Activities (Public Meeting) (Contact: Elizabeth Doroshuk, 301–415–2775)

Wednesday, August 15, 2001

9:30 a.m.—Briefing on EEO Program (Public Meeting) (Contact: Irene Little, 301–415–7380)
1:25 p.m.—Affirmation Session (Public Meeting) (If needed)
1:30 p.m.—Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting (Contact: John Zabko, 301–415–1277)

*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: David Louis Gamberoni (301) 415–1651

The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an elecontric message to dkw@nrc.gov.

Dated: July 5, 2001.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 01–17446 Filed 7–9–01; 8:45 am]

BILLING CODE 7590–01–M

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 June 18, 2001 through June 29, 2001. The last biweekly notice was published on June 27, 2001.

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 and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC 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 August 10, 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. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by email to pdr@nrc.gov.

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

Date of amendment request: May 31, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications to incorporate Cycle 14 specific limits for the variable low reactor coolant system pressure-temperature core protection safety limits. The proposed limits are developed in accordance with the methods described in NRC (Nuclear Regulatory Commission)-approved Topical Report BAW–10179P–A, ''Safety Criteria and Methodology for Acceptable Cycle Reload Analyses.''

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

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

The proposed Technical Specification limits (Figure 2.1–1) are developed in accordance with the methods and assumptions described in NRC-approved Framatome ANP Topical Report BAW–10179P–A, ''Safety Criteria and Methodology for Acceptable Cycle Reload Analyses.''

These limits remain bounded by the existing reactor protection system (RPS) trip setpoints. The TMI Unit 1 Cycle 14 core introduces the Framatome ANP Mark–B12 fuel design. The Mark–B12 fuel design is mechanically and hydraulically similar to fuel designs currently in use at TMI Unit 1. While the designs are hydraulically similar, the Mark–B12 contains a fine mesh debris filter that alters the flow characteristics at the core inlet relative to the resident fuel designs resulting in the identification of a transition core DNB [departure from nucleate boiling] penalty. The higher minimum RCS flow requirement (105.5%) applied to offset the transition core DNB penalty is bounded by the minimum RCS flow assumed in current Updated Final Safety Analyis Report (UFSAR) Chapter 14 accident analyses (102%).

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

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

The proposed Technical Specification limits (Figure 2.1–1) provide core protection safety limits developed in accordance with NRC-approved methods and assumptions. The revised Technical Specifications limits remain bounded by the existing reactor protection trip setpoints. The TMI Unit 1 Cycle 14 core introduces the Framatome ANP Mark–B12 fuel design. The Mark–B12 fuel design is mechanically and hydraulically similar to the fuel designs currently in use at TMI Unit 1. While the designs are hydraulically similar, the Mark–B12 contains a fine mesh debris filter that alters the flow characteristics at the core inlet relative to the resident fuel designs resulting in the identification of a transition core DNB penalty. The higher minimum reactor coolant system flow required for the transition cycles (105.5%) is within the current range of allowable operating flow rates since this value exceeds the minimum flow rate assumed for Chapter 14 accident analyses (102%) and is well below the maximum flow limit for fuel assembly lift which is typically approximately 115% of design flow (depending on fuel type and 4th reactor coolant pump startup temperature).

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

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

The existing RPS reactor coolant pressure and temperature trip setpoints bound the proposed Technical Specification core protection safety limits. The proposed safety limits are developed in accordance with NRC-approved safety methods and assumptions. The higher minimum reactor coolant system flow requirement assures safe operation commensurate with the introduction of the Mark–B12 fuel design into the TMI Unit 1 core.

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

NRC Section Chief: Richard P. Correia, Acting.

Dominion Nuclear Connecticut Inc., et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: March 2, 2001.

Description of amendment request: The proposed changes would modify Technical Specification (TS) 3.3.2, ''Instrumentation—Engineered Safety Features Actuation System Instrumentation.'' The Bases of the affected TS will also be modified to reflect this change. The proposed changes will extend the required surveillance interval for Potter & Brumfield MDR Series slave relays, which are installed in the Millstone Unit No. 3 Engineered Safety Features Actuation System, from a quarterly surveillance interval to an 18-month frequency surveillance interval for those relays that meet the reliability assessment criteria established by Westinghouse Electric Corporation.

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

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

This change to the Technical Specifications does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same ESFAS instrumentation is being used and the same ESFAS system reliability is expected. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter assumptions previously made in evaluating the radiological consequences of an accident described in the SAR [Safety Analysis Report]. Therefore, the proposed amendment does not result in any 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.

This change does not alter the performance of the ESFAS [engineered safety features actuation system] mitigation systems assumed in the plant safety analysis. Changing the surveillance interval for periodically verifying ESFAS slave relays (assuring equipment operability) will not create any new accident initiators or scenarios. Implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not affect the total ESFAS system response assumed in the safety analysis. The periodic slave relay functional verification is relaxed because of the demonstrated high reliability of the relay and its insensitivity to any short term wear or aging effects. It is thus concluded that the proposed license amendment request does not result in a reduction in margin with respect to plant safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

Dominion Nuclear Connecticut Inc., et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: April 23, 2001.

Description of amendment request: The proposed changes would modify Technical Specification (TS) 3.1.2.1, ''Reactivity Control Systems—Boration Systems—Flow Path—Shutdown;'' 3.1.2.2, ''Reactivity Control Systems—Flow Paths—Operating;'' 3.1.2.3, ''Reactivity Control Systems—Charging Pump—Shutdown;'' 3.1.2.4, ''Reactivity Control Systems—Charging Pumps—Operating;'' 3.1.2.5, ''Reactivity Control Systems—Borated Water Source—Shutdown;'' 3.1.2.6, ''Reactivity Control Systems—Borated Water Sources—Operating;'' 3.4.1.2, ''Reactor Coolant System—Hot Standby;'' 3.4.1.3, ''Reactor Coolant System—Hot Shutdown;'' 3.4.1.4.1, ''Reactor Coolant System—Cold Shutdown—Loops Filled;'' 3.4.1.4.2, ''Reactor Coolant System—Cold Shutdown—Loops Not Filled;'' 3.4.1.6, ''Reactor Coolant System—Isolated Loop Startup;'' 3.4.2.1, ''Reactor Coolant System—Safety Valves—Shutdown;'' 3.4.2.2, ''Reactor Coolant System—Operating;'' 3.4.9.1, ''Reactor Coolant System—Pressure/Temperature Limits;'' and 3.4.9.3, ''Reactor Coolant System—Overpressure Protection Systems.'' The Index and the associated Bases for these Technical Specifications will be modified as a result of the proposed changes.

The above proposed TS changes will relocate the boration subsystem and Residual Heat Removal System over-pressurization protection requirements to a licensee-controlled document; modify the Reactor Coolant System (RCS) pressure/temperature limits; modify the Cold Overpressure Protection System (COPPS) set-point curves, COPPS enable temperatures and associated restrictions; modify the reactor vessel material surveillance withdrawal schedule; modify the pressurizer code safety valve requirements; modify the isolated RCS loop startup requirements; and provide numerous minor enhancements to the current requirements.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis, which is based on the representations made by the licensee in the April 23, 2001, application, is presented below:

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

The proposed TS changes associated with the relocation of the boration subsystem requirements to a licensee-controlled document and with the revised reactor vessel analyses will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed TS changes associated with the relocation of the Mode 4 and Mode 5 plant restrictions associated with protection of the Residual Heat Removal System to a licensee-controlled document do not change the design-basis accidents of the same postulated events described in the Millstone Unit No. 3 Final Safety Analysis Report (FSAR). Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed TS changes associated with the pressurizer code safety valves do not change the design-basis accidents described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed TS changes to the Modes 5 and 6 restrictions associated with restoration of an isolated RCS loop do not change the design-basis accidents of the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The additional proposed changes to the TS that will standardize terminology, relocate information to the Bases, remove extraneous information, modify the requirements to prevent rod withdrawal for operational flexibility, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional proposed changes will not increase 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 changes to the TSs do not impact any system or component that could cause an accident, nor will it alter the plant configuration or require any unusual operator actions, nor will it alter the way any structure, system, or component functions. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The revised analyses are based on American Society of Mechanical Engineers (ASME) Section XI Code Case N–640, which provides an alternate reference fracture toughness curve (K^{Ic}) for establishment of the beltline P/T limits. The analyses restrictions are less restrictive than those associated with the current analyses. However, the reduction in the margin of safety is small relative to the conservatism provided by ASME Section XI margins. Therefore, the proposed changes will not result in a significant reduction in a margin of safety.

The proposed TS changes associated with the relocation of the boration subsystem and RHR System overpressure protection requirements to a licensee-controlled document, pressurizer code safety valve requirements, and isolated RCS loop startup will not result in a significant reduction in a margin of safety.

The additional proposed changes to the TSs that will standardize terminology, relocate information to the Bases, remove extraneous information, modify requirements to prevent rod withdrawal for operational flexibility, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional changes will not result in a significant reduction in a margin of safety.

Based on the NRC staff's analysis, 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina, Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: March 22, 2001.

Description of amendment request: The proposed amendments would revise the Catawba Nuclear Station (CNS) Unit 1 and Unit 2, and the McGuire Nuclear Station (MNS) Unit 1 and Unit 2 Technical Specifications (TS). The proposed TS revisions are presented in two parts. Part I affects the current MNS and CNS TS surveillance requirement (SR), and associated TS Bases for the methodology and frequency for the chemical analyses of the ice condenser ice bed (stored ice). The revision results in renumbering the SRs. Also, this proposed amendment adds a new TS SR to address sampling requirements for ice additions to the ice bed. Part II proposes revisions to the current MNS and CNS TS surveillance requirement (SR) acceptance criteria and surveillance frequency for the inspection of ice condenser ice basket flow channel areas. The proposed change also results in renumbering the SRs. Associated changes are also made to the TS Bases.

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

A. The Proposed Change Does Not Involve A Significant Increase In The Probability Or Consequences Of An Accident Previously Evaluated.

The only analyzed accidents of possible consideration in regards to changes potentially affecting the ice condenser are a loss of coolant accident (LOCA) and a High Energy Line Break (HELB) inside containment. However, the ice condenser is not postulated as being the initiator of any LOCA or HELB. This is because it is designed to remain functional following a design basis earthquake, and the ice condenser does not interconnect or interact with any systems that interconnect or interact with the Reactor Coolant or Main Steam Systems. The proposed changes to the TSs and associated TS Bases are solely to revise and provide clarification of the ice sampling and chemical analysis requirements, and flow area verification requirements. Since these proposed changes do not result in, or require, any physical change to the ice condenser, then there can be no change in the probability of an accident previously evaluated in the SAR.

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 HELB, or in the chemical composition of the stored ice.

The proposed changes add an upper limit on boron concentration, which is the bounding value for the boron precipitation analysis. The upper limit boron concentration is an existing DBA analysis input limit that is controlled by existing procedure. Therefore, the addition of a TS requirement for an upper limit on boron concentration does not affect the physical operation or condition of the ice condenser.

Though the frequency of the existing surveillance requirement for sampling the stored ice is changed from once every 18 months to once every 54 months, the sampling requirements are strengthened overall with the requirement to obtain one randomly selected sample from each ice condenser bay (24 total samples) rather than nine "representative" samples, and the addition of a new surveillance requirement to verify each addition of ice meets the existing requirements for boron concentration and pH value.

The proposed changes clarify that each sample of stored ice is individually analyzed for boron concentration and pH, and 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 Condition, provided the averaged results are acceptable.

The proposed changes revise and clarify the flow area verification requirements. Regarding the consequences of analyzed accidents, the ice condenser is an engineered safety feature designed, in part, to limit the containment subcompartment and steel vessel pressures immediately following the initiation of a LOCA or HELB. Conservative sub-compartment pressure analysis shows this criteria will be met if the reduction in the flow area per bay provided for ice condenser air/steam flow paths is ≤15 percent, or if the total flow area blocked within each lumped analysis section is ≤15 percent as assumed in the safety analysis. The present 0.38 inch frost/ice buildup surveillance criteria only addresses the acceptability of any given flow path, and has no existing correlation between flow paths exceeding this criteria and percent of total flow path blockage. In fact, it was never the intent of the current surveillance requirement (SR) to make such a correlation. If problems were encountered in meeting the 0.38 inch criteria, it was expected that additional inspection and analysis, such as provided in the proposed amendment, would be performed to make such a determination. Thus, the proposed amendment for flow blockage determination provides the necessary assurance that flow path requirements are met without additional evaluations.

The proposed amendment also revises the flow area verification surveillance frequency from every 9 months to every 18 months such that it will coincide with refueling outages. Management of ice condenser maintenance activities has successfully limited activities with the potential for significant flow channel degradation to the refueling outage. By verifying an ice bed condition of less than or equal to 15% flow channel blockage following completion of these maintenance activities, the surveillance assures the ice bed is in an acceptable condition for the duration of the operating cycle. During the operating cycle, an expected amount of ice sublimates and reforms as frost on the colder surfaces in the Ice Condenser. However, frost does not degrade flow channel flow area per the Westinghouse definition of frost. The surveillance will effectively demonstrate operability for an allowed 18 month cycle. Therefore, increasing the surveillance frequency does not affect the ice condenser operation or accident response. An ice bed condition of less than or equal to 15% flow channel blockage is assured to be maintained for the operating cycle to address the limiting design basis accident(s) (DBAs).

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 Change Does Not Create The Possibility Of A New Or Different Kind Of Accident From Any Accident Previously Evaluated[.]

Because the TSs 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 TSs, there can be no new accidents created from those already identified and evaluated.

C. The Proposed Change Does Not Involve A Significant Reduction In A Margin Of Safety[.]

The ice condenser Technical Specifications ensure that during a LOCA or HELB 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 analyses.

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 boron precipitation analysis. The upper limit boron concentration is an existing design bases limit that is controlled by existing procedure. Therefore, the addition of a TS requirement for an upper limit on boron concentration does not affect the physical operation or condition of the ice condenser.

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 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 (one 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.

Design Basis Accident analyses have shown that with 85 percent of the total flow area available (uniformly distributed), the ice condenser will perform its intended function. Thus, the safety limit for ice condenser operability is a maximum 15 percent blockage of flow channels. The existing TS surveillance requirement currently uses a specific value of 0.38 inch buildup to determine if unacceptable frost/ice blockage exists in the ice condenser. However, this specific value does not have a direct correlation to the safety limit for blockage of ice condenser flow area. The proposed TS amendment requires more extensive visual inspection (33 percent of the flow area/bay) than is currently described (2 flow channels/bay) in the TS Bases, thus providing greater reliability and a direct relationship to the analytical safety limits. Changing the TS to implement a surveillance program that is more reliable and uses acceptance criteria of less than or equal to 15 percent flow blockage, as allowed by the TMD code analysis, will not reduce the margin of safety of any TS.

The proposed amendment also revises the surveillance frequency of flow area inspection from every 9 months to every 18 months such that it will coincide with refueling outages. Management of ice condenser maintenance activities has successfully limited activities with the potential for significant flow channel degradation to the refueling outage. By verifying an ice bed condition of less than or equal to 15% flow channel blockage following completion of these maintenance activities, the surveillance assures the ice bed is in an acceptable condition for the duration of the operating cycle. During the operating cycle, an expected amount of ice sublimates and reforms as frost on the colder surfaces in the Ice Condenser. However, frost has been determined to not degrade flow channel flow area. Thus, design limits for the continued safe function of containment sub-compartment walls and the steel containment vessel are not exceeded due to this change.

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 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: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard Emch.

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

Date of amendment request: June 12, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.8.1.1.2.a and delete the table referenced in TS 4.8.1.1.2.a, to remove the requirement for an accelerated test frequency for the emergency diesel generators (EDGs); delete TS 4.8.1.1.2.c.1 to remove the requirement to subject the EDG to an inspection in accordance with the manufacturer's recommendations; revise TSs 4.8.1.1.2.c.9, 10 and 13 to allow that EDG surveillances regarding the 24-hour endurance run, the auto-connected loads not exceeding the 2-hour rating, and the fuel transfer pump transferring fuel via the cross-connect lines, are conducted during modes other than during shutdown; and delete TSs 4.8.1.1.3 and 6.9.1.5.d to remove the EDG special reporting requirements.

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

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

There are no previously evaluated accidents associated with these surveillance activities. The EDGs are not accident initiators. The EDGs provide assistance in accident mitigation. There are no technical changes related to the acceptance criteria of any of these surveillances nor are there any physical changes to plant design proposed in this amendment request. The proposed change, requesting that the frequency and scheduling aspects of the surveillance requirements be changed to accommodate improved planning capability for testing and maintenance activities, does not affect the accident analyses. Additionally, the allowance to perform testing and maintenance activities on line will improve EDG availability during periods of shutdown operations.

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

2. Will 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?

The proposed change does not include any physical changes to plant design or a change to any of the current SR [surveillance requirement] acceptance criteria. Performance of any of these surveillance activities while at power does not render the EDGs unavailable in that they can provide station power on demand. Performance of maintenance activities and surveillance requirements while on line, which could result in the equipment being out of service, was included in the development of the Limiting Conditions for Operation (LCO). Quantitative and qualitative evaluations relative to the credit allowed for redundant components and the time allowed for corrective actions were also considered in LCO development. Performance of these activities while on line does not create any new or different kinds of accident. The capability of the EDG to respond to an accident situation while tied to the grid during testing activities is tested as required by existing surveillance requirements. These tests ensure that if tied to the gird [grid] the EDG output breaker will open and the EDG [will] remain running in standby until an under voltage condition is observed, at which time the EDG will automatically tie on to the 4160 V [volt] ESF [engineered safety features] bus.

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

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

The proposed changes are associated with surveillance requirements for the EDGs. The deletion of accelerated testing requirements provides an enhancement to safety by eliminating unnecessary testing. The remaining proposed changes allow certain EDG surveillance requirements to be performed when the plant is at power rather than when shutdown. The operation of, and requirements for, the equipment covered by the affected TSs will remain essentially the same.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 12, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.8.1.1 to provide a one-time extension of the allowed outage time (AOT) for an inoperable emergency diesel generator (EDG) from three days to ten days, provided the alternate alternating-current diesel generator (AACDG) is available. In addition, the proposed amendment would revise TS 3.4.4 to make the action associated with an inoperable emergency power supply to the pressurizer heaters consistent with the proposed EDG AOT.

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

Typically, only one EDG is OOS [out of service] for maintenance activities at any given time. The standby EDG is aligned as required by Technical Specifications (TS) and available for auto start upon demand. Additionally, the AACDG is verified available and capable of being aligned to the Engineered Safety Features (ESF) electrical buses associated with the OOS EDG. The AACDG is sized such that it can carry the loads equivalent to at least one train of Engineered Safety Features (ESF) equipment. In the event of a loss of offsite power while an EDG is OOS, the AACDG will be manually started and loaded. The time delay associated with the manual start of the AACDG will result in a minimal change in the overall risk associated with the ability to reestablish power to ESF equipment upon a loss of offsite power. However, assuming the standby EDG operates as designed, it will start upon receipt of the automatic start signal and sequence on loads as required.

The plant can be maintained in a safe configuration or mitigate any accident situations with only one train of ESF components. Reliance upon the AACDG to provide a backup function ensures a minimal change in risk associated with extending the EDG AOT. The EDG AOT of 72 hours under the existing technical specifications does not consider an additional backup power supply to be available to mitigate a loss of offsite power. The proposed change will ensure that an alternate onsite diesel generator will be available while the EDG is out of service. Therefore, this change is considered a more responsive action than that contained in the current TSs.

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

2. Will 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?

The duration of an AOT is determined considering that there is a minimal possibility that an accident will occur while a component is removed from service. Typically, only the single redundant train is available during the AOT with no backup components available to supply the function of the component. The proposed change allows the EDG AOT to be extended one time for each EDG to 10 days with reliance on the AACDG. If the AACDG is not available, the AOT is 72 hours. No new modifications are required to allow the AACDG to function.

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

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

The EDG AOT will be 10 days for each EDG if the AACDG is available. However, if the AACDG is not available, the EDG AOT will remain at 72 hours. The AACDG supplies backup power to the redundant train of ESF components. The standby EDG, which is typically not aligned in a test mode during the AOT, will be available to automatically start and sequence on loads upon demand. Two trains of ESF components powered from the onsite electrical sources, the standby EDG and the AACDG, will be available in the event of an accident. When the AACDG is not available, the current 72-hour AOT will begin. In conclusion, either the AACDG will be available, which will result in two ESF trains being available in the unlikely event of an accident, or the current AOT will apply.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Exelon Generation Company, LLC, PSEG Nuclear LLC, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: May 30, 2001.

Description of amendment request: The proposed amendment would allow an extension to the interval for integrated leak rate tests (ILRTs) of the reactor containment building. The change involves a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J.'' The current 10-year containment building ILRT for Peach Bottom Atomic Power Station (PBAPS), Unit 3, is due in December 2001 and is currently scheduled to be performed during Refueling Outage 3R13 in October 2001. The proposed exception would allow the next ILRT for PBAPS, Unit 3, to be performed within 16 years (December 2007) from the last ILRT as opposed to the current 10-year frequency.

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

The proposed revision to Technical Specification 5.5.12 (''Primary Containment Leakage Rate Testing Program'') involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than sixteen (16) years from the last Type A test. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. PBAPS, Unit 3 ILRT test history supports this conclusion. NUREG–1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and licensing commitments related to containment coatings serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Primary Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

PBAPS, Unit 3 and industry experience strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the Coatings Program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR containments allows for the detection of gross containment leakage that may develop during power operation. The combination of these factors ensures that the margin of safety that is inherent in plant safety analysis is maintained. Therefore, the proposed Technical Specification 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: Mr. Edward Cullen, Vice President and General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 13, 2001.

Description of amendment request: The proposed amendment clarifies the Kewaunee Nuclear Power Plant Technical Specification 5.3 to permit lead-test-assemblies to be used, regardless of clad material, as long as the Nuclear Regulatory Commission has generically approved the fuel assembly design for use in pressurized water reactors.

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.

Changing the technical specification within limits of the bounding accident analyses cannot change the probability of an accident previously evaluated, nor will it increase radiological consequence predicted by the analyses of record. Controlling the use of lead-test-assemblies, designs of which were approved by the NRC, according to limitations approved by the NRC constrains fuel performance within limits bounded by existing design basis accident and transient analyses. Thus, nothing in this proposal will cause an increase in the probability or consequence of an accident previously evaluated.

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

Inclusion in the reactor core of lead-test-assemblies according to limitations set by the NRC and of a design approved by the NRC ensures that their effect on core performance remains within existing design limits. Use of NRC approved fuel assemblies as lead-test-assemblies is consistent with current plant design bases, does not adversely affect any fission product barrier, and does not alter the safety function of safety significant systems, structures and components or their roles in accident prevention or mitigation. Currently licensed design basis accident and transient analyses of record bound the effect of lead-test-assemblies. Thus, this proposal does not create the possibility of a new or different kind of accident.

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

The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Setpoints, or Limiting Conditions for Operation are determined. This clarification of TS 5.3 is bounded by existing limits on reactor operation. It leaves current limitations for use of lead-test-assemblies in place, conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation within analyzed and licensed boundaries. Thus, changes proposed by this request 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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: Claudia M. Craig.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: November 16, 2000.

Description of amendment request: The proposed change would delete a note to Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.1.1, which permitted a temporary extension to the surveillance interval for testing spectacle flanges 2S299A and 2S299B.

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 change to Technical Specification SR 3.6.1.1.1 is administrative in nature. The note is no longer required as the condition has been corrected and the SR performed with acceptable results. Removal of the note restores the Technical Specification to its original condition and therefore, this proposed amendment does not involve any increase in the probability of occurrence 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 note is no longer required as the condition has been corrected and the SR performed with acceptable results. Removal of the note by this change restores the Technical Specification to its original condition and therefore does not create any possibility of a new or different kind of accident from those previously analyzed.

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

The note is no longer required as the condition has been corrected and the SR performed with acceptable results. Therefore, removal of the note by this change restores the Technical Specification to its original condition and does not involve a reduction in any margin of safety.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Acting Section Chief: Richard Correia.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: March 5, 2001.

Description of amendment request: The proposed amendment would: (1) change the Security Plan provision that a member of the security force escort all vehicles, other than designated licensee vehicles, and delete the related Security Training and Qualification Plan task; (2) change the requirement of the Security Plan that all areas of the protected area be illuminated to a minimum of 0.2 footcandle; and (3) change the frequency of protected area patrols in the Security Plan.

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

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

The proposed changes involving security activities do not reduce the ability for the security organization to prevent radiological sabotage and therefore do not increase the probability or consequences of a radiological release previously evaluated.

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

The proposed changes involve functions of the security organization concerning vehicle control, protected area illumination, and protected area patrol frequency. Analysis of the proposed changes has not indicated nor identified 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?

Analysis of the proposed changes show that they affect only the functions of the Security organization and have no impact upon nor cause a significant reduction in margin of safety for plant operation. The failure points of key safety parameters are not affected.

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of amendment request: May 21, 2001.

Description of amendment request: The proposed amendment will delete the definitions, the limiting conditions for operation, and the surveillance requirements and revise the design features and administrative controls to reflect the transfer of all the spent nuclear fuel from the 10 CFR Part 50 licensed site to the 10 CFR Part 72 licensed Independent Spent Fuel Storage Installation (ISFSI) from the Rancho Seco Technical Specifications.

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

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

No. The proposed changes reflect removing the spent nuclear [fuel] from the 10 CFR Part 50 licensed facility and transferring the fuel to a 10 CFR Part 72 licensed facility. The design basis accidents analyzed in the Rancho Seco Defueled Safety Analysis Report (DSAR) include the fuel handling accident and a loss of offsite power (LOOP). The fuel handling accident is the worst-case design basis accident postulated to occur at Rancho Seco. Both of these accidents are based on spent nuclear fuel being stored in the spent fuel pool at the 10 CFR Part 50 licensed facility.

With the removal of the spent nuclear fuel from the 10 CFR Part 50 licensed facility, there are no remaining important to safety systems required to be monitored and there are no remaining credible accidents that require the actions of a Certified Fuel Handler or Non-Certified Fuel Handler to prevent occurrence or mitigate the consequences.

DSAR Section 14.2 provides a discussion of accidents during decommissioning. The DSAR concludes that the consequences of the accidents evaluated in NUREG/CR–0130 ''Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station'' bound the potential accidents that could occur during decommissioning at Rancho Seco. The proposed Technical Specification changes have no impact on decommissioning activities.

The proposed Technical Specification Section D5.2 precludes the storage of spent nuclear fuel at the 10 CFR Part 50 licensed facility. The probability or consequences of accidents at the ISFSI are evaluated in the ISFSI FSAR [Final Safety Analysis Report] and are independent of the 10 CFR Part 50 license.

Therefore, with all of the spent fuel stored at the Rancho Seco ISFSI, the accidents evaluated in the DSAR are no longer relevant, and the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes reflect the reduced operational risks within the 10 CFR Part 50 licensed facility after the fuel is transferred to the 10 CFR Part 72 licensed ISFSI. The proposed changes do not result in physical changes to the 10 CFR Part 50 facility and the plant conditions for which the design basis accidents have been evaluated are no longer applicable.

No new failure modes are introduced as the result of the proposed changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. As described above, the proposed changes reflect the reduced operational risks within the 10 CFR Part 50 licensed facility after the fuel is transferred to the ISFSI. The design basis and the accident assumptions in the Defueled Safety Analysis Report (DSAR), and the Technical Specification Bases are no longer applicable after the fuel is permanently removed from the 10 CFR Part 50 licensed facility. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Thomas A. Baxter, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N. Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of amendment request: June 7, 2001.

Description of amendment request: The proposed amendment will delete certain administrative requirements from the Rancho Seco Technical Specifications and relocate other administrative requirements from the Rancho Seco Technical Specifications to the Rancho Seco Quality Manual following the transfer of all the spent nuclear fuel from the 10 CFR part 50 licensed site to the 10 CFR Part 72 licensed Independent Spent Fuel Storage Installation (ISFSI).

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

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

No. The proposed changes are administrative and involve deleting certain administrative requirements from the Technical Specifications. Some administrative requirements are no longer applicable after permanently transferring the spent nuclear fuel from the 10 CFR 50 licensed facility to the 10 CFR 72 licensed ISFSI. Other administrative requirements are being relocated to the NRC-approved Rancho Seco Quality Manual (RSQM).

Relocating administrative requirements to the NRC-approved RSQM is consistent with the guidance in NRC Administrative Letter 95–06. Relocating these administrative requirements will not alter the configuration or operation of the facility, and therefore does not involve a significant increase in the probability or consequences of an accident previously evaluated.

In addition, deleting certain administrative requirements (i.e., PRC [Plant Review Committee] and MSRC [Management Safety Review Committee]) is based on permanently removing the spent nuclear fuel from the 10 CFR Part 50 licensed facility and transferring the fuel to a 10 CFR Part 72 licensed facility.

The design basis accidents analyzed in the Rancho Seco Defueled Safety Analysis Report (DSAR) include the fuel handling accident and a loss of offsite power. Both of these accidents are based on spent nuclear fuel being stored in the spent fuel pool at the 10 CFR Part 50 licensed facility.

With all of the spent fuel stored at the Rancho Seco ISFSI, the accidents evaluated in the DSAR are no longer relevant. Therefore, the proposed license amendment does not involve any increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes are administrative and reflect the reduced operational risks within the 10 CFR Part 50 licensed facility after the fuel is transferred to the 10 CFR Part 72 licensed ISFSI. The proposed changes do not result in physical changes to the 10 CFR Part 50 facility, and the plant conditions for which the design basis accidents have been evaluated are no longer applicable.

No new failure modes are introduced as the result of the proposed changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. As described above, the proposed changes are administrative and reflect the reduced operational risks within the 10 CFR Part 50 licensed facility after the fuel is transferred to the ISFSI. The design basis and the accident assumptions in the DSAR and the Technical Specification Bases are no longer applicable after the fuel is permanently removed from the 10 CFR Part 50 licensed facility. Therefore, the proposed changes do not involve any reduction in a margin of safety.

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

Attorney for licensee: Thomas A. Baxter, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N. Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 24, 2001.

Description of amendment request: The proposed change would relocate Technical Specification 3/4.9.6, ''Refueling Machine'' to the Technical Requirements Manual consistent with NUREG–1431, ''Standard (Improved) Technical Specifications—Westinghouse Plants.''

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

Pursuant to 10 CFR 50.92, it has been determined that this proposed amendment involves no significant hazards consideration. This determination was made by applying the Nuclear Regulatory Commission established standards contained in 10 CFR 50.92. These standards assure that operation of South Texas Project in accordance with this request consider the following:

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

Response: No.

This request involves an administrative change only. No actual plant equipment or accident analyses will be affected by the proposed changes. Operability of the refueling machine ensures that the equipment used to handle fuel within the reactor vessel has sufficient load capacity for handling fuel assemblies and/or control rods. Although the refueling machine is designed and has interlocks that can prevent damage to the fuel assemblies, the equipment is not assumed to function or actuate to mitigate the consequences of a design basis accident or transient in the safety analysis. Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This request involves an administrative change only. The proposed change does not alter the performance of the refueling machine and auxiliary hoist or the manner in which the equipment will be operated. The refueling equipment will still be tested before placing the equipment into operational service. Changing the location of these requirements and surveillances from Technical Specifications to the Technical Requirements Manual [TRM] will not create any new accident initiators or scenarios. Since the proposed changes only allow activities that are presently approved and conducted, no possibility exists for a new or different kind of accident from those previously evaluated.

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

Response: No.

This request involves administrative changes only. 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 systems settings, or will not relax the bases for any limiting conditions of operation. Therefore, the proposed changes will not impact the margin of safety.

Conclusion

Based on the above analysis, STPNOC concludes that the proposed amendment to relocate these requirements from Technical Specifications to the TRM involve no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 24, 2001.

Description of amendment request: Revise the Technical Specification definition for CORE ALTERATIONS so that moving the control rods with the integrated head package would not be a core alteration.

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:

STPNOC [South Texas Project Nuclear Operating Company] has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92, ''Issuance of amendment,'' as discussed below.

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

Response: No.

The proposed change in the definition of CORE ALTERATIONS will not alter the way STPNOC handles the integrated head package. No new accident initiators will be introduced. Consequently, there is no significant increase in the probability of an accident previously evaluated.

The evaluation demonstrates that the RCCAs [rod cluster control assemblies] have no effect on reactivity when they are withdrawn into the integrated head package. The proposed change has no effect on assumptions made in any accident previously evaluated. Consequently, there are no significant increases 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 accident previously evaluated?

Response: No.

The proposed change does not involve any new processes, procedures, or significantly different plant configurations. No new reactivity configurations are presented. Consequently, the possibility of a new or different kind of accident is not created.

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

Response: No.

The evaluation shows the RCCAs have no effect on reactivity when they are withdrawn into the integrated head package. Moving the integrated head package with the RCCAs withdrawn provides the same degree of control on reactivity as the original definition. Consequently, the proposed change does not involve a significant reduction in the margin of safety.

Conclusion

Based upon the analysis provided herein, the proposed amendments will not increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a reduction in a margin of safety. Therefore, the proposed amendments meet the requirements of 10 CFR 50.92 and do not involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to pdr@nrc.gov.

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 application for amendments: October 20, 2000, as supplemented by letter dated March 12, 2001.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 3.7.10, ''Control Room Area Ventilation System (CRAVS)'' by eliminating the requirement for the CRAVS high chlorine protection function. The amendments also eliminated the requirement for the safety related chlorine monitor and the capability for automatic isolation of the control room area ventilation system when prompted by a signal from the detectors. Revisions to the corresponding Bases for TS 3.7.10 have been incorporated.

Date of issuance: June 28, 2001.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance June 28, 2001.

Amendment Nos.: 191 and 183.

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

Date of initial notice in Federal Register: January 10, 2001 (66 FR 2013). The supplement dated March 12, 2001, provided clarifying information that did not change the scope of the October 20, 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 June 28, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 19, 2001.

Brief description of amendment: The amendment modifies Technical Specification 3.6.5, ''Vacuum Relief Valves,'' Limiting Condition for Operation, and extends the allowed outage time from 4 hours to 72 hours to restore the vacuum relief line to OPERABLE status. In addition, Attachment 1 to the Waterford Steam Electric Station, Unit 3 Operating License has been deleted and paragraph 2.C.1 revised to reflect the deletion.

Date of issuance: June 18, 2001.

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

Amendment No.: 171.

Facility Operating License No. NPF–38: The amendment revised the Operating License and Technical Specifications.

Date of initial notice in Federal Register: 66 FR 27176, dated May 16, 2001.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 18, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 28, 2000, as supplemented June 12, 2001.

Brief description of amendment: This amendment revises the design basis for the post-trip steam line break analysis to allow less than or equal to 2% fuel failure.

Date of Issuance: June 19, 2001.

Effective Date: June 19, 2001.

Amendment No.: 116.

Facility Operating License No. NPF–16: Amendment does not revise the operating license or its appendices.

Date of initial notice in Federal Register: February 7, 2001 (66 FR 9384). The June 12, 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 June 19, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: January 30, 2001.

Brief description of amendment: The amendment changes two requirements in the Operating License regarding the reporting of changes to the approved fire protection plan and exceeding the licensed steady-state power level.

Date of issuance: June 26, 2001.

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

Amendment No.: 203

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

Date of initial notice in Federal Register: April 4, 2001 (66 FR 17965). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 2, 2001.

Brief description of amendment: The proposed amendment revises Technical Specifications (TS) Section 5.6, ''TS Bases Control Program,'' to delete the term ''unreviewed safety question'' consistent with the recent revision to 10 CFR 50.59. The TS, as amended, would continue to incorporate the criteria of 10 CFR 50.59 by reference.

Date of issuance: June 15, 2001.

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

Amendment No.: 32.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: April 4, 2001 (66 FR 17971).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 15, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland this 2nd day of July 2001.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 01–17223 Filed 7–10–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued revisions of two guides in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

Revision 3 of Regulatory Guide 1.52, ''Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,'' describes methods acceptable to the NRC staff for complying with the NRC's regulations with regard to the design, inspection, and testing criteria for air filtration and iodine adsorption units of engineered-safety-feature atmosphere cleanup systems in light-water-cooled nuclear power plants. This guide applies only to post-accident atmosphere cleanup systems that are designed to mitigate the consequences of postulated accidents.

Revision 2 of Regulatory Guide 1.140, ''Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,'' describes methods acceptable to the NRC staff for complying with the NRC's regulations with regard to the criteria for air filtration and adsorption units installed in the normal ventilation exhaust systems of light-water-cooled nuclear power plants.

Comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time. Written comments may be submitted to the Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Regulatory guides are available for inspection or downloading at the NRC's web site at <*WWW.NRC.GOV*> under Regulatory Guides and in NRC's Electronic Reading Room (ADAMS System) at the same site; Revision 3 of Regulatory Guide 1.52 is under ADAMS Accession Number ML011710176; Revision 2 of Regulatory Guide 1.140 is under ADAMS Accession Number ML011710150. Single copies of regulatory guides may be obtained free of charge by writing the Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by fax to (301)415–2289, or by email to <*DISTRIBUTION@NRC.GOV*>. Issued guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

(5 U.S.C. 552(a))

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

For the Nuclear Regulatory Commission.

Michael E. Mayfield,

Director, Division of Engineering Technology, Office of Nuclear Regulatory Research.

[FR Doc. 01–17349 Filed 7–10–01; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

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

Extension:

Rule 15Bc3–1 and Form MSDW, SEC File No. 270–93, OMB Control No. 3235–0087

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget (''OMB'') a request for approval of extension of the previously approved collection of information discussed below.

Rule 15Bc3–1 under the Securities Exchange Act of 1934 provides that a notice of withdrawal from registration with the Commission as a bank municipal securities dealer must be filed on Form MSDW.

The Commission uses the information submitted on Form MSDW in determining whether it is in the public interest to permit a bank municipal securities dealer to withdraw its registration. This information is also important to the municipal securities dealer's customers and to the public, because it provides, among other things, the name and address of a person to contact regarding any of the municipal securities dealer's unfinished business.

The staff estimates that approximately 20 respondents will utilize this notice annually, with a total burden for all respondents of 10 hours, based upon past submissions. The staff estimates that the average number of hours necessary to comply with the requirements of Rule 15Bc3–1 is .5 hours. The average cost per hour is approximately \$101. Therefore, the total cost of compliance for the respondents is \$1,010 (\$101 × 5 × 20 = \$1,010).

Providing the information on the notice is mandatory in order to withdraw from registration with the Commission as a bank municipal securities dealer. The information contained in the notice will not be confidential. An agency may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid control number.

General comments regarding the above information should be directed to the following persons: (i) Desk Officer for the Securities and Exchange Commission, Office of Information and Regulatory Affairs, Office of Management and Budget, Room 10102, New Executive Office Building, Washington, D.C. 20503; and (ii) Michael E. Bartell, Associate Executive Director, Office of Information Technology, Securities and Exchange Commission, 450 Fifth Street, N.W., Washington, D.C. 20549. Comments must be submitted to OMB within 30 days of this notice.

Dated: July 3, 2001.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 01–17266 Filed 7–10–01; 8:45 am]

BILLING CODE 8010–01–M