December 28, 2005

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 — ISSUANCE OF AMENDMENTS REGARDING RELOCATION OF MULTIPLE TECHNICAL SPECIFICATIONS TO THE TECHNICAL REQUIREMENTS MANUAL (TAC NOS. MC6881 AND MC6882) (TS 04-06)

Dear Mr. Singer:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 305 to Facility Operating License No. DPR-77 and Amendment No. 295 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated April 27, 2005 (TVA-SQN-TS-04-06), and as supplemented by letter dated November 17, 2005.

The amendments relocate several Technical Specification (TS) requirements to the Technical Requirements Manual (TRM). Specifically, the amendments relocate the provisions for TS 3.3.2 (Movable Incore Detectors), TS 3.3.3.4 (Meteorological Instrumentation), TS 3.4.7 (Reactor Coolant System Chemistry), TS 3.4.11 (Reactor Coolant System Head Vents), TS 3.7.2 (Steam Generator Pressure and Temperature Limitations), TS 3.7.10 (Sealed Source Contamination), TS 3.9.5 (Refueling Operations Communications), and TS 3.9.6 (Manipulator Crane) to the TRM. These changes are consistent with the latest version of NUREG-1431, Revision 3, “Standard Technical Specifications for Westinghouse Plants,” and do not diminish the level of safety found in the current TSs.

A copy of the staff's Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 305 to License No. DPR-77
2. Amendment No. 295 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

December 28, 2005

A copy of the staff's Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

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Douglas V. Pickett, Senior Project Manager
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cc w/enclosures: See next page

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.305
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

    A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 27, 2005, as supplemented by letter dated November 17, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

    C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 305, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Jr., Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 28, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 305

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Index Page V	Index Page V
Index Page VI	Index Page VI
Index Page VIII	Index Page VIII
Index Page IX	Index Page IX
Index Page X	Index Page X
Index Page XIII	Index Page XIII
Index Page XIV	Index Page XIV
3/4 3-43	3/4 3-43
3/4 3-47	3/4 3-47
3/4 3-48	3/4 3-48
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3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
3/4 4-18	3/4 4-18
3/4 4-28	3/4 4-28
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3/4 7-30	3/4 7-30
3/4 9-5	3/4 9-5
3/4 9-6	3/4 9-6

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.295
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 27, 2005, as supplemented by letter dated November 17, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 305 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Jr., Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 28, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 295

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Index Page V	Index Page V
Index Page VI	Index Page VI
Index Page VIII	Index Page VIII
Index Page IX	Index Page IX
Index Page X	Index Page X
Index Page XIII	Index Page XIII
Index Page XIV	Index Page XIV
3/4 3-44	3/4 3-44
3/4 3-48	3/4 3-48
3/4 3-49	3/4 3-49
3/4 3-50	3/4 3-50
3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
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3/4 7-42	3/4 7-42
3/4 9-6	3/4 9-6
3/4 9-7	3/4 9-7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 305 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 295 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated April 27, 2005 (ADAMS accession number ML051310280), as supplemented by letter dated November 17, 2005 (ML053410272), the Tennessee Valley Authority (TVA, the licensee) proposed amendments to the Technical Specifications (TSs) for Sequoyah Nuclear Plant (SQN), Units 1 and 2. The requested changes in the April 27, 2005, application would relocate several TS requirements to the Technical Requirements Manual (TRM). Specifically, the proposed amendments would relocate the provisions for TS 3.1.3.4 (Rod Drop Time), TS 3.3.2 (Movable Incore Detectors), TS 3.3.3.4 (Meteorological Instrumentation), TS 3.4.7 (Reactor Coolant System Chemistry), TS 3.4.11 (Reactor Coolant System Head Vents), TS 3.7.2 (Steam Generator Pressure and Temperature Limitations), TS 3.7.10 (Sealed Source Contamination), TS 3.9.5 (Refueling Operations Communications), and TS 3.9.6 (Manipulator Crane) to the TRM.

In the supplemental letter dated November 17, 2005, the licensee withdrew its request to relocate TS 3.1.3.4 related to Rod Drop Time, which will now be retained without change. The November 17, 2005, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The remaining requirements do not meet the four criteria found in Title 10, Part 50, Section 50.36 of the *Code of Federal Regulations* (10 CFR 50.36) and can be removed from the TSs. The current specifications will be moved without changes to the TRM, which is an appropriate owner-controlled document consistent with the safety significance for these functions. As part of the TRM, any subsequent changes to these specifications will be performed in accordance with the regulations of 10 CFR 50.59. These changes are consistent with the latest version of NUREG-1431, Revision 3, “Standard Technical Specifications for Westinghouse Plants,” and do not diminish the level of safety found in the current TSs.

2.0 REGULATORY EVALUATION

Section 182a of the *Atomic Energy Act* (the Act) requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of structures, systems, and components that are required to protect the health and safety of the public. The staff of the Nuclear Regulatory Commission

(NRC) used the regulatory requirements for TS changes set forth in 10 CFR 50.36 for this evaluation. Specifically, 10 CFR 50.36©)(1) specifies safety limits, limiting safety systems settings and control settings, 10 CFR 50.36©)(2) specifies the requirements for limiting conditions for operation, 10 CFR 50.36©)(3) specifies the surveillance requirements, 10 CFR 50.36©)(4) specifies the design requirements, and 10 CFR 50.36©)(5) specifies the administrative controls.

Originally, the requirements of 10 CFR 50.36 established the categories of items for inclusion in the TSs, but not the particular requirements for the TSs of each individual plant. The NRC provided guidance for the specific contents of the TSs in the “Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors” (Final Policy Statement), 58 FR 39132 (July 22, 1993). In particular, the NRC indicated that certain items could be relocated from the TSs to licensee-controlled documents. The Final Policy Statement established four criteria for determining the items required for inclusion in the TSs:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 4 A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

These criteria have been codified in 10 CFR 50.36(c)(2)(ii). See Final Rule, “Technical Specifications,” 60 FR 36593 (July 19, 1995). As a result, TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TSs.

The Final Policy Statement allows the relocation of items not meeting these four specified criteria from the TSs to licensee-controlled documents, such that future changes can be made to these provisions pursuant to 10 CFR 50.59. The NRC also concluded that compliance with the Final Policy Statement satisfied Section 182a of the Act, precipitating a revision to 10 CFR 50.36 which superseded the Final Policy Statement.

The proposed revision to the SQN TSs for Units 1 and 2 will relocate Limiting Conditions for Operation and associated Action and Surveillance Requirements (SRs) that are not required to be contained in the TSs in accordance with 10 CFR 50.36(c)(2)(ii).

In general, there are two classes of changes to TSs: (1) Changes needed to reflect modifications to the design basis (TSs are derived from the design basis), and (2) voluntary changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TSs over time. This amendment deals with only the second class of changes. In determining the acceptability of such changes, the NRC staff interprets the requirements of the current version of 10 CFR 50.36, using as a model the accumulation of generically approved guidance in the improved standard TS (STS) NUREGs.

Within this general framework, licensees may remove material from their TSs on two conditions: (1) the material is not required to be in the TSs based on the staff interpretation of 10 CFR 50.36, including judgments about the level of detail required in the TSs, and (2) there exist suitable alternative regulatory controls for the material. Licensees may revise the remaining TSs to adopt current improved STS format and content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative or provides clarification (i.e., no requirements are materially altered), (2) the change is more restrictive than the licensee's current requirement, or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, are discussed in Section 3.0 in the context of specific proposed changes.

The specifications proposed for amendment include TS 3.3.2 for Movable Incore Detectors, TS 3.3.3.4 for Meteorological Instrumentation, TS 3.4.7 for Reactor Coolant System Chemistry, TS 3.4.11 for Reactor Coolant System Head Vents, TS 3.7.2 for Steam Generator Pressure and Temperature Limitations, TS 3.7.10 for Sealed Source Contamination, TS 3.9.5 for Refueling Operations Communications, and TS 3.9.6 for the Manipulator Crane. These specifications would be relocated in their entirety to the TRM without change to the requirements currently contained within the TSs. The Bases associated with these specifications would also be relocated to the TRM to support the proposed revision. As part of the relocation, associated SRs would be modified to remove references to the subject TSs. Necessary changes to the index pages and the Bases pages would be included to denote the deletion of these specifications from the TSs.

The relocation of the above TSs and Bases will place them in the TRM, which is a 10 CFR 50.59 controlled document that provides an appropriate level of review and approval for the revision of requirements that are important to safety, but do not satisfy the criteria of 10 CFR 50.36(c)(2)(ii) for TS requirements. Changes to the TRM requirements are subject to the regulations of 10 CFR 50.59 because TVA has incorporated the TRM into the SQN Updated Final Safety Analysis Report. The proposed revision will maintain an appropriate level of control of the relocated requirements and an improved level of consistency with NUREG-1431, "Standard TSs for Westinghouse Plants," Revision 3, which does not contain requirements for the TSs proposed for relocation. Based on the current requirements for the TSs proposed for relocation being retained without change, other regulatory requirements and criteria will continue to be satisfied. The application of these requirements as part of the TRM will be identical to those applicable to TSs.

3.0 TECHNICAL EVALUATION

The proposed change will relocate the identified TSs to the licensee-controlled TRM consistent with the 10 CFR 50.36 requirements. The NRC staff has reviewed the licensee's submittal and finds that relocation of these requirements to a licensee-controlled document is acceptable in that the limiting conditions for operation and associated requirements were found not to fall within the scope of the criteria contained in 10 CFR 50.36(c)(2)(ii), and changes to licensee-controlled documents will be adequately controlled by 10 CFR 50.59, as applicable. These provisions will continue to be implemented by appropriate station procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures).

3.1 Movable Incore Detectors

This specification ensures the operability of the movable incore detector system when required to monitor the flux distribution within the core. The movable incore detector system is used for periodic surveillance of the power distribution and calibration of the excore detectors. This surveillance verifies that the peaking factors are within their design envelope. This system is not used continuously and does not initiate any automatic protective action. This system does not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, the movable incore detector requirement may be relocated to the TRM.

This change is acceptable because the movable incore detectors do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. The movable incore detectors are not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. The movable incore detectors do not satisfy criterion 1.

2. The movable incore detectors are not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The movable incore detectors do not satisfy criterion 2.

3. The movable incore detectors are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The movable incore detectors do not satisfy criterion 3.

4. Review of the movable incore detectors has determined that this system is not a significant contributor to the health and safety of the public. Information provided by the movable incore detectors would be of little or no use in mitigating the consequences of a severe accident, and such information is not modeled in probabilistic risk analyses (PRAs). The information provided is not used by operators during the design basis transients, or any severe accidents. Therefore, the movable incore detectors are not risk significant and do not satisfy Criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the movable incore detectors specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.2 Meteorological Instrumentation

The meteorological instrumentation is used to record meteorological data for use in evaluating the effect of an accidental radioactive release from the plant. Operation of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. The meteorological instrumentation is not used to mitigate a DBA or transient. This specification does not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, it may be relocated to the TRM.

This change is acceptable because meteorological instrumentation does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. Meteorological instrumentation is not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. Meteorological instrumentation does not satisfy criterion 1.

2. Meteorological instrumentation is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Meteorological instrumentation does not satisfy criterion 2.

3. Meteorological instrumentation is not assumed to function in the safety analysis. The meteorological instrumentation is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Meteorological instrumentation does not satisfy criterion 3.

4. Offsite dose calculations in PRA studies for large accidental releases of radioactive materials rely on conservative meteorological and public evacuation assumptions and do not credit meteorological instruments cited in this TS to guide emergency measures to protect the public. In addition, routine releases of radioactive materials are not risk significant. Meteorological Instrumentation does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the meteorological instrumentation specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.3 Reactor Coolant System (RCS) Chemistry

This specification places limits on the oxygen, chloride, and fluoride content in the RCS to minimize corrosion. The limitations on the RCS chemistry, provided by this requirement, ensure that corrosion of the RCS is minimized and reduces the potential for RCS leakage or failure due to corrosion. Maintaining the chemistry within the steady-state limits provides adequate corrosion protection to ensure the structural integrity of the RCS over the life of the plant. The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action before significant corrosion can occur. This specification does not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, it may be relocated to the TRM.

This change is acceptable because RCS chemistry does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. RCS chemistry is not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. RCS chemistry does not satisfy criterion 1.

2. RCS chemistry is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. RCS chemistry does not satisfy criterion 2.

3. RCS chemistry is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. RCS chemistry does not satisfy criterion 3.

4. Review of RCS chemistry has determined that it is not a significant contributor to the health and safety of the public. Information provided by RCS chemistry would be of little or no use in mitigating the consequences of a severe accident, and such information is not modeled in PRAs. The information provided is not used by operators during the design basis transients, or any severe accidents. Therefore, RCS chemistry is not risk significant and does not satisfy Criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the RCS chemistry specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.4 RCS Head Vents

RCS head vents are provided to exhaust noncondensable gases and/or steam from the RCS which could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long-term cooling, such as a loss-of-coolant accident (LOCA). Their function, capabilities, and testing requirements are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI [Three-Mile Island] Action Plan Requirements." However, operation of the RCS head vents is not assumed in the safety analysis; this is because the operation of the vents is not part of the primary success path. The operation of these vents is an operator-initiated action after the event has occurred and is only required when there is indication that natural circulation is not occurring. RCS head vents do not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, the RCS head vent specification may be relocated to the TRM.

This change is acceptable because the RCS head vents do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. RCS head vents are not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. RCS head vents do not satisfy criterion 1.

2. RCS head vents are not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. RCS head vents do not satisfy criterion 2.

3. RCS head vents may be used to assist in creating conditions conducive to natural circulation, but are not components that are part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. RCS head vents do not satisfy criterion 3.

4. RCS head vents are not prime contributors to dominant risk sequences of PRAs. The design of Westinghouse PWRs [pressurized-water reactors] is such that a buildup of noncondensable gases or steam within the primary system, which is sufficient to inhibit natural circulation core cooling, is unlikely. While an inadvertent opening of an RCS head vent would be equivalent to a small break LOCA, it would be a small contributor to the overall initiating event frequency and is not a primary contributor to risk. RCS head vents do not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the RCS head vents specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.5 <u>Steam Generator Pressure and Temperature Limitation</u>

This specification places limits on the steam generator pressure and temperature to ensure that the pressure induced stresses are within the maximum allowable fracture toughness stress limits. The pressure and temperature limits are based on a steam generator reference temperature sufficient to prevent brittle fracture. This specification does not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, it may be relocated to the TRM.

This change is acceptable because steam generator pressure and temperature do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. Steam generator pressure and temperature are not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. Steam generator pressure and temperature do not satisfy criterion 1.

2. Steam generator pressure and temperature are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Steam generator pressure and temperature do not satisfy criterion 2.

3. Steam generator pressure and temperature are not structures, systems, or components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Steam generator pressure and temperature do not satisfy criterion 3.

4. Steam generator pressure and temperature limits are intended to prevent the brittle fracture of the steam generator shell. Brittle fractures of steam generators are not specifically modeled in PRAs. During operation in Modes 1, 2, and 3, which are the operating modes typically evaluated in PRAs, violation of these pressure and temperature limits would be unlikely. Relocation of the limitation to the TRM is consistent with the approach of Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." Since GL 96-03 applies to the reactor pressure vessel, for which service-induced embrittlement is possible, and since the mechanical properties of the ferritic steel comprising the steam generator shell are not degraded by service conditions, it is appropriate to apply the same relocation philosophy. Based on these considerations, it can be concluded that the steam generator pressure and temperature limits specification do not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the steam generator pressure and temperature limitation specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.6 Sealed Source Contamination

This specification ensures that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR Part 70.39(a)(3) limits for plutonium. The limits contained in this specification do not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, they may be relocated to the TRM.

This change is acceptable because sealed source contamination limits do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. Sealed source contamination limits are not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. Therefore, sealed source contamination limits do not satisfy criterion 1.

2. Sealed source contamination limits are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, sealed source contamination limits do not satisfy criterion 2.

3. Sealed source contamination limits are not structures, systems, or components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a

fission product barrier. Therefore, sealed source contamination limits do not satisfy criterion 3.

4. The requirements addressed in this TS control removable low level contamination on sealed sources in order to limit leakage to allowable levels. These requirements are not addressed in PRAs and are not relevant to PRA conclusions. Therefore, sealed source contamination limits do not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the sealed source contamination specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.7 Refueling Operations Communications

This specification requires communication between the control room and the refueling station to ensure that any abnormal change in the facility status observed on the control room instrumentation can be communicated to the refueling station personnel. This communication requirement does not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, it may be relocated to the TRM.

This change is acceptable because the capability for communication between the control room and the refueling station does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. Refueling operations communication requirements are not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. Refueling operations communications do not satisfy criterion 1.

2. Refueling operations communication requirements are not process variables that are initial conditions of a DBA or transient analysis that either assume the failure of or present a challenge to the integrity of a fission product barrier. Refueling operations communication requirements do not satisfy criterion 2.

3. Refueling operations communication requirements are not contain structures, systems, or components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Refueling operations communication requirements do not satisfy criterion 3.

4. Direct communications between the control room and refueling station personnel during refueling are necessary to preclude inadvertent criticality. However, these communications are not addressed in PRA studies and are not a factor in accident sequences that are commonly found to dominate risk. Refueling operations communications requirements do not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the refueling operations communications specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.8 Manipulator Crane

This specification ensures that the lifting device on the manipulator crane has adequate capacity to lift the combined weight of a fuel assembly and a rod control cluster assembly, and that an automatic load limiting device is available to prevent damage to a fuel assembly that becomes stuck during fuel movement. This specification also ensures that the auxiliary hoist on the manipulator crane has adequate capacity for latching and unlatching control rod drive shafts. This specification does not meet the criteria of 10 CFR 50.36 for retention in the SQN TSs; therefore, the manipulator crane specification may be relocated to the TRM.

This change is acceptable because the manipulator crane and the auxiliary hoist do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in the TSs. The 10 CFR 50.36(c)(2)(ii) criteria evaluation is the following:

1. This manipulator crane and the auxiliary hoist are not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. The manipulator crane and the auxiliary hoist do not satisfy criterion 1.

2. This manipulator crane and the auxiliary hoist specification are not process variables that are an initial condition of a DBA or transient analysis that either assume the failure of or present a challenge to the integrity of a fission product barrier. The manipulator crane and the auxiliary hoist do not satisfy criterion 2.

3. This manipulator crane and the auxiliary hoist are not structures, systems, or components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The manipulator crane and the auxiliary hoist do not satisfy criterion 3.

4. Review of the manipulator crane and auxiliary hoist has determined that they are not significant contributors to the health and safety of the public. The manipulator crane and auxiliary hoist would be of little or no use in mitigating the consequences of a severe accident, and they are not modeled in PRAs. The manipulator crane and auxiliary hoist are not used by operators during the design basis transients, or any severe accidents. Therefore, the manipulator crane and auxiliary hoist do not satisfy Criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the manipulator crane and auxiliary hoist specification may be relocated to the TRM and any subsequent changes will be controlled by the provisions of 10 CFR 50.59.

3.10 Summary

The NRC staff has reviewed the licensee's proposal to relocate the above TSs to the TRM. The staff finds these changes consistent with 10 CFR 50.36 criteria and precedent established in NUREG-1431, Revision 3. On this basis, the NRC staff concludes that the proposed changes do not diminish the level of safety ensured by operation in accordance with the SQN TSs and are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (70 FR 38723). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: December 28, 2005