UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 13, 2009

Mr. Preston D. Swafford
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT REGARDING ELIMINATING THE TIME LIMIT ON PURGE AND VENT VALVE OPERATIONS AND CONSOLIDATING CONTAINMENT ISOLATION VALVE SPECIFICIATIONS (TS 08-02) (TAC NOS. MD8533 AND MD8534)

Dear Mr. Swafford:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 323 to Facility Operating License No. DPR-77 and Amendment No. 315 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 15, 2008, as supplemented on December 10, 2008. The amendments change and realign several containment isolation subject matter technical specifications to the NRC technical report, NUREG-1431, Revision 3, "Standard Technical Specifications Westinghouse Plants." The amendments also remove the annual limit on purge hours.

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

Sincerely,

Tracy J. Orf

Tracy J. Orf, 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. 323 to License No. DPR-77
2. Amendment No. 315 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 323
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 15, 2008, as supplemented on December 10, 2008, 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 Title 10 of the *Code of Federal Regulations*, 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. 323 , 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

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the license and
Technical Specifications

Date of Issuance: April 13, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 323

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace Page 3 of Operating License DPR-77 with the attached page.

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 VII	Index VII
1-2	1-2
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-15	3/4 6-15
3/4 6-17	3/4 6-17
3/4 6-18	3/4 6-18
3/4 6-18a	3/4 6-18a
3/4 9-4	3/4 9-4
6-10a	6-10a

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis, instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 323 are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at power level different from there described; and

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and

f. Secondary containment bypass leakage is within the limits of Specification 3.6.3.

CONTROLLED LEAKAGE

1.8 This definition has been deleted.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMIT REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BYPASS LEAKAGE (DELETED)

LIMITING CONDITION FOR OPERATION

SECONDARY CONTAINMENT BYPASS LEAKAGE (DELETED)

SURVEILLANCE REQUIREMENTS

CONTAINMENT VENTILATION SYSTEM (DELETED)

LIMITING CONDITION FOR OPERATION

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.*

NOTES

*1. Penetration flow path(s) may be unisolated intermittently under administrative controls.

*2. Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

*3. No more than one pair of Containment purge lines (one set of supply valves and one set of exhaust valves) may be opened.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTES

#1. Isolation devices in high radiation areas may be verified by use of administrative means.

#2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

##3. A check valve with flow through the valve secured is only applicable to penetration flow paths with two containment isolation valves.

a. With one or more penetration flow paths with one containment isolation valve inoperable for reasons other than:
   1. leakage rate limits of containment purge isolation valve(s),
   2. leakage rate limit of BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, or
   3. inoperable containment vacuum relief isolation valve(s),

   isolate the affected penetration within 4 hours by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve## with flow through the valve secured; and,

   verify# the affected penetration flow path is isolated once per 31 days for isolation devices outside containment, and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment.

ACTIONS (continued)

b. With more than one pair of containment purge lines open

or

with one or more penetration flow paths with two containment isolation valves inoperable for reasons other than:

1. leakage rate limits of containment purge isolation valve(s),
2. leakage rate limit of BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, or
3. inoperable containment vacuum relief isolation valve(s),

isolate the affected penetration within 1 hour by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange and verify# the affected penetration flow path is isolated once per 31 days.

c. With one or more containment vacuum relief isolation valve(s) inoperable, the valve(s) must be returned to OPERABLE status within 72 hours.

d. With one or more BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING not within limit, restore within limit within 4 hours.

e. With one or more penetration flow paths with one or more containment purge supply and/or exhaust isolation valves not within leakage limits, isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 24 hours. Verify# the affected penetration flow path is isolated once per 31 days for isolation devices outside containment and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment. Perform SR 4.6.3.6 once per 92 days for the valve used to isolate the affected penetration flow path.

f. With one or more penetration flow paths of a closed system design with one containment isolation valve inoperable, isolate the affected penetration flow path within 72 hours by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange, and verify# the affected penetration is isolated once per 31 days.

g. With any of the above ACTIONS not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

-- NOTE --

* Valves and blind flanges in high radiation areas may be verified by use of administrative means.

4.6.3.1 Verify each purge supply and/or exhaust isolation valve is closed, except when containment purge valves (only one set of supply valves and one set of exhaust valves) are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open, at least once per 31 days.

3/4.6.3 CONTAINMENT ISOLATION VALVES

SURVEILLANCE REQUIREMENTS (continued)

4.6.3.2 Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal, at least once per 18 months.

4.6.3.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls prior to entering Mode 4 from Mode 5 if not performed within the previous 92 days.*

4.6.3.5 Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls, at least once per 31 days.*

4.6.3.6 Perform leakage rate testing for each containment purge supply and exhaust isolation valve at least once per 3 months.

4.6.3.7 Verify each containment purge valve is blocked to restrict the valve from opening greater than or equal to 50 degrees, at least once per 18 months.

4.6.3.8 Verify the combined leakage rate for all BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING is less than or equal to 0.25 L_a when pressurized to greater than or equal to P_a in accordance with the Containment Leakage Rate Test Program.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and both doors of both containment personnel airlocks may be open if:

  1. One personnel airlock door in each airlock is capable of closure, and

  2. One train of the Auxiliary Building Gas Treatment System is OPERABLE in accordance with Technical Specification 3.9.12, and

c. Each penetration* providing direct access from the containment atmosphere to the outside atmosphere shall be either:

  1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or

  2. Be capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve.

APPLICABILTY:

3.9.4.a. Containment Building Equipment Door - During movement of recently irradiated fuel within the containment.

3.9.4.b. and c. Containment Building Airlock Doors and Penetrations - During movement of irradiated fuel within the containment.

ACTION:

1. With the requirements of the above specification not satisfied for the containment building equipment door, immediately suspend all operations involving movement of recently irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

2. With the requirements of the above specification not satisfied for containment airlock doors or penetrations, immediately suspend all operations involving movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve once per 7 days during movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their required condition, or

b. Verifying the Containment Ventilation isolation valves not locked, sealed, or otherwise secured in position, actuate to the isolation position on an actual or simulated actuation signal.

* Penetration flow path(s) providing direct access from the containment atmosphere that transverse and terminate in the Auxiliary Building Secondary Containment Enclosure may be unisolated under administrative controls.

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions and the following:

BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

    1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

    2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$ 6 psig for at least two minutes.

c. For each containment purge supply and exhaust isolation valve, acceptance criteria is measured leakage rate less than or equal to 0.05 L_a.

d. BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING acceptance criteria are:

    1. The combined bypass leakage rate to the auxiliary building shall be less than or equal to 0.25 L_a by applicable Type B and C tests.

    2. Penetrations not individually testable shall have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (12 psig) during each Type A test.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

i. Configuration Risk Management Program (DELETED)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 315
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 15, 2008, as supplemented on December 10, 2008, 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 Title 10 of the *Code of Federal Regulations*, 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. 315 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

T H Boyce

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the license and
Technical Specifications

Date of Issuance: April 13, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 315

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace Page 3 of Operating License DPR-79 with the attached page.

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 VII	Index VII
1-2	1-2
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-15	3/4 6-15
3/4 6-17	3/4 6-17
3/4 6-18	3/4 6-18
3/4 6-18a	3/4 6-18a
3/4 9-5	3/4 9-5
6-9	6-9
6-10	6-10
6-10a	6-10a
6-10b	6-10b
6-10c	6-10c
6-10d	6-10d
6-11	6-11

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 315 are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at power level different from there described; and

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed.
c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,
e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and
f. Secondary containment bypass leakage is within the limits of Specification 3.6.3.

CONTROLLED LEAKAGE

1.8 This definition has been deleted.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BYPASS LEAKAGE (DELETED)

LIMITING CONDITION FOR OPERATION

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BYPASS LEAKAGE (DELETED)

SURVEILLANCE REQUIREMENTS

CONTAINMENT VENTILATION SYSTEM (DELETED)

LIMITING CONDITION FOR OPERATION

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.*

-- NOTES --

*1. Penetration flow path(s) may be unisolated intermittently under administrative controls.

*2. Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

*3. No more than one pair of Containment purge lines (one set of supply valves and one set of exhaust valves) may be opened.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

-- NOTES --

#1. Isolation devices in high radiation areas may be verified by use of administrative means.

#2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

##3. A check valve with flow through the valve secured is only applicable to penetration flow paths with two containment isolation valves.

----

a. With one or more penetration flow paths with one containment isolation valve inoperable for reasons other than:
   1. leakage rate limits of containment purge isolation valve(s),
   2. leakage rate limit of BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, or
   3. inoperable containment vacuum relief isolation valve(s),

   isolate the affected penetration within 4 hours by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve## with flow through the valve secured; and,

   verify# the affected penetration flow path is isolated once per 31 days for isolation devices outside containment, and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment.

ACTIONS (continued)

b. With more than one pair of containment purge lines open

or

with one or more penetration flow paths with two containment isolation valves inoperable for reasons other than:

1. leakage rate limits of containment purge isolation valve(s),
2. leakage rate limit of BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, or
3. inoperable containment vacuum relief isolation valve(s),

isolate the affected penetration within 1 hour by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange and verify# the affected penetration flow path is isolated once per 31 days.

c. With one or more containment vacuum relief isolation valve(s) inoperable, the valve(s) must be returned to OPERABLE status within 72 hours.

d. With one or more BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING not within limit, restore within limit within 4 hours.

e. With one or more penetration flow paths with one or more containment purge supply and/or exhaust isolation valves not within leakage limits, isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 24 hours. Verify# the affected penetration flow path is isolated once per 31 days for isolation devices outside containment and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment. Perform SR 4.6.3.6 once per 92 days for the valve used to isolate the affected penetration flow path.

f. With one or more penetration flow paths of a closed system design with one containment isolation valve inoperable, isolate the affected penetration flow path within 72 hours by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange, and verify# the affected penetration is isolated once per 31 days.

g. With any of the above ACTIONS not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

----- NOTE -----

* Valves and blind flanges in high radiation areas may be verified by use of administrative means.

4.6.3.1 Verify each purge supply and/or exhaust isolation valve is closed, except when containment purge valves (only one set of supply valves and one set of exhaust valves) are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open, at least once per 31 days.

SURVEILLANCE REQUIREMENTS (continued)

4.6.3.2 Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal, at least once per 18 months.

4.6.3.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls prior to entering Mode 4 from Mode 5 if not performed within the previous 92 days.*

4.6.3.5 Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls, at least once per 31 days.*

4.6.3.6 Perform leakage rate testing for each containment purge supply and exhaust isolation valve at least once per 3 months.

4.6.3.7 Verify each containment purge valve is blocked to restrict the valve from opening greater than or equal to 50 degrees, at least once per 18 months.

4.6.3.8 Verify the combined leakage rate for all BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING is less than or equal to 0.25 L_a when pressurized to greater than or equal to P_a in accordance with the Containment Leakage Rate Test Program.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, or both doors of both containment personnel airlocks may be open if:

   1. One personnel airlock door in each airlock is capable of closure, and

   2. One train of the Auxiliary Building Gas Treatment System is OPERABLE in accordance with Technical Specification 3.9.12, and

c. Each penetration* providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or

   2. Be capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve.

APPLICABILITY:

3.9.4.a. Containment Building Equipment Door - During movement of recently irradiated fuel within the containment.

3.9.4.b. and c. Containment Building Airlock Doors and Penetrations - During movement of irradiated fuel within the containment.

ACTION:

1. With the requirements of the above specification not satisfied for the containment building equipment door, immediately suspend all operations involving movement of recently irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

2. With the requirements of the above specification not satisfied for containment airlock doors or penetrations, immediately suspend all operations involving movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve once per 7 days during movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their required condition, or

b. Verifying the Containment Ventilation isolation valves not locked, sealed, or otherwise secured in position, actuate to the isolation position on an actual or simulated actuation signal.

* Penetration flow path(s) providing direct access from the containment atmosphere that transverse and terminate in the Auxiliary Building Secondary Containment Enclosure may be unisolated under administrative controls.

6.8.4 f. Radioactive Effluent Controls Program (Cont.)

of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be in accordance with the following:

1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and
2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radio-nuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

The provisions of SR 4.0.2 and 4.0.3 are applicable to the radioactive effluent controls program surveillance frequency.

g. Radiological Environmental Monitoring Program (DELETED)

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions and the following:

BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the combined Type B and Type C tests, and ≤ 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

2) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 6 psig for at least two minutes.

c. For each containment purge supply and exhaust isolation valve, acceptance criteria is measured leakage rate less than or equal to 0.05 L_a.

d. BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING acceptance criteria are:

1. The combined bypass leakage rate to the auxiliary building shall be less than or equal to 0.25 L_a by applicable Type B and C tests.

2. Penetrations not individually testable shall have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (12 psig) during each Type A test.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

i. Configuration Risk Management Program (DELETED)

j. Technical Specification (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these TSs.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.8.4.j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

k. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for Condition Monitoring Assessments.

Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. Provisions for Performance Criteria for SG Tube Integrity.

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents (DBAs). This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and except for flaws addressed through application of the alternate repair criteria discussed in TS 6.8.4.k.c.1, a safety factor of 1.4 against burst applied to the DBA primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the DBAs, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   For predominantly axially oriented outside diameter stress corrosion cracking (ODSCC) at the tube support plate elevations, (refer to 6.8.4.k.c.1) the probability of burst (POB) of one or more indications given a steam line break shall be less than 1×10^{-2}.

2. Accident induced leakage performance criterion: The accident-induced leakage from all sources, excluding the leakage attributed to the degradation described in 6.8.4.k.c.1 and .2, is not to exceed 1.0 gpm for the faulted SG and 0.1 gpm for each of the non-faulted SGs. The primary-to-secondary accident induced leakage rate for any DBA, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG.

3. The operational leakage performance criterion is specified in Limiting Condition for Operation (LCO) 3.4.6.2, "Reactor Coolant System, Operational Leakage."

c. Provisions for SG Tube Repair Criteria.

Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube repair criteria (ARC) may be applied as an alternative to the 40% depth based criteria:

1. <u>NRC Generic Letter (GL) 95-05 Voltage-Based ARC (Tube Support Plate [TSP])</u>

A voltage-based TSP repair criteria is used for the disposition of an alloy 600 SG tube for continued service that is experiencing predominately axially oriented ODSCC confined within the thickness of the tube support plates (TSPs). At TSP intersections, the repair criteria is described below:

a) SG tubes, whose degradation is attributed to ODSCC within the bounds of the TSP with bobbin voltages less than or equal to 2.0 volts, will be allowed to remain in service.

b) SG tubes, whose degradation is attributed to ODSCC within the bounds of the TSP with a bobbin voltage greater than 2.0 volts will be plugged, except as noted in Item 6.8.4.k.c.1.c) below.

c) SG tubes, with indications of potential degradation attributed to ODSCC within the bounds of the TSP with a bobbin voltage greater than 2.0 volts, but less than or equal to the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented), may remain in service if a rotating pancake coil inspection or comparable technology does not detect degradation.

d) SG tubes with indications of ODSCC degradation with a bobbin coil voltage greater than the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) will be plugged.

e) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in Items 6.8.4.k.c.1.a), .b), .c) and .d).

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\frac{(CL - \Delta t)}{CL}}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\frac{(CL - \Delta t)}{CL}$$

where:

V_{URL} = upper voltage repair limit

V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled SG inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS items 6.8.4.k.c.1.a), .b), .c) and .d).

2. W* Methodology

The following terms/definitions apply to the W*.

a) Bottom of WEXTEX Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the top of tubesheet (TTS), as determined by eddy current testing.

b) W* Distance is the larger of the following two distances as measured from the TTS: (a) 8 inches below the TTS or (b) 7 inches below the bottom of the WEXTEX transition plus the uncertainty associated with determining the distance below the bottom of the WEXTEX transition as defined by WCAP-14797, Revision 2.

Service induced flaws identified in the W* distance shall be plugged on detection. Flaws located below the W* distance may remain in service regardless of size.

d. Provisions for SG Tube Inspections.

Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, d.4, and d.5, below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SGs shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

4. GL 95-05 Voltage-Based ARC for TSP

   Indications left in service as a result of application of the TSP voltage-based repair criteria shall be inspected by bobbin coil probe every 24 effective full power months or every refueling outage, whichever is less.

   Implementation of the SG tube/TSP repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg TSP intersections down to the lowest cold-leg TSP with known ODSCC indications. The determination of the lowest cold-leg TSP intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

5. W* Inspection

   When the W* methodology has been implemented, inspect 100 percent of the inservice tubes in the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 6.8.4.k.c.2.

e. Provisions for Monitoring Operational Primary-to-Secondary Leakage.

l. <u>Component Cyclic and Transient Limit</u>

This program provides controls to track the FSAR, Section 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 DELETED

6.9.1.2 DELETED

6.9.1.3 DELETED

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 DELETED

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 323 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 315 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 15, 2008, as supplemented on December 10, 2008, the Tennessee Valley Authority (the licensee) proposed amendments to the Technical Specifications (TSs) for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The December 10, 2008, supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The requested changes would modify the SQN Unit 1 and Unit 2 TSs to be more consistent with those of the Nuclear Regulatory Commission (NRC or Commission) technical report, NUREG-1431, Revision 3.0, "Standard Technical Specifications [STSs] Westinghouse Plants."

The primary intent of the license amendment request is to eliminate the cumulative time limit of 1000 hours per year for purge and vent valve operation specified in Limiting Condition for Operation (LCO) 3.6.1.9, "Containment Ventilation System." Following the March 1979 accident at Three Mile Island Unit 2, the NRC issued NUREG-0737. Section II.E.4.2, "Containment Isolation Dependability," of this document provided an interim position on containment purge and vent valve operation pending resolution of open issues related to isolation valve operability. The position stated that "whenever the containment integrity is required, emphasis should be placed on operating the containment in a passive mode as much as possible and on limiting all purging and venting times to as low as achievable." An allowable number of hours for purging was not specified. A limit of 90 hours per 365 days was imposed on SQN Unit 1 as documented in Supplement No. 2 to the SQN Unit 1 safety evaluation report. The 1000-hour TS purge time limit was subsequently established for Unit 1 by License Amendment No. 5 dated April 15, 1981, based on the need to allow containment venting for operational needs. The staff's basis for finding this acceptable was that the design of the system conforms to the operability criteria and associated dose criteria described in Branch Technical Position (BTP) CSB 6-4, "Containment Purging during Normal Plant Operation." The same limit was established for Unit 2 by License Amendment No. 10 dated December 23, 1982. The licensee provided justification that deletion of the 1000-hour limit would not affect requirements regarding valve operability and dose consequences of a postulated accident.

Westinghouse Electric Company report WCAP 12159, submitted by the Westinghouse Owners Group, proposed eliminating the time restriction on containment purging from the STSs. NUREG-1431, Revision 3.0 does not contain this limitation.

The SQN Updated Final Safety Analysis Report (UFSAR), Section 9.4.7, describes the reactor building purge ventilating (RBPV) system. The RBPV system maintains the environment in the primary and secondary containment within acceptable limits for equipment operation and personnel access. The RBPV system is not safety-related except for containment penetrations. The UFSAR provides the following description of the safety function:

> The containment purge penetrations are safety-related in that they must not jeopardize the integrity of the containment boundary. These penetrations are designed to withstand (with essentially no leakage) the forces produced by a Loss-of-Coolant Accident (LOCA), or a Main Steam Line Break (MSLB).

The containment ventilation isolation of the purge valves is initiated either by manual action, by a containment purge air exhaust monitor activity-high initiating signal, or by a safety injection signal (UFSAR Table 7.3.1-2). The isolation mechanism has 100-percent redundancy in both equipment and power sources.

The licensee is proposing a TS surveillance requirement (SR 4.3.6.7), which verifies that the purge and vent valves are restricted from opening more than 50 degrees. In addition to limiting dose release, this restriction helps to ensure the valves are capable of closure when required under accident conditions. SQN Unit 1 and Unit 2 TSs also require that each containment ventilation isolation valve actuates to its isolation position at least once per 18 months.

Thus, design requirements and TSs surveillances ensure that the RBPV system isolation valves will close against accident pressure if open at the initiation of a design basis accident.

By a letter dated September 17, 2007, the licensee requested a 400-hour increase in allowed purge operation for Unit 2. The NRC approved this request in Amendment No. 308 dated October 11, 2007. The licensee's April 15, 2008, letter stated that 300 of the requested 400 additional hours were used. The licensee is proposing to remove the footnote associated with Amendment No. 308.

2.0 <u>REGULATORY EVALUATION</u>

Section 182a of the Atomic Energy Act (Act) requires applicants for nuclear power plant operating licenses to include TSs as part of the license. These TSs are derived from the plant safety analyses.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36 contains the requirements for the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) Surveillance Requirements; (4) design features; and (5) administrative controls.

10 CFR 50.36(c)2(ii) lists the criteria used to determine whether or not LCOs must be established in the TSs for items related to plant operation. If the item falls in to one of the four categories below, an LCO must be established in the TSs to ensure the lowest functional capability or performance level of equipment required for safe operation of the facility will be met. The four criteria are:

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 a fission product barrier.

Criterion 3 A structure, system, or component (SSC) 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.

Criterion 4 An SSC which operating experience or probabilistic risk assessment (PRA) has shown to be significant to public health and safety.

For items that do not meet any of the above criteria, a TS LCO is not required. The commission's policy on TSs has evolved over the years and some licensees have not yet submitted requests to remove plant-specific TSs content which may not meet any of the 4 criteria outlined above.

10 CFR 50.36 does not specify each particular requirement to be included in a plant's TSs, nor does it specify the format of a plant's TSs. Rather, the NRC publishes generic guidance on TSs format and content.

The STSs in NUREG-1431 are a guide to what a plant's TSs should contain with regard to format and content. The STSs are not requirements in a regulatory sense, but licensees adopting portions of the improved STSs to existing TSs should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

The staff reviewed the proposed changes for compliance with 10 CFR 50.36 and agreement with the precedent as established in NUREG-1431. In general, licensees cannot justify TS changes solely on the basis of adopting the model STSs. To ensure this, the staff makes a determination that proposed changes maintain adequate safety. Changes that result in relaxation (less restrictive condition) of current TS requirements require detailed justification.

In general, there are two classes of changes to TSs: (1) changes needed to reflect contents of 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 the second class of change, namely, the removal of the restriction on cumulative hours of operation with purge and vent valves open and TSs content not contained in the latest version of NUREG-1431.

Licensees may revise the TSs to adopt 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.

10 CFR Part 100, "Reactor Site Criteria," establishes siting criteria to ensure that radiological doses from normal operation and postulated accidents will be acceptably low. Subpart A states,"It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure."

In addition to providing regulatory dose criteria for protection of the public, the NRC requires that control room personnel be protected from the potential radiological consequences of a DBA. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the design criteria for water-cooled nuclear power plants. General Design Criteria (GDC)-19, "Control room," states, "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."

A design basis radiological consequence analysis is intended to be based upon a major accident, or possible event, resulting in dose consequences not exceeded by those from any accident considered credible (maximum hypothetical accident (MHA)). Unlike the design basis LOCA, used to evaluate the emergency core cooling system requirements of 10 CFR 50.46, the general scenario used to postulate a maximum hypothetical dose consequence accident does not represent any specific accident sequence. Rather, the MHA is intended to be a surrogate to enable a deterministic evaluation of the response of a facility's engineered safety features such as the primary containment system.

Regulatory Position 2.8 of Appendix A to Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," states that, "If the primary containment is purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100 percent of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the TS reactor coolant system equilibrium activity. Iodine spikes need not be considered."

10 CFR Part 50, Appendix A, GDC 54, "Piping systems penetrating containment," requires that the reliability and performance capabilities of containment isolation valves reflect the importance to safety of isolating the systems penetrating the containment boundary. Purge valves are of large diameter and provide a direct path from the containment atmosphere to the outside environment. The capability to isolate is therefore important to safety.

BTP CSB 6-4 provides guidance on the design and use of the purge system during operation. Included in the guidance are the following positions:

- The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum, and the supply line and exhaust line flows as a function of time up to and during valve closure.

- The number of supply and exhaust lines that may be used should be limited to one supply line and one exhaust line to improve the reliability of the isolation function.

- Valve closure times should not exceed five seconds.

- Debris should not interfere with isolation valve function.

- The valves should be actuated by diverse signals.

BTP CSB 6-4 guidance also states that a radiological consequence analysis should demonstrate that the containment purge system design results in dose limits within the guideline values. Closure within 5 seconds allows the assumption that the radiological source term consists only of the fuel rod gap activity.

3.0 TECHNICAL EVALUATION

The licensee submitted proposed changes to TS 3.6.1.2, "Secondary Containment Bypass Leakage," TS 3.6.1.9, "Containment Ventilation System," TS 3.6.3, "Containment Isolation Valves," TS 6.8.4.h, "Containment Leakage Rate Testing Program," and TS 3.9.4, "Containment Building Penetrations." Changes to the TSs Table of Contents and Definitions sections were also submitted to provide needed consistency due to the other changes. In keeping with the framework for adopting improved STS format and content outlined in Section 2.0, the licensee categorized the proposed changes as either: Administrative, Less Restrictive, or More Restrictive. The licensee also commented on the degree to which each change conformed to the content of NUREG-1431.

The NRC staff evaluated the proposed TS changes to determine if the proposed TSs continue to meet the requirements of 10 CFR 50.36, and if the proposed TSs are consistent with SQN's current licensing basis. This ensures that the proposed changes maintain adequate safety because it can be assumed that the current TSs for SQN maintain adequate safety. The framework mentioned in Section 2.0 was used to categorize the proposed changes as Administrative, Less Restrictive, or More Restrictive. Changes to individual TSs can have one or more types of changes in them. Therefore, each change will be called out in the applicable section, which discusses the changes in detail, below.

The staff also compared the proposed TSs to the content of NUREG-1431. Differences between the content of the proposed TSs and the content of NUREG-1431 are also addressed below in the applicable section. Minor differences between NUREG-1431 and plant-specific TSs are expected since each plant has a unique licensing basis that may not be totally reflected in the NRC's generic guidance.

The format of SQN TSs differ from the format of STSs. Licensees are not required to adopt the NUREG-1431 format. Formatting differences are not addressed below.

3.1 TS 3.6.1.2, "Secondary Containment Bypass Leakage"

The licensee proposed moving TS 3.6.1.2 requirements to TS 3.6.3, "Containment Isolation Valves" and TS 6.8.4.h, "Containment Leakage Rate Testing Program."

The LCO statement and footnote for TS 3.6.1.2 contain limits and requirements for bypass leakage paths to the auxiliary building. The licensee proposed moving the requirements of the LCO statement and footnote for TS 3.6.1.2 to SR 4.6.3.8, and the LCO action statements of LCO 3.6.3. The staff evaluated the proposed change and determined that it is an administrative change because the same actions will be required within the same lengths of time for the situation when one or more bypass leakage paths to the auxiliary building are not within limits. The staff determined that the proposed change is acceptable.

The ACTION requirements for LCO 3.6.1.2 contain requirements for when the combined bypass leakage for paths to the auxiliary building exceed .25 L_a during MODES 1,2,3, and 4. The licensee proposed revising the ACTION requirements for LCO 3.6.1.2 and moving them to ACTIONs d and g of LCO 3.6.3. The staff evaluated the proposed change and determined that the revised actions are equivalent to the ACTIONs of LCO 3.6.1.2 and apply during the same modes, therefore, this change is administrative. The staff determined that the proposed change is acceptable.

SR 4.6.1.2 contains requirements to ensure Secondary Containment is operable. The licensee proposed revising the language or SR 4.6.1.2 for consistency with NUREG-1431 and moving the SR to SR 4.6.3.8 and TS 6.8.4.h. The staff evaluated the proposed change and determined that the changes are administrative because the TS requirements are not materially altered and continue to ensure Secondary Containment Operability. The staff determined that the proposed change is acceptable.

Finally, the licensee proposed deleting TS 3.6.1.2, SR 4.6.1.2 and references to the TSs in the index. The staff reviewed the change and found that all TS requirements continue to be maintained by the proposed changes mentioned above. Therefore, the staff found all proposed changes to TS 3.6.1.2 acceptable.

3.2 TS 3.6.1.9, "Containment Ventilation System"

The licensee proposed moving parts of TS 3.6.1.9 requirements to TS 3.6.3, "Containment Isolation Valves," TS 6.8.4.h, "Containment Leakage Rate Testing Program," and deleting parts of TS 3.6.1.9 that are no longer required per 10 CFR 50.36.

The LCO statement for TS 3.6.1.9 restricts operation of containment purge supply and exhaust lines and containment purge valves. Operation of the containment purge system is limited to 1000 hours per 365 days. Operation of the containment purge system is also limited to one pair of lines open at a time (one purge supply line and one purge exhaust line).

The licensee proposed deleting the limit on cumulative time of operation for the containment purge system. The staff evaluated the proposed change and determined that it is less restrictive than current requirements. The licensee provided justification for the less restrictive change. The licensee satisfactorily demonstrated that the limit on cumulative time of operation for the containment purge system does not meet any of the 4 criteria of 10 CFR 50.36(c)2(ii). The staff, therefore, concluded that based on the 4 criteria of 10 CFR 50.36(c)2(ii), the proposed deletion is acceptable. It is also noted that NUREG-1431 does not include a limit on purge hours.

The deletion of the 1000-hour TS limit is not expected to result in a significant increase in containment purge times. The SQN operating history indicates that annual purge times have consistently been below the limit, with only one exception. In 2007, component leaks at Unit 2 resulted in a buildup of aldehydes inside containment and a need for additional purging. The NRC approved a one-time increase in the limit to 1400 hours. Following maintenance activities in early 2008, the annual purge time returned to below the limit. The staff concludes that there is reasonable assurance that the purging and venting times would remain as low as reasonably achievable, consistent with the NUREG-0737 guidelines, if the TS limit were deleted.

The primary intent of the BTP CSB 6-4 guidance was to provide assurance that the valves would close under accident conditions. As previously noted, the basis for the staff's approval of the 1000-hour limit was that the design of the system conformed to the guidance in BTP CSB 6-4. Subsequent to the establishment of the 1000-hour limit, the containment ventilation system and purge valves were upgraded to conform to NUREG-0737, Section II.E.4.2, by addition of valve stops on the purge valves with 8, 12, and 24-inch valve diameters to limit purge valve opening to 50 degrees. This modification and the addition of debris screens, in conjunction with testing and analysis, provided additional assurance that the valves would close when required under accident conditions. Deletion of the 1000-hour limit would have no effect on the functionality and reliability of the valves to close under accident conditions.

BTP CSB 6-4 guidance also states that a radiological consequence analysis should demonstrate that the containment purge system design results in dose limits within the guideline values. In case of a LOCA, the containment pressure must be demonstrated to be sufficient so that the safety criteria of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," are satisfied. The LOCA analyses for SQN Unit 1 and Unit 2 discussed in the SQN UFSAR (Section 15.4.1.5, "Effect of Containment Purging") states that the licensee considered the most conservative combination of valves to be open. The loss of containment air mass due to the purge valves being open is included in calculating the peak cladding temperature and the acceptance criteria for peak cladding temperature continue to be met. The staff finds this conservative and acceptable.

The licensee's dose consequence design basis accident (DBA) analyses of record are described in SQN UFSAR Chapter 15, Section 15.5 "Environmental Consequences of Accidents." The dose consequences from a maximum hypothetical LOCA, which is bounding for the postulated rod ejection accident, are the only dose consequence DBA's that involve a radioactive release from the RBPV system. The licensee's analysis as described in the amendment submittal, assumed for its maximum hypothetical LOCA that one pair of purge valves would be open at the onset of the accident and a valve closure time of 5 seconds. The licensee based these calculation assumptions on NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 6.2.4, Revision 2 and BTP CSB 6-4. In this regard, it is assumed that reactor coolant source term activity is released from the purge valves prior to closure. This release path dose is then summed with the postulated doses from other release paths for the licensee maximum hypothetical LOCA. The licensee based this coolant activity source term on the criteria in ANSI-ANS18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors. The licensee analysis assumed a reactor coolant iodine activity of 10 times the equilibrium value providing a pre-existing iodine spike. Since it was determined that the purge valve operating time limitation does not affect the licensee's assumptions regarding the purge valve functionality and reliability, the current dose consequence maximum hypothetical LOCA analysis of record is not affected. Therefore, the NRC staff finds the elimination of the purge valve time limits TS acceptable in regards to the dose consequence DBA analyses as described in SQN UFSAR Chapter 15, Section 15.5.

In summary, the staff concludes that the elimination of the purge valve time limit is acceptable because the limit is not required by 10 CFR 50.36, the elimination of the purge valve operating time limitation does not affect the licensee's assumptions regarding the purge valve functionality and reliability, and the results of the current dose consequence analyses as described in the SQN UFSAR are not affected.

The staff notes that in its submittal, the licensee identified deviations in its analysis of record with respect to valve closure time and single failure of the emergency gas treatment system, which could result in greater radioactive releases following an accident. The licensee has made a preliminary determination that the deviations are acceptable and has entered it into its corrective action program for complete evaluation. The NRC staff did not evaluate the licensee's analysis in this safety evaluation, since the deletion of the purging time limit does not affect the dose consequence analysis. When the licensee completes its evaluation, any changes to the analysis of record should be reported or submitted for staff review in accordance with applicable regulatory requirements.

For Unit 2 only, the proposed change would remove a footnote that authorized an increase in purge hours during calendar year 2007. This is an administrative change to remove a footnote that is no longer applicable, and is acceptable.

The licensee proposed moving the restriction on the number of open containment purge lines to TS 3.6.3 ACTION b. The staff evaluated the proposed change and determined that the changes are administrative because the TS requirements are not materially altered and continue to ensure the restriction on open containment purge lines will be met during all Modes applicable to the current requirement. The staff determined that the proposed change is acceptable.

ACTION a for LCO 3.6.1.9 contains actions for when the operating restrictions of LCO 3.6.1.9 are not met. The licensee proposed deleting this ACTION. The staff evaluated the proposed change and determined that the changes are administrative because the TS requirements are not materially altered and continue to ensure the restriction on open containment purge lines will be met during all Modes applicable to the current requirement. The staff determined that the proposed change is acceptable.

ACTION b for LCO 3.6.1.9 contains actions for when the containment purge supply and/or exhaust isolation valves have a measured leakage rate in excess of .05 L_a. The licensee proposed revising the ACTION requirements and moving them to TS 3.6.3 ACTION e. The revised ACTIONS contain all the requirements of the current ACTIONS and would apply during the same MODES as the old ACTIONS. An extra requirement to perform SR 4.3.6.6 would be added to the revised ACTIONS. The staff reviewed the proposed change and determined that proposed change is more restrictive than the current requirement. The staff determined that the proposed change is acceptable.

SR 4.6.1.9.1 contains requirements to ensure the restrictions on containment purge supply and exhaust isolation valves are met. The licensee proposed revising and moving the SR to 4.6.3.1. The revised SR language is similar to the language contained in NUREG-1431. The staff evaluated the proposed change and determined that the changes are administrative because the TS requirements are not materially altered and continue to ensure the restrictions on containment purge supply and exhaust isolation valves are met. The staff determined that the proposed change is acceptable.

SR 4.6.1.9.2 contains requirements for when to determine the cumulative time that the purge supply and exhaust isolation valves have been open. The licensee proposed deleting this SR along with the requirement to limit cumulative time of operation for the containment purge system. The staff evaluated the proposed change and determined that it is less restrictive than current requirements. The licensee provided justification for the less restrictive change. As previously stated, the staff determined that the proposed change to delete the limit on cumulative time of operation for the containment purge system is acceptable. Consequently, there is no need for a requirement on determining the cumulative time that the valves are open. Therefore, the proposed change is acceptable.

SR 4.6.1.9.3 contains requirements for when and how to demonstrate containment purge supply and exhaust isolation valve OPERABILITY. The licensee proposed revising and moving the requirements to SR 4.6.3.6 and TS 6.8.4.h, "Containment Leakage Rate Testing Program." The requirements for when to perform the testing to demonstrate OPERABILITY would be moved to SR 4.6.3.6, and the requirements for how to demonstrate OPERABILITY would be moved to TS 6.8.4.h.c. The staff evaluated the proposed change and determined that the changes are administrative because the TS requirements are not materially altered and continue to ensure the restrictions on containment purge supply and exhaust isolation valves are met. The staff determined that the proposed change is acceptable.

Finally, the licensee proposed deleting TS 3.6.1.9 and references to the TS in the index. The staff found all proposed changes to TS 3.6.1.9 outlined above to be acceptable. Since all requirements of TS 3.6.1.9 were moved to other locations in TS, or deleted, the staff found it acceptable to delete TS 3.6.1.9 and its associated references in the index.

3.3 TS 3.6.3, "Containment Isolation Valves"

As discussed in Section 3.1 and 3.2 above, the licensee proposed moving some requirements located in TS 3.6.1.2, and TS 3.6.1.9 to TS 3.6.3. The proposed relocation of requirements would bring the SQN TS content into closer agreement with the content of NUREG-1431. The acceptability of each change is addressed at the end of Sections 3.1 and 3.2, respectively. The licensee also proposed changes to TS 3.6.3 that have not been addressed in previous sections.

The licensee proposed to add ACTION f to TS 3.6.3 to provide actions for when one or more penetration flow paths of a closed system design has an inoperable containment isolation valve. ACTION f would require the licensee to isolate the affected flow path within 72 hours and verify the affected penetration is isolated once per 31 days. The licensee stated that this change is less restrictive because the other action requirements for non-closed system flow paths with an inoperable containment isolation valve require isolation in less than 72 hours. The licensee justified the less restrictive change by stating that ACTION f will only apply to equipment that meets GDC 57 design criteria. The licensee also provided TS Bases pages which justify the 72-hour completion time as such: "The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4." The proposed ACTION f is identical in content to ACTIONS C.1 and C.2 of NUREG-1431. The staff evaluated the proposed less restrictive change and determined that the changes are acceptable.

SR 4.6.3.2 contains requirements for when and how to demonstrate automatic containment isolation valve OPERABILITY for each of the five listed types of isolation valves. The licensee proposed deleting the unnecessary technical detail and modifying the language in SR 4.6.3.2 to include a more general statement that reads "Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal, at least once per 18 months." The new SR language allows removal of unnecessary technical detail and also clarifies the fact that valves that are locked, sealed or otherwise secured in position are not required to be tested. The licensee stated that the technical detail will be moved to the TS Bases and that the Bases will be controlled using the TS Bases Control Program. The revised language is consistent with NUREG-1431. The staff evaluated the proposed change and determined that the changes are administrative because the TS requirements are not materially altered. The staff determined that the proposed change is acceptable.

The licensee proposed adding SR 4.6.3.7 to TS 3.6.3. The proposed language for SR 4.6.3.7 is as follows: "Verify each containment purge valve is blocked to restrict the valve from opening greater than or equal to 50 degrees, at least once per 18 months." The licensee stated that this is a more restrictive change, since the SR does not currently exist in the SQN TS. The licensee also stated that SR 4.6.3.7 is similar, but not identical to, NUREG-1431 SR 3.6.3.10. The staff compared the proposed change to the corresponding SR in NUREG-1431. The language used in SR 4.6.3.7 is not identical to that used in NUREG-1431. The staff determined that the language differences are due to SQN's plant-specific licensing basis. The staff reviewed the proposed change and determined that the change is more restrictive than current SQN TS and will provide an additional level of assurance that containment ventilation purges valves will remain OPERABLE during design basis event conditions. The staff determined that the proposed change is acceptable.

The licensee proposed administrative changes to TS 3.6.3 that accommodate the relocations and additions mentioned above. These changes are editorial revisions to ensure the alphabetical and numerical order of the ACTIONS and surveillance requirements are correct, as well as minor language revisions to make the TS more consistent with NUREG-1431. The staff evaluated the proposed changes and determined that the changes are acceptable.

3.4 TS 3.9.4, "Containment Building Penetrations"

SR 4.9.4.b contains requirements for when and how to demonstrate containment isolation valve OPERABILITY during movement of irradiated fuel in the containment building. SR 4.9.4.b currently states: "Testing per the applicable portions of Specification 4.6.3.2." The licensee proposed revising language in SR 4.6.3.2 to remove excess technical detail. The staff determined that the proposed change to SR 4.6.3.2 was acceptable, as detailed in section 3.3 above. In order to maintain consistency in the TS, the licensee proposed revising language in SR 4.9.4.b to read: "Verifying the Containment Ventilation isolation valves not locked, sealed, or otherwise secured in position, actuate to the isolation position on an actual or simulated actuation signal." The licensee stated that the proposed change is administrative. The staff evaluated the proposed change and determined that the change is administrative because the TS requirements are not materially altered. The staff also determined that the proposed change makes the TS more consistent with NUREG-1431. The staff determined that the proposed change is acceptable.

3.5 TS 6.8.4.h, "Containment Leakage Rate Testing Program"

As discussed in Section 3.2 and 3.3 above, the licensee proposed moving some requirements located in TS 3.6.1.2, and TS 3.6.1.9 to TS 6.8.4.h. The proposed relocation of requirements would bring the SQN TS content into closer agreement with the content of NUREG-1431. The acceptability of each change is addressed at the end of Sections 3.2 and 3.3, respectively. The licensee also proposed changes to TS 6.8.4.h that have not been addressed in previous sections.

The licensee proposed deleting the following sentence from SQN Unit 1 TS 6.8.4.h: "Performance of the spring 2003 containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than spring 2008." The licensee proposed deleting the following sentence from SQN Unit 2 TS 6.8.4.h: "Performance of the spring 2003 containment integrated leakage rate (Type A) test may be deferred to no later than spring 2007." The licensee stated that the integrated leakage rate (Type A) tests were performed during fall 2007, and fall 2006 for SQN Units 1 and 2, respectively. The staff evaluated the proposed change and determined that it is administrative and removes requirements that are no longer necessary. The staff determined that the proposed changes are acceptable.

The licensee proposed administrative changes to TS 6.8.4.h that accommodate the relocations and additions mentioned above. These changes are editorial revisions to ensure the alphabetical and numerical order items listed in the TS are correct, as well as minor language revisions to make the TS more consistent with NUREG-1431. The staff evaluated the proposed changes and determined that the changes are 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 and change surveillance requirements. 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 (73 FR 29164). 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.

Principal Contributors: Matthew E. Hamm
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Date: April 13, 2009

April 13, 2009

Mr. Preston D. Swafford
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Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT REGARDING ELIMINATING THE TIME LIMIT ON PURGE AND VENT VALVE OPERATIONS AND CONSOLIDATING CONTAINMENT ISOLATION VALVE SPECIFICIATIONS (TS 08-02) (TAC NOS. MD8533 AND MD8534)

Dear Mr. Swafford:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 323 to Facility Operating License No. DPR-77 and Amendment No. 315 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 15, 2008, as supplemented on December 10, 2008. The amendments change and realign several containment isolation subject matter technical specifications to the NRC technical report, NUREG-1431, Revision 3, "Standard Technical Specifications Westinghouse Plants." The amendments also remove the annual limit on purge hours.

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

Sincerely,

/RA/

Tracy J. Orf, 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. 323 to License No. DPR-77
2. Amendment No. 315 to License No. DPR-79
3. Safety Evaluation

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