TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

July 5, 2013

10 CFR 50.4
10 CFR 50.71(e)

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Units 1 and 2
Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Subject: **Revisions to the Sequoyah Nuclear Plant Technical Requirements Manual and Units 1 and 2 Technical Specification Bases**

References:
1. NRC Letter to TVA, "Issuance of Exemption to 10 CFR 71(e)(4) for the Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. MA0646 and MA0647)," dated March 9, 1998
2. TVA Letter to NRC, "Revisions to the Sequoyah Nuclear Plant Technical Requirements Manual and Units 1 and 2, Technical Specification Bases," dated December 16, 2011

Pursuant to 10 CFR 50.71(e) and the Reference 1 letter, updates to the Sequoyah Nuclear Plant (SQN) Updated Final Safety Analysis Report (UFSAR) for both Units 1 and 2 are to be submitted within six months after each refueling outage, not to exceed 24 months between successive revisions. The SQN Technical Requirements Manual (TRM) is incorporated by reference into the SQN UFSAR. In addition, SQN Technical Specification 6.8.4.j, "Technical Specification (TS) Bases Control Program," requires changes to the SQN TS Bases to be submitted in accordance with 10 CFR 50.71(e). This letter provides the required updates to the SQN TRM and TS Bases since the previous update submitted via the Reference 2 Letter. The last Unit 2 refueling outage ended on January 6, 2013, and as such, these updates are required by July 5, 2013. The enclosure to this letter provides a description of the TRM and TS Bases revisions with attachments of the updated pages, respectively.

A001
NRR

There are no commitments contained in this letter. If you have any questions, please contact Michael McBrearty at (423) 843-7170.

I certify that I am duly authorized by TVA, and that, to the best of my knowledge and belief, the information contained herein accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure:
Description of Revisions for the Sequoyah Nuclear Plant (SQN), Technical Requirements Manual and SQN, Units 1 and 2 Technical Specification Bases

cc (Enclosure):
NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

ENCLOSURE

DESCRIPTION OF REVISIONS FOR THE SEQUOYAH NUCLEAR PLANT (SQN), TECHNICAL REQUIREMENTS MANUAL AND SQN, UNITS 1 AND 2 TECHNICAL SPECIFICATION BASES

Technical Requirements Manual Revisions

Technical Requirements Manual (TRM) Revision 47 was approved on September 27, 2012, and implemented on October 5, 2012. A change was made to TR 3.1.2.2, "Flow Paths – Operating," and TR 3.1.2.4, "Charging Pumps – Operating," to allow an operational provision similar to the technical specifications (TSs) allowance for temporarily disabling one half of the boron injection function of the Chemical and Volume Control System (i.e., one charging pump and associated flow path) to support transition between Modes 3 and 4. Provisions are provided in the TSs that allow the Emergency Core Cooling System (ECCS) pumps to be made incapable of injecting, in Mode 3, for a limited amount of time or system conditions to support Low Temperature Over Pressure Protection (LTOP) System operations. This provision prevents TS non-compliance when entering into and out of Mode 3 when two charging pumps are required to be operable. This change aligns the TRM to be consistent with the TSs for support of LTOP System operations.

On May 15, 2007, TRM Revision 37 was reported to NRC. TRM Revision 37 was in support of SQN TS Amendment Nos. 305 for Unit 1 and 295 for Unit 2. A typographical error has been identified involving symbol characters with the issued page for TR 3.3.3.2, "Moveble Incore Detectors." The corrected page is submitted herein without change to the revision bar.

Technical Specification Bases Revisions

Revision 38 to the SQN, Units 1 and 2 Technical Specification (TS) Bases was approved on March 24, 2012, and implemented on March 26, 2012. The revision was in light of TS Change 07-05, "Emergency Core Cooling Systems (ECCS)" for SQN, Units 1 and 2,and associated with TS Amendment Nos. 326 and 319 approved on January 28, 2010. TS Bases Section 3.5.3, "ECCS - Shutdown," was revised to support primary and secondary residual heat removal check valve testing during Mode 4 operation. The changes differentiated TS Bases Section 3.5.2, "ECCS - Operating," Mode 1 through 3 safety analysis conditions from Mode 4 conditions, defines the necessary ECCS operation and flow paths, and added a reference.

Revision 39 to the SQN Unit 1 TS Bases was approved on October 5, 2012, and implemented on October 25, 2012. This revision was in concert with TS Amendment No. 330. Limiting Condition for Operation (LCO), 3.7.5 "Ultimate Heat Sink," was amended to support maintenance activities during the Unit 2 refueling outage No. 18. The TS Bases were changed to identify additional LCO restrictions with respect to maximum average Essential Raw Cooling Water (ERCW) system supply header water temperature during large heavy load lifts performed to support the refueling outage.

Revision 40 to the Unit 1 and Revision 39 to the Unit 2 TS Bases were approved on October 10, 2012. Unit 2 was implemented on November 28, 2012 during its refueling outage. Unit 1 will implement this revision during its next refueling outage in the Fall of 2013; therefore is not provided in this update. These TS Bases revisions are associated with TS Amendment Nos. 331 and 324 for approved TS Change 11-07, "Application to Modify Technical Specifications for Use of AREVA Advanced W17 HTP Fuel." This change affected TS Bases Section 2.1, "Safety

Limits," and Section 3/4.2.5, "DNB Parameters," as it provided clarifications associated with the evaluation methodology for the new fuel design.

Revision 40 to the Unit 2 TS Bases was approved on October 5, 2012, and implemented on November 2, 2012. This TS Bases revision to Section 3/4.4.5, "Reactor Coolant System," and Section 3/4.4.6.2, "Operation Leakage," was associated with replacement of the Unit 2 steam generators. The revision is associated with TS Amendment No. 323, in which previously approved steam generator inspections, specific repair criteria and reporting requirements had been modified or removed.

Revision 41 to the SQN, Units 1 and 2 Bases was approved on December 21, 2012, and implemented on December 27, 2012. This revision incorporated changes to the Bases for Specification 3.8.1, "A. C. Sources," to describe a new surveillance requirement approved under TS Amendment Nos. 332 and 325 for Units 1 and 2, respectively. Other changes to the TS Bases section include example descriptions of offsite power configurations that would meet the requirements of TS LCO 3.8.1.1.a.

Revision 42 to the SQN, Units 1 and 2 Bases was approved on March 5, 2013, and implemented on March 25, 2013. TS Bases Section 3/4.3.3.7, "Accident Monitoring Instrumentation," was revised. Statements describing accident monitoring instrumentation, specifically the SQN hydrogen monitoring channels were deleted. This change was associated with TS Amendment Nos. 296 and 286 for Units 1 and 2, respectively, which eliminated the requirements for hydrogen recombiners and hydrogen monitoring.

Also, enclosed is a typographical correction to TS Bases Table B 3/4.4-1, SQN Unit 1 Reactor Vessel Toughness Data with no indication of a revision bar. This change corrects the value of nickel in the weld material of the reactor vessel.

Attachments:

1. Sequoyah Nuclear Plant, Technical Requirements Manual - Changed Pages
2. Sequoyah Nuclear Plant, Unit 1, Technical Specification Bases - Changed Pages
3. Sequoyah Nuclear Plant, Unit 2, Technical Specification Bases - Changed Pages

ATTACHMENT 1

SEQUOYAH NUCLEAR PLANT
TECHNICAL REQUIREMENTS MANUAL
CHANGED PAGES

TRM Affected Pages

EPL-1
EPL-2
EPL-5
EPL-8
Index Page III
3/4 1-3 through 3/4 1-13
3/4 3-2
B 3/4 1-2
B 3/4 1-3

EFFECTIVE PAGE LISTING

Page	Revision
Index Page I	09/28/03
Index Page II	02/02/98
Index Page III	09/27/12
Index Page IV	01/20/06
Index Page V	01/20/06
Index Page VI	01/20/06
Index Page VII	02/02/98
Index Page VIII	02/02/98
1-1	02/02/98
1-2	05/18/09
1-3	09/28/03
1-4	07/19/02
1-5	07/25/02
1-6	02/02/98
1-7	02/02/98
1-8	02/02/98
3/4 0-1	05/27/05
3/4 0-2	05/27/05
3/4 0-3	07/25/06
3/4 0-4	07/25/06
3/4 1-1	01/04/01
3/4 1-2	10/12/05

EFFECTIVE PAGE LISTING

Page	Revision
3/4 1-3	09/27/12
3/4 1-4	09/27/12
3/4 1-5	09/27/12
3/4 1-6	09/27/12
3/4 1-7	09/27/12
3/4 1-9	09/27/12
3/4 1-10	09/27/12
3/4 1-11	09/27/12
3/4 1-12	09/27/12
3/4 1-13	09/27/12
3/4 3-1	01/20/06
3/4 3-2	01/20/06
3/4 3-3	01/20/06
3/4 3-4	01/20/06
3/4 3-5	01/20/06
3/4 3-6	10/17/06
3/4 3-7	04/26/06
3/4 3-8	04/26/06
3/4 3-9	01/20/06
3/4 3-10	01/20/06
3/4 4-1	01/20/06
3/4 4-2	01/20/06
3/4 4-3	01/20/06

EFFECTIVE PAGE LISTING

Page	Revision
B 3/4 1-1	01/04/01
B 3/4 1-2 Through B 3/4 1-3	09/27/12
B 3/4 1-4 Through B 3/4 1-6	01/04/01
B 3/4 3-1	01/20/06
B 3/4 3-2	01/20/06
B 3/4 3-3	01/20/06
B 3/4 3-4	01/20/06
B 3/4 3-5	01/20/06
B 3/4 3-6	01/20/06
B 3/4 3-7	01/20/06
B 3/4 3-8	01/20/06
B 3/4 3-9	01/20/06
B 3/4 3-10	01/20/06
B 3/4 3-11	01/20/06
B 3/4 3-12	01/20/06
B 3/4 3-13	01/20/06
B 3/4 3-14	01/20/06
B 3/4 4-1	01/20/06
B 3/4 4-2	01/20/06
B 3/4 4-3	01/20/06

REVISION LISTING

Revision	Date
Initial Issue, Revision 0	02/02/98
Revision 1	10/01/98
Revision 2	02/12/99
Revision 3	03/18/99
Revision 4	09/14/99
Revision 5	10/24/99
Revision 6	09/29/99
Revision 7	12/09/99
Revision 8	03/23/00
Revision 9	06/02/00
Revision 10	06/13/00
Revision 11	06/15/00
Revision 12	11/09/00
Revision 13	01/04/01
Revision 14	04/05/01
Revision 15	07/11/01
Revision 16	04/05/02
Revision 17	03/27/02
Revision 18	07/19/02
Revision 19	07/25/02
Revision 20	10/11/02
Revision 21	03/06/03
Revision 22	08/11/03
Revision 23	09/14/03
Revision 24	09/28/03
Revision 25	10/31/03
Revision 26	09/26/03
Revision 27	09/26/03
Revision 28	05/15/04
Revision 29	10/13/04
Revision 30	10/13/04
Revision 31	04/22/05
Revision 32	05/27/05
Revision 33	06/20/05
Revision 34	06/24/05
Revision 35	10/12/05
Revision 36	10/19/05
Revision 37	01/20/06
Revision 38	03/08/06
Revision 39	03/17/06
Revision 40	04/26/06
Revision 41	07/25/06
Revision 42	09/15/06
Revision 43	10/17/06
Revision 44	11/14/06
Revision 45	05/18/09
Revision 46	11/29/10
Revision 47	09/27/12

SECTION PAGE

TR 3/4.0 APPLICABILITY 3/4 0-1

TR 3/4.1 REACTIVITY CONTROL SYSTEMS

TR 3/4.1.1 (No current requirements) 3/4 1-1

TR 3/4.1.2 BORATION SYSTEMS

TR 3/4.1.2.1 FLOW PATHS - SHUTDOWN 3/4 1-2

TR 3/4.1.2.2 FLOW PATHS - OPERATING 3/4 1-3

TR 3/4.1.2.3 CHARGING PUMP - SHUTDOWN 3/4 1-5

TR 3/4.1.2.4 CHARGING PUMPS - OPERATING 3/4 1-6

TR 3/4.1.2.5 BORATED WATER SOURCES - SHUTDOWN 3/4 1-7

TR 3/4.1.2.6 BORATED WATER SOURCES - OPERATING 3/4 1-9

TR 3/4.1.3.1 Through TR 3/4.1.3.2 (No current requirements) 3/4 1-12

TR 3/4.1.3.3 POSITION INDICATION SYSTEM - SHUTDOWN 3/4 1-13

TR 3/4.2 POWER DISTRIBUTION LIMITS

No current requirements

TR 3/4.3 INSTRUMENTATION

TR 3/4.3.1 Through TR 3/4.3.3.1 (No current requirements) 3/4 3-1

TR 3/4.3.3 MONITORING INSTRUMENTATION

TR 3/4.3.3.2 MOVABLE INCORE DETECTORS 3/4 3-2

TR 3/4.3.3.3 SEISMIC INSTRUMENTATION 3/4 3-3

TR 3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION 3/4 3-6

TR 3/4.3.3.5 Through TR 3/4.3.3.14 (No current requirements) 3/4 3-9

TR 3/4.3.3.15 PLANT CALORIMETRIC MEASURMENT 3/4 3-10

TR 3/4.4 REACTOR COOLANT SYSTEM

TR 3/4.4.1 Through TR 3/4 4.6 (No current requirements) 3/4 4-1

TR 3/4 4.7 CHEMISTRY 3/4 4-2

TR 3/4.4.8 Through TR 3/4 4.9.1 (No current requirements) 3/4 4-5

TR 3/4 4.9.2 PRESSURIZER TEMPERATURE LIMITS 3/4 4-6

TR 3/4.4.10 (No current requirements) 3/4 4-7

TR 3/4 4.11 REACTOR COOLANT SYSTEM HEAD VENTS 3/4 4-8

TR 3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

No current requirements

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

TR 3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

------------------------------ NOTE ------------------------------

In MODE 3, one charging pump may be made incapable of injecting to support transition into or from the APPLICABILITY of Technical Specification LCO 3.4.12, “Low Temperature Overpressure Protection (LTOP) System,” for up to 4 hours or until the temperature of all RCS cold legs exceeds LTOP arming temperature (350°F) specified in the Pressure and Temperature Limits Report (PTLR) plus 25°F, whichever comes first.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

TR 4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the areas containing flow path components from the boric acid tanks to the blending tee is greater than or equal to 63°F when it is a required water source.

b. Whenever the area temperature(s) is(are) less than 63°F and the boric acid tank is a required water source, the solution temperature in the flow path components from the boric acid tank must be measured to be greater than or equal to 63°F within 6 hours and every 24 hours thereafter until the area temperature(s) has(have) returned to greater than or equal to 63°F.

SURVEILLANCE REQUIREMENTS (continued)

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

d. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

e. At least once per 18 months by verifying that the flow path required by TR 3.1.2.2a delivers at least 35 gpm to the Reactor Coolant System.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

TR 3.1.2.3 One charging pump in the boron injection flow path required by TR 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE shutdown board.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

MODE 4 - With no charging pump OPERABLE, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of Technical Specification LCO 3.1.1.1 and restore one changing pump as soon as possible.

MODE 5 - With no charging pump OPERABLE, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of Technical Specification LCO 3.1.1.2.

MODE 6 - With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS and suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet Technical Specification LCO 3.9.1.

SURVEILLANCE REQUIREMENTS

TR 4.1.2.3 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2400 psig when tested pursuant to TR 4.0.5.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

TR 3.1.2.4 At least two charging pumps shall be OPERABLE.

-- NOTE --
In MODE 3, one charging pump may be made incapable of injecting to support transition into or from the APPLICABILITY of Technical Specification LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for up to 4 hours or until the temperature of all RCS cold legs exceeds LTOP arming temperature (350°F) specified in the Pressure and Temperature Limits Report (PTLR) plus 25°F, whichever comes first.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

TR 4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 2400 psig when tested pursuant to TR 4.0.5.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

TR 3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 6400 gallons,

2. Between 6120 and 6990 ppm of boron, and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 55,000 gallons,

2. A minimum boron concentration of 2500 ppm, and

3. A minimum solution temperature of 60°F.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

MODE 4 - With no borated water source OPERABLE, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of Technical Specification LCO 3.1.1.1.

MODE 5 - With no borated water source OPERABLE, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of Technical Specification LCO 3.1.1.2.

MODE 6 - With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS and suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet Technical Specification LCO 3.9.1.

SURVEILLANCE REQUIREMENTS

TR 4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. For the boric acid storage system, when it is the source of borated water by:

1. Verifying the boron concentration at least once per 7 days,

2. Verifying the borated water volume at least once per 7 days, and

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the boric acid storage tank solution temperature is greater than or equal to 63°F at least once per 7 days by verifying the area temperature to be greater than or equal to 63°F, or

4. When the boric acid tank area temperature is less than 63°F and the boric acid storage system being used as the source of borated water, within 6 hours and every 24 hours thereafter, verify the boric acid tank solution temperature to be greater than or equal to 63°F until the boric acid tank area temperature has returned to greater than or equal to 63°F.

b. For the refueling water storage tank by:

1. Verifying the boron concentration at least once per 7 days,

2. Verifying the borated water volume at least once per 7 days, and

3. Verifying the solution temperature at least once per 24 hours while in Mode 4 or while in Modes 5 or 6 when it is the source of borated water.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

TR 3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by TR 3.1.2.2:

a. A boric acid storage system with:

1. A contained volume of borated water in accordance with Figure 3.1.2.6,

2. A boron concentration in accordance with Figure 3.1.2.6, and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A contained borated water volume of between 370,000 and 375,000 gallons,

2. Between 2500 and 2700 ppm of boron,

3. A minimum solution temperature of 60°F, and

4. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TR 4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. For the boric acid storage system, when it is the source of borated water by:

   1. Verifying the boron concentration at least once per 7 days,

   2. Verifying the borated water volume at least once per 7 days, and

   3. Verifying the boric acid storage tank solution temperature is greater than or equal to 63°F at least once per 7 days by verifying the area temperature to be greater than or equal to 63°F, or

   4. Whenever the boric acid tank area temperature is less than 63°F and the boric acid storage system being used as the source of borated water, within 6 hours and every 24 hours thereafter, verify the boric acid tank solution temperature to be greater than or equal to 63°F until the boric acid tank area temperature has returned to greater than or equal to 63°F.

b. For the refueling water storage tank by:

   1. Verifying the boron concentration at least once per 7 days,

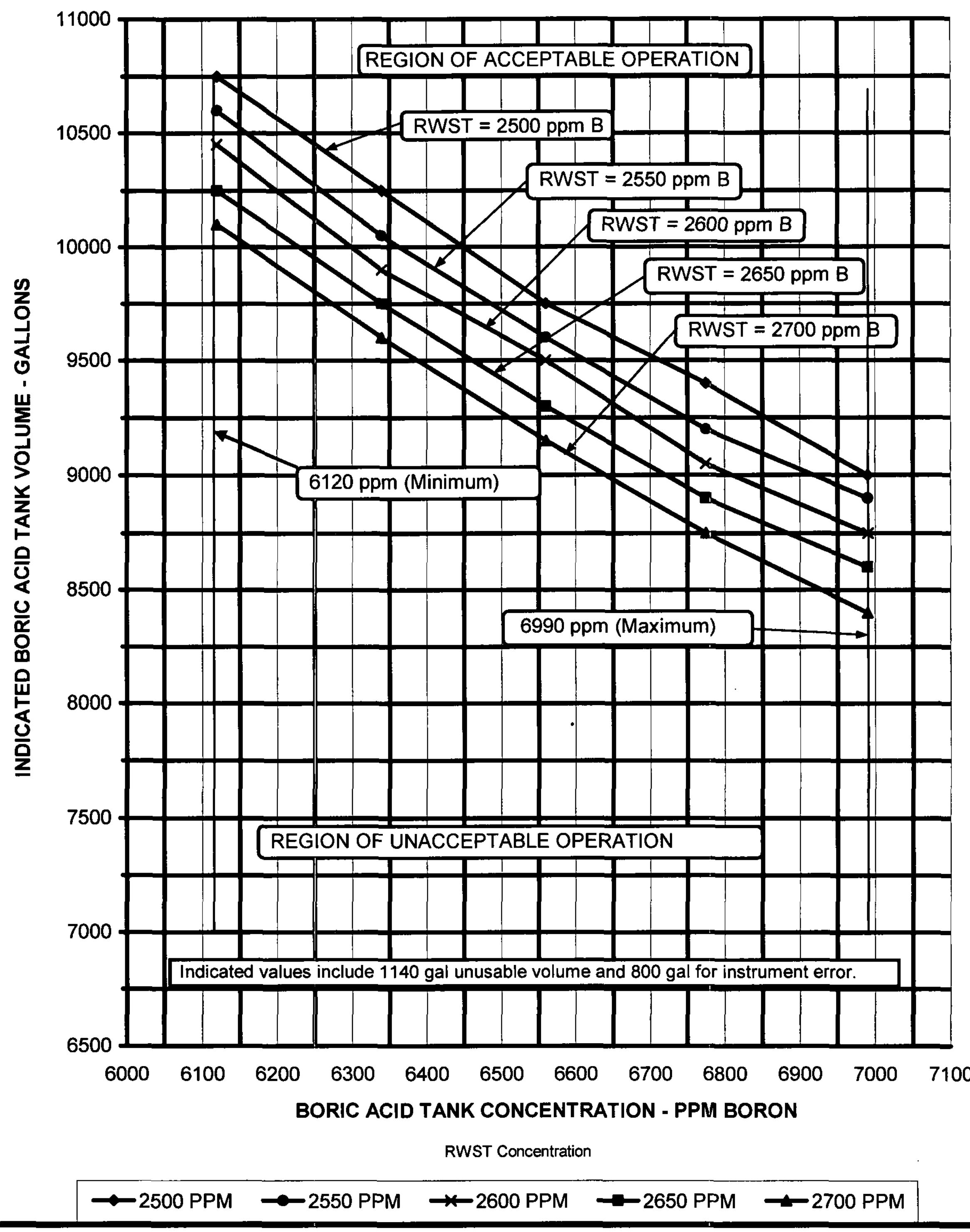
   2. Verifying the borated water volume at least once per 7 days, and

   3. Verifying the solution temperature at least once per 24 hours.

TRM FIGURE 3.1.2.6 (Units 1 & 2)

BORIC ACID TANK LIMITS

BASED ON RWST BORON CONCENTRATION



TR 3/4.1 REACTIVITY CONTROL SYSTEMS

TR 3/4 1.3.1 No current requirements

TR 3/4 1.3.2 No current requirements

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

TR 3.1.3.3 The group demand position indicator shall be OPERABLE and capable of determining within ± 2 steps, the demand position for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*, 4* and 5*.

ACTION:

With less than the above required group demand position indicator(s) OPERABLE, immediately open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

TR 4.1.3.3 Each of the above required group demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days.

*With the reactor trip system breakers in the closed position.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

TR 3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$.

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification TR 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

TR 4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$.

TRB 3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at near EOL from full power peak xenon conditions and requires borated water from a boric acid tank in accordance with Figure 3.1.2.6, and additional makeup from either: (1) the common boric acid tank and/or batching, or (2) a minimum of 26,000 gallons of 2500 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a minimum of 57,000 gallons of 2500 ppm borated water is required.

TR 3.1.2.4 and TR 3.1.2.2 are modified by a Note. Operation in MODE 3 with one charging pump made incapable of injecting, in order to facilitate entry into or exit from the Applicability of Technical Specification LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. Technical Specification LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make a pump incapable of injecting prior to entering the LTOP Applicability, and provide time to restore the inoperable pump to OPERABLE status on exiting the LTOP Applicability.

The boric acid tanks, pumps, valves, and piping contain a boric acid solution concentration of between 3.5% and 4.0% by weight. To ensure that the boric acid remains in solution, the air temperature is monitored in strategic locations. By ensuring the air temperature remains at 63°F or above, a 5°F margin is provided to ensure the boron will not precipitate out. To provide operational flexibility, if the area temperature should fall below the required value, the solution temperature (as determined by the pipe or tank wall temperature) will be monitored at an increased frequency to compensate for the lack of solution temperature alarm in the main control room.

With the RCS temperature below 350°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and operations involving positive reactivity additions that could result in loss of required SDM (Modes 4 or 5) or boron concentration (Mode 6) in the event the single injection system becomes inoperable. Suspending positive reactivity additions that could result in failure to meet minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than or equal to that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

The boron capability required below 350°F, is sufficient to provide a SHUTDOWN MARGIN of 1.6% delta k/k after xenon decay and cooldown from 350°F to 200°, and a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 6400 gallons of 6120 ppm borated water from the boric acid storage tanks or 13,400 gallons of 2500 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The 6400 gallon limit in the boric acid tank for Modes 4, 5, and 6 is based on 4,431 gallons required for shutdown margin, 1,140 gallons for the unusable volume in the heel of the tank, 800 gallons for instrument error, and an additional 29 gallons due to rounding up. The 55,000 gallon limit in the refueling water storage tank for modes 4, 5, and 6 is based upon 22,182 gallons that is undetectable due to lower tap location, 19,197 gallons for instrument error, 13,400 gallons required for shutdown margin, and an additional 221 gallons due to rounding up.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

ATTACHMENT 2

SEQUOYAH NUCLEAR PLANT, UNIT 1
TECHNICAL SPECIFICATION BASES
CHANGED PAGES

TS Bases Affected Pages

EPL Page 18
EPL Page 20
EPL Page 21
EPL Page 22
EPL Page 25
EPL Page 26
EPL Page 27
EPL Page 28
EPL Page 29
EPL Page 30
EPL Page 31
EPL Page 32
EPL Page 33
EPL Page 34
Index Page XIV
B 3/4 3-3a
B 3/4 4-12
B 3/4 5-12
B 3/4 5-13
B 3/4 5-14
B 3/4 5-15
B 3/4 5-16
B 3/4 5-17
B 3/4 5-18
B 3/4 5-19
B 3/4 5-20
B 3/4 7-4a
B 3/4 7-4b
B 3/4 8-1
B 3/4 8-2
B 3/4 8-3
B 3/4 8-4
B 3/4 8-5
B 3/4 8-6
B 3/4 8-7
B 3/4 8-8
B 3/4 8-9

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 3-3	12/28/05
B3/4 3-3a	03/05/13
B3/4 3-4	08/12/97
B3/4 3-5 through B3/4 3-9	09/14/06
B3/4 4-1	03/09/05
B3/4 4-2	06/16/06
B3/4 4-2a	02/23/06
B3/4 4-3	02/23/06
B3/4 4-3a	05/18/06
B3/4 4-3b	02/23/06
B3/4 4-3c through B3/4 4-3d	05/18/06
B3/4 4-4	02/23/06
B3/4 4-4a	02/23/06
B3/4 4-4b	12/04/08
B3/4 4-4c	12/04/08
B3/4 4-4d	04/11/05
B3/4 4-4e	12/04/08
B3/4 4-4f	12/04/08
B3/4 4-4g	05/18/06
B3/4 4-4h	05/18/06
B3/4 4-4i	02/23/06
B3/4 4-4j	02/23/06
B3/4 4-4k	02/23/06
B3/4 4-4l	02/23/06
B3/4 4-4m	08/04/00
B3/4 4-4n	08/04/00

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 4-23	11/09/04
B3/4 5-1	03/25/10
B3/4 5-2	03/25/10
B3/4 5-3	03/25/10
B3/4 5-4	03/25/10
B3/4 5-5	03/25/10
B3/4 5-6	03/25/10
B3/4 5-7	03/25/10
B3/4 5-8 through B3/4 5-11	03/25/10
B3/4 5-12 through B3/4 5-20	03/24/12
B3/4 6-1 through B3/4 6-2	04/13/09
B3/4 6-3	05/27/10
B3/4 6-4 through B3/4 6-6	04/13/09
B3/4 6-7 through B3/4 6-12	04/13/09
B3/4 6-13 through B3/4 6-18	04/13/09
B3/4 6-19 through B3/4 6-20	04/13/09
B3/4 6-21	04/13/09
B3/4 7-1	04/30/02
B3/4 7-2	08/14/01
B3/4 7-2a	11/17/95

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 7-2b	04/11/05
B3/4 7-3	06/12/09
B3/4 7-3a	06/08/98
B3/4 7-4	09/28/07
B3/4 7-4a	10/05/12
B3/4 7-4b	10/05/12
B3/4 7-4c thru B3/4 7-4m	10/28/08
B3/4 7-5	08/18/05
B3/4 7-6 (DELETED)	08/28/98
B3/4 7-7	12/28/05
B3/4 7-8	08/12/97
B3/4 7-9	12/19/00
B3/4 7-10	12/19/00
B3/4 7-11	12/19/00
B3/4 7-12	12/19/00
B3/4 7-13	12/19/00
B3/4 7-14	12/19/00
B3/4 7-15	12/19/00
B3/4 7-16	01/31/05
B3/4 7-17	02/27/02
B3/4 7-18	02/27/02
B3/4 8-1	12/21/12
B3/4 8-2	12/21/12
B3/4 8-3	12/21/12
B3/4 8-4	12/21/12
B3/4 8-5	12/21/12

*Original pages are not dated (2/29/80).

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 8-6	12/21/12
B3/4 8-7	12/21/12
B3/4 8-8	12/21/12
B3/4 8-9	12/21/12
B3/4 9-1	09/20/04
B3/4 9-2	12/28/05
B3/4 9-3	04/19/04
B3/4 10-1	09/20/04
B3/4 11-1	12/09/93
B3/4 11-2	11/16/90
B3/4 12-1	11/16/90
5-1	08/02/06
5-2	08/02/06
5-3	08/02/06
5-4	08/02/06
5-5	12/19/00
5-5a	12/19/00
5-5b	08/02/06
5-5c	12/19/00
5-5d	12/19/00
5-5e	12/19/00

EFFECTIVE PAGE LISTING

Appendix C

Appendix C Cover Sheet	09/06/12
Table of Contents	09/06/12
C-1	09/06/12
C-2	09/06/12

AMENDMENT LISTING

Amendments	Revisions
Lower Power License and Technical Specifications issued	02/29/80 (Original)
Amendment 1 issued by NRC	04/22/80 (R1)
Amendment 2 issued by NRC	07/01/80 (R2)
Amendment 3 issued by NRC	07/01/80 (R2)
Amendment 4 issued by NRC	07/10/80 (R2)
Full Power License and Technical Specifications issued	09/17/80 (R3)
NRC Order issued	11/06/80 (R4)
* Amendment 1 issued by NRC	12/22/80 (R5)
Amendment 2 issued by NRC	02/09/81 (R6)
Amendment 3 issued by NRC	02/13/81 (R6)
Amendment 4 issued by NRC	03/06/81 (R7)
Amendment 5 issued by NRC	04/15/81 (R8)
Amendment 6 issued by NRC	04/27/81 (R9)
EPL Revised	06/01/81 (R10)
Amendment 7 issued by NRC	06/26/81 (R11)
Amendment 8 issued by NRC	07/15/81 (R12)
Amendment 9 issued by NRC	09/08/81 (R13)
Amendment 10 issued by NRC	12/31/81 (R14)
Amendment 11 issued by NRC	01/29/82 (R15)
Amendment 12 issued by NRC	03/25/82 (R16)
Amendment 13 issued by NRC	05/04/82 (R17)
Amendment 14 issued by NRC	06/18/82 (R18)
Amendment 15 issued by NRC	08/03/82 (R19)
Amendment 16 issued by NRC	10/21/82 (R20)
Amendment 17 issued by NRC	12/23/82 (R21)
Amendment 18 issued by NRC	12/23/82 (R22)
Amendment 19 issued by NRC	12/23/82 (R23)
Amendment 20 issued by NRC	12/23/82 (R24)
Amendment 21 issued by NRC	12/23/82 (R25)
Amendment 22 issued by NRC	12/27/82 (R26)
Amendment 23 issued by NRC	12/27/82 (R27)
Amendment 24 issued by NRC	12/29/82 (R28)
Amendment 25 issued by NRC	12/29/82 (R29)
Amendment 26 issued by NRC	03/14/83 (R30)
Amendment 27 issued by NRC	03/16/83 (R31)
Amendment 28 issued by NRC	05/03/83 (R32)
Amendment 29 issued by NRC	05/05/83 (R33)

*Amendments to Full Power License

AMENDMENT LISTING

Amendments	Revisions
Amendment 30 issued by NRC	05/05/83 (R34)
Amendment 31 issued by NRC	09/30/83 (R35)
Amendment 32 issued by NRC	11/10/83 (R36)
Amendment 33 issued by NRC	03/29/84 (R37)
Amendment 34 issued by NRC	04/12/84 (R38)
Amendment 35 issued by NRC	04/24/84 (R39)
Amendment 36 issued by NRC	11/23/84 (R40)
Amendment 37 issued by NRC	01/24/85 (R41)
Amendment 38 issued by NRC	06/11/85 (R42)
Amendment 39 issued by NRC	06/20/85 (R43)
Amendment 40 issued by NRC	06/25/85 (R44)
Amendment 41 issued by NRC	09/03/85 (R45)
Amendment 42 issued by NRC	01/14/86 (R46)
Amendment 43 issued by NRC	01/29/86 (R47)
EPL Revised	07/17/86
Amendment 44 issued by NRC	07/28/86 (R48)
Amendment 45 issued by NRC	09/15/86 (R49)
Amendment 46 issued by NRC	09/16/86 (R50)
Amendment 47 issued by NRC	09/17/86 (R51)
Amendment 48 issued by NRC	10/02/86 (R52)
Amendment 49 issued by NRC	10/28/86 (R53)
Amendment 50 issued by NRC	12/01/86 (R54)
Amendment 51 issued by NRC	12/02/86 (R55)
Special Amendment Authorized by NRC**	01/05/87 (SR)
Amendment 52 issued by NRC	02/03/87 (R56)
Amendment 53 issued by NRC	02/12/87 (R57)
Amendment 54 issued by NRC	03/16/87 (R58)
Amendment 55 issued by NRC	05/12/87 (R59)
Amendment 56 issued by NRC	07/20/87 (R60)
Bases Revision issued by NRC	08/18/87 (BR)
Amendment 57 issued by NRC	09/09/87 (R61)
Amendment 58 issued by NRC	09/10/87 (R62)
Amendment 59 issued by NRC	09/18/87 (R63)
Amendment 60 issued by NRC	10/19/87 (R64)
Amendment 61 issued by NRC	10/22/87 (R65)
Amendment 62 issued by NRC	10/30/87 (R66)
Amendment 63 issued by NRC	12/31/87 (R67)
Amendment 64 issued by NRC	01/07/88 (R68)
Amendment 65 issued by NRC	01/11/88 (R69)
Amendment 66 issued by NRC	01/25/88 (R70)
Additional Exemptions issued by NRC	01/15/88 (AE)
Amendment 67 issued by NRC	02/11/88 (R71)

** Special Revision authorized by 10 CFR Parts 50 and 51 Final Rule as noted in the Federal Register on November 6, 1986 and effective January 5, 1987.

AMENDMENT LISTING

Amendments	Revisions
Amendment 68 issued by NRC	02/17/88 (R72)
Amendment 69 issued by NRC	04/04/88 (R73)
Amendment 70 issued by NRC	05/16/88 (R74)
Amendment 71 issued by NRC	05/18/88 (R75)
Amendment 72 issued by NRC	05/23/88 (R76)
Amendment 73 issued by NRC	06/24/88 (R77)
Amendment 74 issued by NRC	06/30/88 (R78)
Amendment 75 issued by NRC	07/06/88 (R79)
Amendment 76 issued by NRC	07/12/88 (R80)
Amendment 77 issued by NRC	08/05/88 (R81)
Amendment 78 issued by NRC	08/15/88 (R82)
Amendment 79 issued by NRC	08/15/88 (R83)
Amendment 80 issued by NRC	08/16/88 (R84)
Amendment 81 issued by NRC	09/01/88 (R85)
Amendment 82 issued by NRC	09/09/88 (R86)
Amendment 83 issued by NRC	09/21/88 (R87)
Amendment 84 issued by NRC	09/22/88 (R88)
Amendment 85 issued by NRC	09/22/88 (R89)
Amendment 86 issued by NRC	10/14/88 (R90)
Amendment 87 issued by NRC	10/14/88 (R91)
Amendment 88 issued by NRC	10/14/88 (R92)
Amendment 89 issued by NRC	10/14/88 (R93)
Amendment 90 issued by NRC	10/14/88 (R94)
Amendment 91 issued by NRC	12/05/88 (R95)
Amendment 92 issued by NRC	12/29/88 (R96)
Amendment 93 issued by NRC	12/29/88 (R97)
Amendment 94 issued by NRC	12/30/88 (R98)
Amendment 95 issued by NRC	01/23/89 (R99)
Amendment 96 issued by NRC	01/22/89 (R100)
Amendment 97 issued by NRC	01/22/89 (R101)
Amendment 98 issued by NRC	01/30/89 (R102)
Amendment 99 issued by NRC	01/30/89 (R103)
Amendment 100 issued by NRC	01/31/89 (R104)
Amendment 101 issued by NRC	02/28/89 (R105)
Amendment 102 issued by NRC	03/02/89 (R106)
Amendment 103 issued by NRC	03/06/89 (R107)
Amendment 104 issued by NRC	03/09/89 (R108)
Amendment 105 issued by NRC	03/09/89 (R109)
Amendment 106 issued by NRC	03/13/89 (R110)
Amendment 107 issued by NRC	03/15/89 (R111)
Amendment 108 issued by NRC	03/28/89 (R112)
Amendment 109 issued by NRC	04/03/89 (R113)
Amendment 110 issued by NRC	04/03/89 (R114)
Amendment 111 issued by NRC	04/03/89 (R115)
Amendment 112 issued by NRC	04/28/89 (R116)

AMENDMENT LISTING

Amendments	Revisions
Amendment 113 issued by NRC	05/04/89 (R117)
Amendment 114 issued by NRC	05/05/89 (R118)
Amendment 115 issued by NRC	05/11/89 (R119)
Amendment 116 issued by NRC	06/01/89 (R120)
Amendment 117 issued by NRC	06/19/89 (R121)
Amendment 118 issued by NRC	06/23/89 (R122)
Amendment 119 issued by NRC	07/05/89 (R123)
Amendment 120 issued by NRC	07/05/89 (R124)
Amendment 121 issued by NRC	07/31/89 (R125)
Amendment 122 issued by NRC	08/03/89 (R126)
Amendment 123 issued by NRC	08/11/89 (R127)
Amendment 124 issued by NRC	08/11/89 (R128)
Amendment 125 issued by NRC	08/14/89 (R129)
Amendment 126 issued by NRC	09/19/89 (R130)
Amendment 127 issued by NRC	09/29/89 (R131)
Amendment 128 issued by NRC	11/01/89 (R132)
Amendment 129 issued by NRC	11/28/89 (R133)
Amendment 130 issued by NRC	02/16/90 (R134)
Amendment 131 issued by NRC	03/02/90 (R135)
Amendment 132 issued by NRC	03/19/90 (R136)
Amendment 133 issued by NRC	03/22/90 (R137)
Bases Revision	03/23/90 (BR-1)
Amendment 134 issued by NRC	04/02/90 (R138)
Amendment 135 issued by NRC	04/27/90 (R139)
Amendment 136 issued by NRC	04/27/90 (R140)
Amendment 137 issued by NRC	04/27/90 (R141)
Bases Revision	05/01/90 (BR-2)
Amendment 138 issued by NRC	05/08/90 (R142)
Amendment 139 issued by NRC	05/09/90 (R143)
Amendment 140 issued by NRC	05/11/90 (R144)
Amendment 141 issued by NRC	05/16/90 (R145)
Amendment 142 issued by NRC	07/27/90 (R146)
Amendment 143 issued by NRC	07/31/90 (R147)
Amendment 144 issued by NRC	08/31/90 (R148)
Amendment 145 issued by NRC	09/20/90 (R149)
Amendment 146 issued by NRC	09/21/90 (R150)
Amendment 147 issued by NRC	11/02/90 (R151)
Amendment 148 issued by NRC	11/16/90 (R152)
Amendment 149 issued by NRC	12/07/90 (R153)
Amendment 150 issued by NRC	03/18/91 (R154)
Amendment 151 issued by NRC	07/24/91 (R155)
Amendment 152 issued by NRC	08/22/91 (R156)
Amendment 153 issued by NRC	09/10/91 (R157)
Amendment 154 issued by NRC	10/18/91 (R158)
Amendment 155 issued by NRC	10/23/91 (R159)
Amendment 156 issued by NRC	12/16/91 (R160)
Amendment 157 issued by NRC	03/30/92 (R161)
Amendment 158 issued by NRC	03/31/92 (R162)
Amendment 159 issued by NRC	07/09/92 (R163)
Amendment 160 issued by NRC	07/24/92 (R164)

AMENDMENT LISTING

Amendments	Revisions
Amendment 161 issued by NRC	08/10/92 (R165)
Amendment 162 issued by NRC	08/13/92 (R166)
Amendment 163 issued by NRC	09/28/92 (R167)
Amendment 164 issued by NRC	11/06/92 (R168)
Bases Revision	11/25/92 (BR-3)
Amendment 165 issued by NRC	12/08/92 (R169)
Bases Revision	12/08/92 (BR-4)
Amendment 166 issued by NRC	01/12/93 (R170)
Amendment 167 issued by NRC	04/28/93 (R171)
Amendment 168 issued by NRC	06/25/93 (R172)
Amendment 169 issued by NRC	08/02/93 (R173)
Amendment 170 issued by NRC	08/27/93 (R174)
Amendment 171 issued by NRC	10/26/93 (R175)
Amendment 172 issued by NRC	11/26/93 (R176)
Amendment 173 issued by NRC	11/29/93 (R177)
Amendment 174 issued by NRC	12/09/93 (R178)
Amendment 175 issued by NRC	01/03/94 (R179)
Amendment 176 issued by NRC	02/10/94 (R180)
Amendment 177 issued by NRC	03/15/94 (R181)
Amendment 178 issued by NRC	03/31/94 (R182)
Amendment 179 issued by NRC	04/18/94 (R183)
Amendment 180 issued by NRC	04/18/94 (R184)
Amendment 181 issued by NRC	05/23/94 (R185)
Amendment 182 issued by NRC	05/24/94 (R186)
Amendment 183 issued by NRC	05/27/94 (R187)
Amendment 184 issued by NRC	07/11/94 (R188)
Amendment 185 issued by NRC	07/26/94 (R189)
Amendment 186 issued by NRC	09/13/94 (R190)
Amendment 187 issued by NRC	10/17/94 (R191)
Amendment 188 issued by NRC	10/17/94 (R192)
Amendment 189 issued by NRC	10/20/94 (R193)
Amendment 190 issued by NRC	11/09/94 (R194)
Amendment 191 issued by NRC	11/22/94 (R195)
Amendment 192 issued by NRC	12/27/94 (R196)
Amendment 193 issued by NRC	01/03/95 (R197)
Amendment 194 issued by NRC	01/24/95 (R198)
Amendment 195 issued by NRC	02/09/95 (R199)
Bases Revision	03/02/95 (BR-5)
Amendment 196 issued by NRC	04/04/95 (R200)
Amendment 197 issued by NRC	04/28/95 (R201)
Amendment 198 issued by NRC	05/10/95 (R202)
Amendment 199 issued by NRC	05/30/95 (R203)
Amendment 200 issued by NRC	05/30/95 (R204)
Amendment 201 issued by NRC	06/01/95 (R205)
Amendment 202 issued by NRC	06/13/95 (R206)
Amendment 203 issued by NRC	06/13/95 (R207)
Amendment 204 issued by NRC	06/14/95 (R208)
Amendment 205 issued by NRC	06/29/95 (R209)
Amendment 206 issued by NRC	08/02/95 (R210)
Bases Revision	08/11/95 (BR-6)

AMENDMENT LISTING

Amendments	Revisions
Amendment 207 issued by NRC	08/22/95 (R211)
Amendment 208 issued by NRC	08/22/95 (R212)
Amendment 209 issued by NRC	09/06/95 (R213)
Amendment 210 issued by NRC	09/13/95 (R214)
Amendment 211 issued by NRC	09/15/95 (R215)
Amendment 212 issued by NRC	10/02/95 (R216)
Amendment 213 issued by NRC	10/04/95 (R217)
Amendment 214 issued by NRC	10/11/95 (R218)
Bases Revision	10/27/95 (BR-7)
Bases Revision	11/17/95 (BR-8)
Amendment 215 issued by NRC	11/21/95 (R219)
Amendment 216 issued by NRC	12/11/95 (R220)
Amendment 217 issued by NRC	02/05/96 (R221)
Amendment 218 issued by NRC	02/07/96 (R222)
Bases Revision	02/15/96 (BR-9)
Amendment 219 issued by NRC	03/01/96 (R223)
Amendment 220 issued by NRC	03/04/96 (R224)
Amendment 221 issued by NRC	04/26/96 (R225)
Bases Revision	09/13/96 (BR-10)
Bases Revision	01/02/97 (BR-11)
Amendment 222 issued by NRC	04/09/97 (R226)
Amendment 223 issued by NRC	04/21/97 (R227)
Amendment 224 issued by NRC	06/10/97 (R228)
Amendment 225 issued by NRC	07/01/97 (R229)
Amendment 226 issued by NRC	07/14/97 (R230)
Amendment 227 issued by NRC	08/12/97 (R231)
Amendment 228 issued by NRC	09/23/97 (R232)
Amendment 229 issued by NRC	09/29/97 (R233)
Amendment 230 issued by NRC	01/13/98 (R234)
Amendment 231 issued by NRC	02/20/98 (R235)
Amendment 232 issued by NRC	06/08/98 (R236)
Amendment 233 issued by NRC	07/01/98 (R237)
Amendment 234 issued by NRC	07/22/98 (R238)
Amendment 235 issued by NRC	08/28/98 (R239)
Bases Revision	09/09/98 (BR-12)
Amendment 236 issued by NRC	11/17/98 (R240)
Amendment 237 issued by NRC	11/17/98 (R241)
Amendment 238 issued by NRC	11/19/98 (R242)
Amendment 239 issued by NRC	11/19/98 (R243)
Amendment 240 issued by NRC	12/07/98 (R244)
Amendment 241 issued by NRC	12/16/98 (R245)
Bases Revision	01/25/99 (BR-13)
Amendment 242 issued by NRC	02/09/99 (R246)
Amendment 243 issued by NRC	03/16/99 (R247)
Amendment 244 issued by NRC	05/04/99 (R248)
Amendment 245 issued by NRC	09/07/99 (R249)
Amendment 246 issued by NRC	09/23/99 (R250)
Amendment 247 issued by NRC	10/06/99 (R251)

AMENDMENT LISTING

Amendments	Revisions
Amendment 248 issued by NRC	10/12/99 (R252)
Amendment 249 issued by NRC	02/11/00 (R253)
Amendment 250 issued by NRC	02/22/00 (R254)
Amendment 251 issued by NRC	02/29/00 (R255)
Amendment 252 issued by NRC	03/08/00 (R256)
Amendment 253 issued by NRC	03/29/00 (R257)
Amendment 254 issued by NRC	03/29/00 (R258)
Amendment 255 issued by NRC	04/14/00 (R259)
Bases Revision	05/25/00 (BR-14)
Bases Revision	05/25/00 (BR-15)
Amendment 256 issued by NRC	05/31/00 (R260)
Amendment 257 issued by NRC	07/18/00 (R261)
Amendment 258 issued by NRC	07/31/00 (R262)
Amendment 259 issued by NRC	08/04/00 (R263)
Amendment 260 issued by NRC	08/28/00 (R264)
Amendment 261 issued by NRC	10/02/00 (R265)
Amendment 262 issued by NRC	10/06/00 (R266)
Amendment 263 issued by NRC	11/02/00 (R267)
Amendment 264 issued by NRC	12/18/00 (R268)
Amendment 265 issued by NRC	12/19/00 (R269)
Amendment 266 issued by NRC	02/16/01 (R270)
Amendment 267 issued by NRC	03/22/01 (R271)
Amendment 268 issued by NRC	05/09/01 (R272)
Bases Revision	06/28/01 (BR-16)
Bases Revision	07/11/01 (BR-17)
Amendment 269 issued by NRC	07/12/01 (R273)
Amendment 270 issued by NRC	07/18/01 (R274)
Bases Revision	07/20/01 (BR-18)
Bases Revision	08/14/01 (BR-19)
Amendment 271 issued by NRC	10/24/01 (R275)
Amendment 272 issued by NRC	01/14/02 (R276)
Bases Revision	02/14/02 (BR-20)
Amendment 273 issued by NRC	02/27/02 (R277)
Amendment 274 issued by NRC	03/08/02 (R278)
Amendment 275 issued by NRC	04/30/02 (R279)
Bases Revision	05/17/02 (BR-21)
Amendment 276 issued by NRC	05/24/02 (R280)
Amendment 277 issued by NRC	09/05/02 (R281)
Amendment 279 issued by NRC	09/30/02 (R283)
Amendment 280 issued by NRC	02/05/03 (R284)
Amendment 281 issued by NRC	02/11/03 (R285)
Amendment 282 issued by NRC	03/04/03 (R286)
Amendment 283 issued by NRC	04/24/03 (R287)
Amendment 284 issued by NRC	04/25/03 (R288)
Amendment 285 issued by NRC	05/22/03 (R289)
Bases Revision	05/22/03 (BR-22)
Amendment 286 issued by NRC	05/27/03 (R290)
Amendment 287 issued by NRC	05/29/03 (R291)
Bases Revision	06/26/03 (BR-23)

AMENDMENT LISTING

Amendments	Revisions
Amendment 288 issued by NRC	10/28/03 (R292)
Bases Revision	12/22/03 (BR-24)
Bases Revision	04/19/04 (BR-25)
Amendment 290 issued by NRC	04/21/04 (R294)
Amendment 291 issued by NRC	06/18/04 (R295)
Amendment 292 issued by NRC	05/21/04 (R296)
Amendment 293 issued by NRC	07/08/04 (R297)
Amendment 294 issued by NRC	09/15/04 (R298)
Amendment 295 issued by NRC	09/20/04 (R299)
Amendment 296 issued by NRC	09/20/04 (R300)
License Condition Issued by NRC	10/28/04
Amendment 297 issued by NRC	11/09/04 (R301)
Bases Revision	10/13/04 (BR-26)
Amendment 298 issued by NRC	01/31/05 (R302)
Bases Revision	02/25/05 (BR-27)
Bases Revision	03/04/05 (BR-28)
Amendment 299 issued by NRC	03/09/05 (R303)
Amendment 300 issued by NRC	04/05/05 (R304)
Amendment 301 issued by NRC	04/11/05 (R305)
Amendment 302 issued by NRC	05/24/05 (R306)
Amendment 303 issued by NRC	08/18/05 (R307)
Amendment 304 issued by NRC	09/02/05 (R308)
Amendment 305 issued by NRC	12/28/05 (R309)
Amendment 306 issued by NRC	02/23/06 (R310)
Amendment 307 issued by NRC	04/06/06 (R311)
Bases Revision	05/18/06 (BR-29)
Amendment 308 issued by NRC	06/16/06 (R312)
Amendment 309 issued by NRC	08/02/06 (R313)
Amendment 310 issued by NRC	09/13/06 (R314)
Amendment 311 issued by NRC	09/14/06 (R315)
Amendment 312 issued by NRC	10/04/06 (R316)
Amendment 313 issued by NRC	11/07/06 (R317)
Amendment 314 issued by NRC	11/16/06 (R318)
Amendment 315 issued by NRC	12/11/06 (R319)
License Condition Issued by NRC	02/08/07
Bases Revision	03/07/07 (BR-30)
License Condition Issued by NRC	08/09/07 (B.5.b)
Amendment 316 issued by NRC	09/20/07 (R320)
Amendment 317 issued by NRC	09/28/07 (R321)
Bases Revision	12/12/07 (BR-31)
Amendment 318 issued by NRC	04/02/08 (R322)
Bases Revision	08/29/08 (BR-32)
Amendment 319 issued by NRC	08/29/08 (R323)
Bases Revision	08/28/08 (BR-33)
Amendment 320 issued by NRC	09/24/08
Amendment 321 issued by NRC	10/28/08
Amendment 322 issued by NRC	12/04/08
Amendment 323 issued by NRC	04/13/09
Amendment 324 issued by NRC	06/12/09
Bases Revision	06/12/09 (BR-34)
Amendment 325 issued by NRC	08/14/09
Amendment 326 issued by NRC	01/28/10
Amendment 327 issued by NRC	02/02/10

AMENDMENT LISTING

Amendments	Revisions
Bases Revision	03/25/10 (BR-35)
Bases Revision	05/27/10 (BR-36)
Amendment 328 issued by NRC	12/21/10
Bases Revision	03/24/12 (BR-38)
Amendment 330 issued by NRC	09/06/12
Bases Revision	10/05/12 (BR-39)
Amendment 332 issued by NRC	10/31/12
Bases Revision	12/21/12 (BR-41)
Bases Revision	03/05/13 (BR-42)

SECTION PAGE

3/4.7.4 ESSENTIAL RAW COOLING WATER SYSTEM .... B 3/4 7-3a

3/4.7.5 ULTIMATE HEAT SINK (UHS) .... B 3/4 7-4

3/4.7.6 FLOOD PROTECTION .... B 3/4 7-4

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM .... B 3/4 7-4

3/4.7.8 AUXILIARY BUILDING GAS TREATMENT SYSTEM .... B 3/4 7-5

3/4.7.9 SNUBBERS (Deleted) .... B 3/4 7-5

3/4.7.10 SEALED SOURCE CONTAMINATION (Deleted) .... B 3/4 7-7

3/4.7.11 FIRE SUPPRESSION SYSTEMS (Deleted) .... B 3/4 7-7

3/4.7.12 FIRE BARRIER PENETRATIONS (Deleted) .... B 3/4 7-8

3/4.7.13 SPENT FUEL POOL MINIMUM BORON CONCENTRATION .... B 3/4 7-9

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION .... B 3/4 7-13

3/4.7.15 CONTROL ROOM AIR-CONDITIONING SYSTEM (CRACS) .... B 3/4 7-16

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS .... B 3/4 8-1

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES (Deleted) .... B 3/4 8-9

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION .... B 3/4 9-1

3/4.9.2 INSTRUMENTATION .... B 3/4 9-1

3/4.9.3 DECAY TIME .... B 3/4 9-1

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS .... B 3/4 9-1

3/4.9.5 COMMUNICATIONS (Deleted) .... B 3/4 9-2

3/4.9.6 MANIPULATOR CRANE (Deleted) .... B 3/4 9-2

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT AREA (Deleted) .... B 3/4 9-2

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION .... B 3/4 9-2

3/4.9.9 CONTAINMENT VENTILATION SYSTEM .... B 3/4 9-3

ACCIDENT MONITORING INSTRUMENTATION (Continued)

- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release and to determine if a gross breach of a barrier has occurred.

For Sequoyah, the redundant channel capability for Auxiliary Feedwater (AFW) flow consists of a single AFW flow channel for each Steam Generator with the second channel consisting of three AFW valve position indicators (two level control valves for the motor driven AFW flowpath and one level control valve for the turbine driven AFW flowpath) for each steam generator.

TABLE B 3/4.4-1

SEQUOYAH-UNIT 1 REACTOR VESSEL TOUGHNESS DATA

COMPONENT	HEAT NO.	MATERIAL GRADE	Cu (%)	Ni (%)	NDT (°F)	MINIMUM 50 ft-lb/35 mil temp. TEMP.(°F) PMWD1	NMWD2	RT_{NDT} (°F)	AVERAGE UPPER SHELF ENERGY (ft-lb) PMWD1	NMWD2
Clos Hd. Dome	52841-1	A533B,C1.1	-	-	-40	+14	+34	-26	104[a]	-
Clos Hd. Ring	(D75600)	A508,C1.2	-	-	+ 5	+36	+56*	+5	125[a]	-
Hd Flange	4842	A508,C1.2	-	-	-40	-24	-4*	-40	131[a]	-
Vessel Flange	4866	A508,C1.2	-	-	-49	-47	-27	-49	158[a]	-
Inlet Nozzle	4846	A508,C1.2	-	-	-58	+25	+45	-15	94.5[a]	-
Inlet Nozzle	4949	A508,C1.2	-	-	-40	+39	+59*	-1	93[a]	-
Inlet Nozzle	4863	A508,C1.2	-	-	-22	+16	+36*	-22	118[a]	-
Inlet Nozzle	4865	A508,C1.2	-	-	-67	+ 9	+29*	-31	94[a]	-
Outlet Nozzle	4845	A508,C1.2	-	-	-49	+21	+41*	-19	94[a]	-
Outlet Nozzle	4850	A508,C1.2	-	-	-58	+30	+50*	-10	79.5[a]	-
Outlet Nozzle	4862	A508,C1.2	-	-	-58	+16	+36*	-24	103[a]	-
Outlet Nozzle	4864	A508,C1.2	-	-	-49	0	+20	-40	126[a]	-
Upper Shell	4841	A508,C1.2	-	-	-40	+43	+83	+23	83[a]	113[b]
Inter Shell	4829	A508,C1.2	0.15	0.86	-4	+10	+100	+40	116	73[b,c]
Lower Shell	4836	A508,C1.2	0.13	0.76	+5	+28	+133	+73	109	70[b]
Trans. Ring	4879	A508,C1.2	-	-	+5	+27	+47*	+ 5	98[a]	-
Bot. Hd. Rim	52703/2-1	A533B,C1.1	-	-	-31	+23	+43*	-17	104[a]	-
Bot. Hd. Rim	52703/2-2	A533B,C1.1	-	-	-13	+36	+56*	-4	63[a]	
Bot. Hd. Rim	52704/2	A533B,C1.1	-	-	-49	-24	-4*	-49	114[a]	
Bot. Hd. Rim	52703/2-2	A533B,C1.1	-	-	-31	+43	+63*	+3	86[a]	
Bot. Hd. Rim	52704/2	A533B,C1.1	-	-	-58	-13	+4	-53	120[a]	
Bot. Hd.	52704/11	A533B,C1.1	-	-	-58	-47	-27*	-58	139[a]	
Weld	-	Weld	0.33	0.17	-40	-	-4	-40	-	116[b]
HAZ	-	Weld	-	-	-22	-	+41	-19	-	86[b]

1-Paralled to Major Working Direction

2-Normal to Major Working Direction

* Estimate based on USAEC Regulatory Standard Review Plan, Section 5.3.2 MTEB

a-%Shear Not reported

b-Minimum upper shelf energies

c-Minimum upper shelf energy decreased to 51 at a test temperature of 300°F. This anomaly will be reevaluted when the results of Generic task A-11 are available.

B 3/4.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3/4.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications. In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head). For the RHR subsystem during the injection phase, water is taken from the refueling storage tank (RWST) and injected in the Reactor Coolant System (RCS) through at least two cold legs.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation.

One train of ECCS during the injection phase provides sufficient flow for core cooling, by the centrifugal charging subsystem supplying each of the four cold legs and the RHR subsystem supplying at least two cold legs, to meet the analysis requirements for a credible Mode 4 Loss of Coolant Accident (LOCA.)

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

LCO (continued)

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to the cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.

Either RHR cold leg injection valve FCV-63-93 or FCV-63-94 may be closed when in Mode 4, for testing of the primary/secondary check valves in the injection lines. Closing one of the two cold leg injection flow paths does not make ECCS RHR subsystem inoperable.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. The manual actions necessary to realign the RHR subsystem may include actions to cool the RHR system piping due to the potential for steam voiding in piping or for inadequate NPSH available at the RHR pumps. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.1.4, "Reactor Coolant System Cold Shutdown." MODE 6 core cooling requirements are addressed by LCO 3.9.8.1 “Residual Heat Removal and Coolant Circulation - All Water Levels,” and LCO 3.9.8.2 “Residual Heat Removal and Coolant Circulation - Low Water Level.”

ACTIONS

A Note prohibits the application of LCO 3.0.4b to an inoperable ECCS high head subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS high head subsystem and the provisions of LCO 3.0.4b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A second Note allows the required ECCS RHR subsystem to be inoperable because of surveillance testing of RCS pressure isolation valve leakage (FCV-74-1 and FCV-74-2). This allows testing while RCS pressure is high enough to obtain valid leakage data and following valve closure for RHR decay heat removal path. The condition requiring alternate heat removal methods ensures that the RCS heatup rate can be controlled to prevent MODE 3 entry and thereby ensure that the reduced ECCS operational requirements are maintained. The condition requiring manual realignment capability, FCV-74-1 and FCV-74-2 can be opened from the main control room ensures that in the unlikely event of a DBA during the one hour of surveillance testing, the RHR subsystem can be placed in ECCS recirculation mode when required to mitigate the event.

Action a.

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The action time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

ACTIONS (continued)

Action b.

With no ECCS high head subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour action time to restore at least one ECCS high head subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

When Action b cannot be completed within the required action time, within one hour, a controlled shutdown should be initiated. Twenty four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE REQUIREMENTS

SR 4.5.3

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

1. The applicable references from Bases 3.5.2 apply.
2. NRC Safety Evaluation Report, NUREG-0011, Section 1.1, "Introduction," regarding Amendment 49 dated January 6, 1978.

3/4.5.4 BORON INJECTION SYSTEM

This specification was deleted.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

Additionally, the OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.5.6 SEAL INJECTION FLOW

BACKGROUND

The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of the NRC Policy Statement.

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

LCO (continued)

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection needle valves to provide a total seal injection flow in the acceptable region of Technical Specification Figure 3.5.6-1. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Technical Specification Figure 3.5.6-1 are consistent with the accident analysis.

The limits on seal injection flow must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTION

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this condition, action must be taken to restore the flow to below its limit. The operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The completion time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the completion times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

ACTIONS(continued)

When the actions cannot be completed within the required completion time, a controlled shutdown must be initiated. The completion time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown from MODE 3, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

Surveillance 4.5.6

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The differential pressure that is above the reference minimum value is established between the charging header (PT 62-92) and the RCS, and total seal injection flow is verified to be within the limits determined in accordance with the ECCS safety analysis (Ref. 3). The seal water injection flow limits are shown in Technical Specification Figure 3.5.6-1. The frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve surveillance frequencies. The frequency has proven to be acceptable through operating experience.

The requirements for charging flow vary widely according to plant status and configuration. When charging flow is adjusted, the positions of the air-operated valves, which control charging flow, are adjusted to balance the flows through the charging header and through the seal injection header to ensure that the seal injection flow to the RCPs is maintained between 8 and 13 gpm per pump. The reference minimum differential pressure across the seal injection needle valves ensures that regardless of the varied settings of the charging flow control valves that are required to support optimum charging flow, a reference test condition can be established to ensure that flows across the needle valves are within the safety analysis. The values in the safety analysis for this reference set of conditions are calculated based on conditions during power operation and they are correlated to the minimum ECCS flow to be maintained under the most limiting accident conditions.

As noted, the surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the surveillance is timely. Performance of this surveillance within the 4-hour allowance is required to maintain compliance with the provisions of Specification 4.0.3.

REFERENCES

1. FSAR, Chapter 6.3 "Emergency Core Cooling System" and Chapter 15.0 "Accident Analysis."
2. 10 CFR 50.46.
3. Westinghouse Electric Company Calculation CN-FSE-99-48

LCO (continued)

head (NPSH), and without exceeding the maximum design temperature of the equipment served by the ERCW. To meet this condition, the UHS temperature should not exceed 87°F, when the ERCW System is not in the alignment to support large heavy load lifts associated with the Unit 2 refueling outage 18 steam generator replacement project, and the level should not fall below the 674 feet mean sea level during normal unit operation. When the ERCW System is in the alignment to support large heavy load lifts associated with the Unit 2 refueling outage 18 steam generator replacement project, the UHS temperature should not exceed 74°F. The alignment to support these large heavy load lifts, which maintains the ERCW System OPERABLE in the event of large heavy load drop, is described in Appendix C, "Additional Conditions," of the Operating License.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

The maximum allowed UHS temperature value is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit and the configuration of the ERCW System. Measurement of this temperature is in accordance with NUREG/CR-3659 methodology which includes measurement uncertainties (Ref: 5).

With average water temperature of the UHS ≤ 87°F (when the ERCW System is not in the alignment to support large heavy load lifts) or ≤ 74°F (when the ERCW System is in the alignment to support large heavy load lifts), the associated design basis assumptions remain bounded for all accidents, transients, and shutdown. Long-term cooling capability is provided to the Emergency Core Cooling System (ECCS) and Emergency Diesel Generator loads.

If the water temperature of the UHS exceeds the limits of the LCO, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within the following 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 4.7.5.1

This SR verifies that the ERCW is available to cool the CCS to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident.

This SR also verifies that adequate long-term (30 day) cooling can be maintained. The specified level ensures that sufficient reservoir volume exists at the initiation of a LBLOCA concurrent with loss of downstream dam to meet the short-term recovery. NPSH of the ERCW pumps are not challenged with loss of downstream dam. The 24-hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR verifies that the average water temperature of the UHS is ≤ 87°F (when the ERCW Sysem is not in the alignment to support large heavy load lifts) and ≤ 74°F (when the ERCW System is in the alignment to support large heavy load lifts) and that the UHS water level is ≥ 674 feet mean sea level.

REFERENCES

1. UFSAR, Section 9.2.5, Ultimate Heat Sink
2. UFSAR, Section 6.2.1, Containment Functional Design
3. UFSAR, Section 9.2.2, Essential Raw Cooling Water (ERCW)
4. Regulatory Guide 1.27 R0, "Ultimate Heat Sink For Nuclear Power Plants," 1972
5. NUREG/CR-3659, "A Mathematical Model For Assessing The Uncertainties Of Instrumentation Measurements For Power And Flow Of PWR Reactors," February 1985.

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The electrically powered AC safety loads are separated into redundant load groups such that loss of any one load group will not prevent the minimum safety functions from being performed. Specification 3.8.1.1 requires two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System and four separate and independent diesel generator sets to be OPERABLE in MODES 1, 2, 3, and 4. These requirements ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated design basis accident.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident. Minimum required switchyard voltages are determined by evaluation of plant accident loading and the associated voltage drops between the transmission network and these loads. These minimum voltage values are provided to TVA's Transmission Operations for use in system studies to support operation of the transmission network in a manner that will maintain the necessary voltages. Transmission Operations is required to notify SQN Operations if it is determined that the transmission network may not be able to support accident loading or shutdown operations as required by 10 CFR 50, Appendix A, GDC-17. Any offsite power circuits supplied by that transmission network that are not able to support accident loading or shutdown operations are inoperable.

The unit station service transformers (USSTs) utilize auto load tap changers to provide the required voltage response for accident loading. The load tap changer associated with a USST is required to be functional and in "automatic" for the USST to supply power to a 6.9 kV Unit Board.

The inability to supply offsite power to a 6.9 kV Shutdown Board constitutes the failure of only one offsite circuit, as long as offsite power is available to the other load group's Shutdown Boards. Thus, if one or both 6.9 kV Shutdown Boards in a load group do not have an offsite circuit available, then only one offsite circuit would be inoperable. If one or more Shutdown Boards in each load group, or all four Shutdown Boards, do not have an offsite circuit available, then both offsite circuits would be inoperable. An "available" offsite circuit meets the requirements of GDC-17, and is either connected to the 6.9 kV Shutdown Boards or can be connected to the 6.9 kV Shutdown Boards within a few seconds.

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network (beginning at the switchyard) to one load group of Class 1E 6.9 kV Shutdown Boards (ending at the supply side of the normal or alternate supply circuit breaker). Each required offsite circuit is that combination of power sources described below that are normally connected to the Class 1E distribution system, or can be connected to the Class 1E distribution system through automatic transfer at the 6.9 kV Unit Boards.

The following offsite power configurations meet the requirements of LCO 3.8.1.1.a:
(Note that common station service transformer (CSST) B is a spare transformer with two sets of secondary windings that can be used to supply a total of two Start Buses for CSST A and/or CSST C, with each supplied Start Bus on a separate CSST B secondary winding.)

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

1. Two offsite circuits consisting of a AND b (no board transfers required; a loss of either circuit will not prevent the minimum safety functions from being performed):

   a. From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C), and CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

   b. From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

2. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.2)(b), or a.2) to b.1)(a) on a loss of USSTs 1A and 1B, OR relies on automatic transfer from alignment a.3) to b.2)(a), or a.4) to b.1)(b) on a loss of USSTs 2A and 2B):

   a. Normal power source alignments

      1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B);

      2) From the 500 kV switchyard through USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);

      3) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B); AND

      4) From the 161 kV switchyard through USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C).

   b. Alternate power source alignments

      1) From the 161 kV transmission network, through:

         (a) CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

         (b) CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); OR

      2) From the 161 kV transmission network, through:

         (a) CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), AND

         (b) CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

3. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 2 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):

   a. Normal power source alignments

      1) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C);

      2) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

      3) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

   b. Alternate power source alignments

      1) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

      2) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).

4. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 1 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):

   a. Normal power source alignments

      1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B), and USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);

      2) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

      3) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).

   b. Alternate power source alignments

      1) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

      2) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Other offsite power configurations are possible using different combinations of available USSTs and CSSTs, as long as the alignments are consistent with the analyzed configurations, and the alignments otherwise comply with the requirements of GDC 17.

For example, to support breaker testing, offsite power to the 6.9 kV Shutdown Boards can be realigned from normal feed to alternate feed. This would result in Shutdown Boards 1A-A and 2A-A being fed from Unit Boards 1A and 2A, respectively, and Shutdown Boards 1B-B and 2B-B being fed from Unit Boards 1D and 2D, respectively. The CSST being utilized as the alternate power source to one load group of Shutdown Boards would also be realigned (normally CSST A available to Shutdown Boards 1B-B and 2B-B or CSST C available to Shutdown Boards 1A-A and 2A-A, would be realigned to CSST A available to Shutdown Boards 1A-A and 2A-A or CSST C available to Shutdown Boards 1B-B and 2B-B).

LCO 3.8.1.1 is modified by Note @ that specifies CSST A and CSST C are required to be available via automatic transfer at the associated 6.9 KV Unit Boards, when USST 2A and USST 2B are being utilized as normal power sources to the offsite circuits. (Note that CSST B can be substituted for CSST A or CSST C.) This offsite power alignment is consistent with Configuration 3, as stated above. Note @ remains in effect until November 30, 2013, or until the USST modifications are implemented on Units 1 and 2, whichever occurs first. (The scheduled startup from the Unit 1 fall 2013 refueling outage is November 2013.) Following expiration of Note @, Configuration 3 can continue to be used.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The footnote for Action b of LCO 3.8.1.1 requires completion of a determination that the OPERABLE diesel generators are not inoperable due to common cause failure or performance of Surveillance 4.8.1.1.2.a.4 if Action b is entered. The intent is that all diesel generator inoperabilities must be investigated for common cause failures regardless of how long the diesel generator inoperability persists.

Action b of LCO 3.8.1.1 is further modified by a second note which precludes making more than one diesel generator inoperable on a pre-planned basis for maintenance, modifications, or surveillance testing. The intent of this footnote is to explicitly exclude the flexibility of removing a diesel generator set from service as a part of a pre-planned activity. While the removal of a diesel generator set (A or B train) is consistent with the initial condition assumptions of the accident analysis, this configuration is judged as imprudent. The term pre-planned is to be taken in the context of those activities which are routinely scheduled and is not relative to conditions which arise as a result of emergent or unforeseen events. As an example, this footnote is not intended to preclude the actions necessary to perform the common mode testing requirements required by Action b. As another example, this footnote is not intended to prevent the required surveillance testing of the diesel generators should one diesel generator maintenance be unexpectedly extended and a second diesel generator fall within its required testing frequency. Thus, application of the note is intended for pre-planned activities.

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

In addition, this footnote is intended to apply only to those actions taken directly on the diesel generator. For those actions taken relative to common support systems (e.g. ERCW), the support function must be evaluated for impact on the diesel generator.

The action to determine that the OPERABLE diesel generators are not inoperable due to common cause failure provides an allowance to avoid unnecessary testing of OPERABLE diesel generators. If it can be determined that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generators, Surveillance Requirement 4.8.1.1.2.a.4 does not have to be performed. If the cause of inoperability exists on other diesel generator(s), the other diesel generator(s) would be declared inoperable upon discovery and Action e of LCO 3.8.1.1 would be entered as applicable. Once the common failure is repaired, the common cause no longer exists, and the action to determine inoperability due to common cause failure is satisfied. If the cause of the initial inoperable diesel generator cannot be confirmed not to exist on the remaining diesel generators, performance of Surveillance 4.8.1.1.2.a.4 suffices to provide assurance to continued OPERABILITY of the other diesel generators.

According to Generic Letter 84-15, 24 hours is reasonable to confirm that the OPERABLE diesel generators are not affected by the same problem as the inoperable diesel generator.

Action f prohibits the application of LCO 3.0.4.b to an inoperable diesel generator. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable diesel generator and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

With the minimum required AC power sources not available, it is required to suspend CORE ALTERATIONS and operations involving positive reactivity additions that could result in loss of required SDM (Mode 5) or boron concentration (Mode 6). Suspending positive reactivity additions that could result in failure to meet minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than or equal to that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

The requirements of Specification 3.8.2.1 provide those actions to be taken for the inoperability of A.C. Distribution Systems. Action a of this specification provides an 8-hour action for the inoperability of one or more A.C. boards. Action b of this specification provides a relaxation of the 8-hour action to 24-hours provided the Vital Instrument Power Board is inoperable solely as a result of one inoperable inverter and the board has been energized within 8 hours. In this condition the requirements of Action a do not have to be applied. Action b is not intended to provide actions for inoperable inverters, which is

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

addressed by the operability requirements for the boards, and is included only for relief from the 8-hour action of Action a when only one inverter is affected. More than one inverter inoperable will result in the inoperability of the associated 120 Volt A.C. Vital Instrument Power Board(s) in accordance with Action a. With more than one inverter inoperable entry into the actions of TS 3.0.3 is not applicable because Action a includes provisions for multiple inoperable inverters as attendant equipment to the boards.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137 "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979. The Surveillance Requirements for the diesel generator load-run test and the 24-hour endurance and margin test are in accordance with Regulatory Guide 1.9, Revision 3, July 1993, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants." During the diesel generator endurance and margin surveillance test, momentary transients outside the kw and kvar load ranges do not invalidate the test results. Similarly, during the diesel generator load-run test, momentary transients outside the kw load range do not invalidate the test results.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. 6800 volts is the minimum steady state output voltage and the 10 second transient value. 6800 volts is 98.6% of nominal bus voltage of 6900 volts and is based on the minimum voltage required for the diesel generator supply breaker to close on the 6.9 kV Shutdown Board. The specified maximum steady state output voltage of 7260 volts is based on the degraded over voltage relay setpoint and is equivalent to 110% of the nameplate rating of the 6600 volt motors. The specified minimum and maximum frequencies of the diesel generator are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in regulatory Guide 1.9.

Where the SRs discuss maximum transient voltages during load rejection testing, the following is applicable. The maximum transient voltage of 8880 volts represents a conservative limit to ensure the resulting voltage will not exceed a level that will cause component damage. It is based on the manufacturer's recommended high potential test voltage of 60% of the original factory high potential test voltage (14.8 kV). The diesel generator manufacturer has determined that the engine and/or generator controls would not experience detrimental effects for transient voltages < 9000 volts. The maximum transient voltage of 8276 volts is retained from the original technical specifications to ensure that the voltage transient following rejection of the single largest load is within the limits originally considered acceptable. It was based on 114% of 7260 volts, which is the Range B service voltage per ANSI-C84.1.

The Surveillance Requirement (SR) to transfer the power supply to each 6.9 kV Unit Board from the normal supply to the alternate supply demonstrates the OPERABILITY of the alternate supply to power the shutdown loads. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This SR is modified by two Notes. The reason for Note # is that, during operation with the reactor critical, performance of this SR for the Unit 1 Unit Boards could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Note ## specifies that transfer

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

capability is only required to be met for 6.9 kV Unit Boards that require normal and alternate power supplies. When both load groups are being supplied power by the USSTs, only the 6.9 kV Unit Boards associated with one load group are required to have normal and alternate power supplies. Therefore, only one CSST is required to be OPERABLE and available as an alternate power supply. Additionally, manual transfers between the normal supply and the alternate supply are not relied upon to meet the accident analysis. Manual transfer capability is verified to ensure the availability of a backup to the automatic transfer feature.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacture's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If the results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The test, limits, and applicable ASTM Standards are as follows:

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

a. Sample the new fuel in accordance with D4057-1988 (ref.);

b. Verify in accordance with the test specified in ASTM D975-1990 (Ref.) that the sample has an absolute specific gravity at 60/60 degrees F of $\geq$ 0.83 and $\leq$ 0.89 or an API gravity at 60 degrees F of $\geq$ 27 degrees and $\leq$ 39 degrees, a kinematic viscosity at 40 degrees C of $\geq$ 1.9 centistokes and $\leq$ 4.1 centistokes, and a flash point of $\geq$ 125 degrees F; and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-1986 (Ref.).

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-1990 (Ref.) are met, except that the analysis for sulfur may be performed in accordance with ASTM D1552-1990 (Ref.) or ASTM D2622-1987 (Ref.). The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on D/G operation. This surveillance ensures availability of high quality fuel oil for the D/Gs.

Fuel oil degradation during long-term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-94, Method A (Ref.). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each of the four interconnected tanks which comprise a 7-day tank must be considered and tested separately.

The frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between frequency intervals.

References:

ASTM Standards

D4057-1988, "Practice for manual sampling of petroleum and petroleum Products."

D975-1990, "Standard Specifications for Diesel Fuel oils."

D4176-1986, "Free Water and Particulate Contamination in Distillate Fuels."

D1552-1990, "Standard Test Method for Sulfur in Petroleum Products (High Temperature Method)."

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

D2622-1987, "Standard Test Method for Sulfur in Petroleum Products (X-Ray Spectrographic Method)."

D2276-1994, "Standard Test Method for Particulate Containment in Aviation Turbine Fuels."

D1298-1985, "Standard Test Method for Density, Specific Gravity, or API Gravity of Crude Petroleum and Liquid Petroleum Products by Hydrometer Method."

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

This specification is deleted.

ATTACHMENT 3

SEQUOYAH NUCLEAR PLANT, UNIT 2
TECHNICAL SPECIFICATION BASES
CHANGED PAGES

TS Bases Affected Pages

EPL Page 2
EPL Page 3
EPL Page 16
EPL Page 17
EPL Page 19
EPL Page 20
EPL Page 21
EPL Page 22
EPL Page 31
EPL Page 32
Index Page III
Index Page XIV
B 2-1
B 2-2
B 2-3
B 2-4
B 2-5
B 2-6
B 2-7
B 2-8
B 2-9
B 2-10
B 2-11
B 3/4 2-4
B 3/4 3-3a
B 3/4 4-3a
B 3/4 4-3b
B 3/4 4-3c
B 3/4 4-3d
B 3/4 4-3e
B 3/4 4-3f
B 3/4 4-3g

Removed B 3/4 4-3h
through B 3/4 4-3k
B 3/4 4-4f
B 3/4 5-12
B 3/4 5-13
B 3/4 5-14
B 3/4 5-15
B 3/4 5-16
B 3/4 5-17
B 3/4 5-18
B 3/4 5-19
B 3/4 5-20
B 3/4 8-1
B 3/4 8-2
B 3/4 8-3
B 3/4 8-4
B 3/4 8-5
B 3/4 8-6
B 3/4 8-7
B 3/4 8-8
B 3/4 8-9

EFFECTIVE PAGE LISTING

Page	Revision
Index Page VII	01/28/10
Index Page VIII	12/28/05
Index Page IX	12/28/05
Index Page X	12/28/05
Index Page XI	12/18/00
Index Page XII	03/09/05
Index Page XIII	01/28/10
Index Page XIV	12/21/12
Index Page XV	12/18/00
Index Page XVI	08/02/06
Index Page XVII	05/24/02
1-1	05/18/88
1-2	04/13/09
1-3	02/29/00
1-4	05/22/07
1-5	05/22/07
1-6	08/02/06
1-7	09/15/04
1-8	09/15/04
1-9	05/18/88
1-10	05/18/88
2-1	09/26/12
2-2	09/26/12
2-3 (DELETED)	09/03/85

EFFECTIVE PAGE LISTING

Page	Revision
2-4	09/13/06
2-5	09/26/12
2-6	09/13/06
2-7	09/20/07
2-8	09/13/06
2-9	09/13/06
2-10	09/13/06
2-11	09/13/06
2-12	09/13/06
B(Note)	Original
B2-1	10/10/12
B2-2	10/10/12
B2-3	10/10/12
B2-4	10/10/12
B2-5	10/10/12
B2-6	10/10/12
B2-7	10/10/12
B2-8	10/10/12
B2-9	10/10/12
B2-10	10/10/12
B2-11	10/10/12
3/4 0-1	10/04/06
3/4 0-2	10/04/06
3/4 0-3	10/04/06
3/4 0-4	10/04/06
3/4 1-1	11/26/93
3/4 1-2	Original
3/4 1-3	11/26/93

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 0-4	02/05/03
B3/4 0-5	06/16/06
B3/4 0-6	06/16/06
B3/4 1-1	10/26/93
B3/4 1-2	12/18/00
B3/4 1-3	12/18/00
B3/4 1-3a	03/07/07
B3/4 1-4	04/21/97
B3/4 1-4a	11/21/95
B3/4 2-1	04/21/97
B3/4 2-2	04/21/97
B3/4 2-3 (Figure B3/4 2-1 DELETED)	09/29/83
B3/4 2-4	10/10/12
B3/4 3-1	09/13/06
B3/4 3-2	09/13/06
B3/4 3-2a	08/29/08
B3/4 3-3	12/28/05
B3/4 3-3a	03/05/13
B3/4 3-4	08/12/97
B3/4 3-5 through B3/4 3-9	09/14/06
B3/4 4-1	03/30/92
B3/4 4-2	06/16/06
B3/4 4-2a	05/25/00
B3/4 4-3	05/22/07

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 4-3a	10/05/12
B3/4 4-3b	10/05/12
B3/4 4-3c	10/05/12
B3/4 4-3d	10/05/12
B3/4 4-3e	10/05/12
B3/4 4-3f	10/05/12
B3/4 4-3g	10/05/12
B3/4 4-3h (Deleted)	10/05/12
B3/4 4-3i (Deleted)	10/05/12
B3/4 4-3j (Deleted)	10/05/12
B3/4 4-3k (Deleted)	10/05/12
B3/4 4-4	12/04/08
B3/4 4-4a	12/04/08
B3/4 4-4b	04/11/05
B3/4 4-4c	12/04/08
B3/4 4-4d	12/04/08
B3/4 4-4e	05/22/07
B3/4 4-4f	10/05/12
B3/4 4-4g	05/22/07
B3/4 4-4h	05/22/07
B3/4 4-4i	05/22/07
B3/4 4-4j	05/22/07
B3/4 4-4k	08/04/00
B3/4 4-4l	08/04/00

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 5-1	03/25/10
B3/4 5-2	03/25/10
B3/4 5-3	03/25/10
B3/4 5-4	03/25/10
B3/4 5-5	03/25/10
B3/4 5-6	03/25/10
B3/4 5-7	03/25/10
B3/4 5-8 through B3/4 5-11	03/25/10
B3/4 5-12 through B3/4 5-20	03/24/12
B3/4 6-1 through B3/4 6-2	04/13/09
B3/4 6-3	05/27/10
B3/4 6-4 through B3/4 6-6	04/13/09
B3/4 6-7 through B3/4 6-12	04/13/09
B3/4 6-13 through B3/4 6-18	04/13/09
B3/4 6-19 through B3/4 6-20	04/13/09
B3/4 6-21	04/13/09
B3/4 7-1	04/30/02
B3/4 7-2	08/14/01
B3/4 7-2a	11/17/95
B3/4 7-2b	04/11/05
B3/4 7-3	06/12/09

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 7-3a	06/08/98
B3/4 7-4	09/28/07
B3/4 7-4a	09/28/07
B3/4 7-4b	09/28/07
B3/4 7-4c thru B3/4 7-4m	10/28/08
B3/4 7-5	08/18/05
B3/4 7-6 (DELETED)	08/28/98
B3/4 7-6a	12/28/05
B3/4 7-7 through B3/4 7-8 (DELETED)	08/12/97
B3/4 7-9	12/19/00
B3/4 7-10	12/19/00
B3/4 7-11	12/19/00
B3/4 7-12	12/19/00
B3/4 7-13	12/19/00
B3/4 7-14	12/19/00
B3/4 7-15	12/19/00
B3/4 7-16	01/31/05
B3/4 7-17	02/27/02
B3/4 7-18	02/27/02
B3/4 8-1	12/21/12
B3/4 8-2	12/21/12
B3/4 8-3	12/21/12
B3/4 8-4	12/21/12
B3/4 8-5	12/21/12
B3/4 8-6	12/21/12
B3/4 8-7	12/21/12
B3/4 8-8	12/21/12
B3/4 8-9	12/21/12

EFFECTIVE PAGE LISTING

Page	Revision
B3/4 9-1	09/20/04
B3/4 9-2	12/28/05
B3/4 9-3	04/19/04
B3/4 10-1	09/20/04
B3/4 11-1	12/09/93
B3/4 11-2	11/16/90
B3/4 12-1	11/16/90
5-1	08/02/06
5-2	08/02/06
5-3	08/02/06
5-4	08/02/06
5-5	12/19/00
5-5a	12/19/00
5-5b	08/02/06
5-5c	12/19/00
5-5d	12/19/00
5-5e	12/19/00
5-5f	12/19/00
5-5g	12/19/00
5-5h	12/19/00
5-5i	12/19/00
5-5j	12/19/00
5-6	08/02/06

EFFECTIVE PAGE LISTING

Page	Revision
6-1	02/16/01
6-2	02/02/10
6-3 through 6-4 (DELETED)	02/16/01
6-5	02/11/03
6-6	05/24/02
6-7	02/11/03
6-8	02/11/03
6-9	04/13/09
6-10	04/13/09
6-10a	07/10/12
6-10b	07/10/12
6-10c (Deleted)	07/10/12
6-10d (Deleted)	07/10/12
6-11	04/13/09
6-12	08/02/93
6-13	09/26/12
6-14	09/26/12
6-14a	09/26/12
6-15	07/10/12
6-16	02/11/03
6-16a	02/11/03
6-16b	02/11/03
6-17	07/01/98
6-18	02/11/03
6-19	10/28/08
6-20	10/28/08

AMENDMENT LISTING

Amendments	Date and Revision
Amendment 288 Issued by NRC	03/09/05 (R288)
Amendment 289 Issued by NRC	04/05/05 (R289)
Amendment 290 Issued by NRC	04/11/05 (R290)
Amendment 291 Issued by NRC	05/03/05 (R291)
Amendment 292 Issued by NRC	05/24/05 (R292)
Amendment 293 Issued by NRC	08/18/05 (R293)
Amendment 294 Issued by NRC	09/02/05 (R294)
Bases Revision	09/11/03 (BR-28)
Amendment 295 Issued by NRC	12/28/05 (R295)
Amendment 296 Issued by NRC	04/06/06 (R296)
Amendment 297 Issued by NRC	06/16/06 (R297)
Amendment 298 Issued by NRC	08/02/06 (R298)
Amendment 299 Issued by NRC	09/13/06 (R299)
Amendment 300 Issued by NRC	09/14/06 (R300)
Amendment 301 Issued by NRC	10/04/06 (R301)
Amendment 302 Issued by NRC	11/07/06 (R302)
Amendment 303 Issued by NRC	11/16/06 (R303)
Amendment 304 Issued by NRC	12/11/06 (R304)
License Condition Issued by NRC	02/08/07
Bases Revision	03/07/07 (BR-29)
Amendment 305 Issued by NRC	05/22/07 (R305)
EPL Revised	05/22/07
License Condition Issued by NRC	08/09/07 (B.5.b)
Amendment 306 Issued by NRC	09/20/07 (R306)
Amendment 307 Issued by NRC	09/28/07 (R307)
Amendment 308 Issued by NRC	10/11/07 (R308)
Bases Revision	12/12/07 (BR-30)
Amendment 309 Issued by NRC	03/24/08 (R309)
Amendment 310 Issued by NRC	04/02/08 (R310)
Amendment 311 Issued by NRC	04/04/08 (R311)
Amendment 312 Issued by NRC	08/29/08 (R312)
Bases Revision	08/29/08 (BR-31)
Bases Revision	08/28/08 (BR-32)
Amendment 313 Issued by NRC	10/28/08
Amendment 314 Issued by NRC	12/04/08
Amendment 315 Issued by NRC	04/13/09
Amendment 316 Issued by NRC	06/12/09
Bases Revision	06/12/09 (BR-33)
Amendment 317 Issued by NRC	08/14/09
Amendment 318 Issued by NRC	10/19/09
Bases Revision	10/19/09 (BR-34)
Amendment 319 Issued by NRC	01/28/10
Amendment 320 Issued by NRC	02/02/10
Bases Revision	03/25/10 (BR-35)
Bases Revision	05/27/10 (BR-36)
Amendment 321 Issued by NRC	12/21/10
Bases Revision	03/24/12 (BR-38)
Amendment 323 Issued by NRC	07/10/12
Bases Revision	10/05/12 (BR-40)
Amendment 324 Issued by NRC	09/26/12
Bases Revision	10/10/12 (BR-39)

SEQUOYAH NUCLEAR PLANT UNIT 2
TECHNICAL SPECIFICATIONS

AMENDMENT LISTING

Amendments	Date and Revision
Amendment 325 Issued by NRC	10/31/12
Bases Revision	12/21/12 (BR-41)
Bases Revision	03/05/13 (BR-42)

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION PAGE

2.1 SAFETY LIMITS

Reactor Core 2-1

Reactor Coolant System Pressure 2-1

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Trip System Instrumentation Setpoints 2-4

BASES

SECTION PAGE

2.1 SAFETY LIMITS

Reactor Core .... B 2-1

Reactor Coolant System Pressure .... B 2-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Trip System Instrumentation Setpoints .... B 2-3

SECTION		PAGE
3/4.7.4	ESSENTIAL RAW COOLING WATER SYSTEM	B 3/4 7-3a
3/4.7.5	ULTIMATE HEAT SINK	B 3/4 7-4
3/4.7.6	FLOOD PROTECTION	B 3/4 7-4
3/4.7.7	CONTROL ROOM EMERGENCY VENTILATION SYSTEM	B 3/4 7-4
3/4.7.8	AUXILIARY BUILDING GAS TREATMENT SYSTEM	B 3/4 7-5
3/4.7.9	SNUBBERS	B 3/4 7-5
3/4.7.10	SEALED SOURCE CONTAMINATION (DELETED)	B 3/4 7-6a
3/4.7.11	FIRE SUPPRESSION SYSTEMS (DELETED)	B 3/4 7-7
3/4.7.12	FIRE BARRIER PENETRATIONS (DELETED)	B 3/4 7-8
3/4.7.13	SPENT FUEL POOL MINIMUM BORON CONCENTRATION	B 3/4 7-9
3/4.7.14	CASK PIT POOL MINIMUM BORON CONCENTRATION	B 3/4 7-13
3/4.7.15	CONTROL ROOM AIR-CONDITIONING SYSTEM (CRACS)	B 3/4 7-16
3/4.8 ELECTRICAL POWER SYSTEMS		
3/4.8.1	and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS	B 3/4 8-1
3/4.8.3	ELECTRICAL EQUIPMENT PROTECTIVE DEVICES (DELETED)	B 3/4 8-9
3/4.9 REFUELING OPERATIONS		
3/4.9.1	BORON CONCENTRATION	B 3/4 9-1
3/4.9.2	INSTRUMENTATION	B 3/4 9-1
3/4.9.3	DECAY TIME	B 3/4 9-1
3/4.9.4	CONTAINMENT BUILDING PENETRATIONS	B 3/4 9-1
3/4.9.5	COMMUNICATIONS (Deleted)	B 3/4 9-2
3/4.9.6	MANIPULATOR CRANE (Deleted)	B 3/4 9-2
3/4.9.7	CRANE TRAVEL - SPENT FUEL PIT AREA (DELETED)	B 3/4 9-2
3/4.9.8	RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION	B 3/4 9-2
3/4.9.9	CONTAINMENT VENTILATION SYSTEM	B 3/4 9-3

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel cladding (due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt), either of which could result in cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs.

Operation above the upper boundary of the nucleate boiling regime could result in excessive temperatures because of the onset of departure from nucleate boiling (DNB) and the corresponding significant reduction in heat transfer coefficient from the outer surface of the cladding to the reactor coolant water. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant. DNB is not a directly measurable parameter during operation and; therefore, THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is that there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. These lines are bounding for all fuel types. The curves in Figure 2.1-1 are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (RPS trip limits). The plant trip setpoints are verified to be less than the limits defined by the safety limit lines in Figure 2.1-1 converted from power to delta-temperature and adjusted for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the acceptance criteria in the safety analysis.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance, or Delta-I (ΔI), is within the limits of the f_1 (ΔI) function of the Overtemperature Delta-Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the f_1 (ΔI) trip reset function, the Overtemperature Delta-Temperature trip setpoint is reduced by the values in the CORE OPERATING LIMITS REPORT to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for centerline fuel melt are higher than those calculated for the range of all control rods from the fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance, or Delta-I (ΔI), is within the limits of the f_2 (ΔI) function of the Overpower Delta-Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the f_2 (ΔI) trip resent function, the Overpower Delta-Temperature trip setpoint is reduced by the values specified in the CORE OPERATING LIMITS REPORT to provide protection required by the core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Nominal Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Nominal Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Nominal Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Nominal Trip Setpoint and the Allowable Value is equal to or less than the rack allowance assumed for each trip in the safety analyses.

Technical specifications are required by 10 CFR 50.36 to contain Limiting Safety System Settings (LSSS) defined by the regulation as ". . . settings for automatic protective devices . . . so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." The analytic limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the analytic limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protective devices must be chosen to be more conservative than the analytic limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the analytic limit and thus ensuring that the SL would not be exceeded. As such, the Nominal Trip Setpoint accounts for uncertainties in setting the device (e.g., calibration), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the Nominal Trip Setpoint plays an important role in ensuring that SLs are not exceeded. As such, the Nominal Trip Setpoint meets the definition of an LSSS in accordance with Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," and could be used to meet the requirements that they be contained in the technical specifications.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Technical specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in the technical specifications as ". . . being capable of performing its safety function(s)." For automatic protective devices, the required safety function is to ensure that a SL is not exceeded and therefore the LSSS as defined by 10 CFR 50.36 is the same as the OPERABILITY limit for these devices. However, use of the Nominal Trip Setpoint to define OPERABILITY in technical specifications and its corresponding designation as the LSSS required by 10 CFR 50.36 would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as found" value of a protective device setting during a surveillance. This would result in technical specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the Nominal Trip Setpoint due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the Nominal Trip Setpoint and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to the Nominal Trip Setpoint to account for further drift during the next surveillance interval.

Use of the Nominal Trip Setpoint to define "as found" OPERABILITY and its designation as the LSSS under the expected circumstances described above would result in actions required by both the rule and technical specifications that are clearly not warranted. However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This value needs to be specified in the technical specifications in order to define OPERABILITY of the devices and is designated as the Allowable Value, which as stated above, is the same as the LSSS.

The Allowable Value specified in Table 2.2-1 serves as the LSSS such that a channel is OPERABLE if the trip setpoint is found not to exceed the Allowable Value during the CHANNEL FUNCTIONAL TEST (CFT). As such, the Allowable Value differs from the Nominal Trip Setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a Safety Limit is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is "OPERABLE" under these circumstances, the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual setting of the device is found to have exceeded the Allowable Value, the device would be considered inoperable from a technical specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

A channel is OPERABLE with a trip setpoint value outside its calibration tolerance band provided the trip setpoint "as-found" value does not exceed its associated Allowable Value and provided the trip setpoint "as-left" value is adjusted to a value within the "as-left" calibration tolerance band of the Nominal Trip Setpoint. A trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions. The conservative direction is established by the direction of the inequality applied to the Allowable Value.

A detailed description of the methodology used to calculate the Allowable Value and trip setpoints, including their explicit uncertainties, is provided in the Westinghouse Electric Company setpoint methodology study which incorporates all of the known uncertainties applicable to each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint and

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

corresponding Allowable Value. The trip setpoint entered into the channel is more conservative than that specified by the Allowable Value (LSSS) to account for measurement errors detectable by the CFT. The Allowable Value serves as the Technical Specification OPERABILITY limit for the purpose of the CFT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the channel is considered OPERABLE.

The trip setpoint is the value at which the channels are set and is the expected value to be achieved during calibration. The trip setpoint value ensures the LSSS and safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on the stated channel uncertainties. Any channel is considered to be properly adjusted when the "as-left" setpoint value is within the band for CHANNEL CALIBRATION uncertainty allowance (i.e. ± rack calibration + comparator setting uncertainties). The trip setpoint value is therefore, considered a "nominal" value (i.e., expressed as a value without inequalities) for the purposes of the CFT and CHANNEL CALIBRATION.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides a manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the safety analysis DNBR limit for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single dropped rods with a reactivity insertion of greater than 500 pcm or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate

Intermediate and Source Range, Nuclear Flux (Continued)

Range Channels will initiate a reactor trip at approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transit, thermowell, and RTD response time delays from the core to the temperature detectors (about 8 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to the RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in
Table 2.2-1.

The $f_1(\Delta I)$ trip reset term in the Overtemperature Delta T trip function precludes power distributions that cause the DNB limit to be exceeded during a limiting Condition II event. The negative and positive ΔI limits at which the $f_1(\Delta I)$ term begins to reduce the trip setpoint and the dependence of $f_1(\Delta I)$ on THERMAL POWER are determined on a cycle-specific basis using approved methodology and are specified in the COLR per Specification 6.9.1.14.

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature Delta T setpoint.

Delta-T_O, used in the Overtemperature and Overpower ΔT trips, represents the 100 percent RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flows greater than thermal design flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Accurate determination of the loop specific ΔT value should be made quarterly and under steady state conditions (i.e., power distributions not affected by xenon or other transient conditions.).

Overpower ΔT

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in axial power distribution, density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to the RTD output indication. The setpoint is automatically reduced according to the notations in Table 2.2-1 to account for adverse axial flux differences.

The $f_2(\Delta I)$ trip reset term in the Overpower Delta T trip function precludes power distributions that cause the fuel melt limit to be exceeded during a limiting Condition II event. The negative and positive ΔI limits at which the $f_2(\Delta I)$ term begins to reduce the trip setpoint and the dependence of $f_2(\Delta I)$ on THERMAL POWER are determined on a cycle-specific basis using approved methodology and are specified in the COLR per Specification 6.9.1.14.

The Overpower Delta T trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Delta-T_O, as used in the Overtemperature and Overpower ΔT trips, represents the 100 percent RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flows greater than thermal design flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Accurate determination of the loop specific ΔT value should be made quarterly and under steady state conditions (i.e., power distributions not affected by xenon or other transient conditions.).

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90 percent of nominal full loop flow. Above the P-8 interlock, automatic reactor trip will occur if the flow in any single loop drops below 90 percent of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature Delta T trip setpoint is adjusted to the value specified for all loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater or a feedwater system pipe break, outside of containment. This function also provides input to the steam generator level control system. IEEE 279 requirements are satisfied by 2/3 logic for protection function actuation, thus allowing for a single failure of a channel and still performing the protection function. Control/protection interaction is addressed by the use of the Median Signal Selector which prevents a single failure of a channel providing input to the control system requiring protection function action. That is, a single failure of a channel providing input to the control system does not result in the control system initiating a condition requiring protection function action. The Median Signal Selector performs this by not selecting the channels indicating the highest or lowest steam generator levels as input to the control system.

With the transmitters located inside containment and thus possibly experiencing adverse environmental conditions (due to a feedline break), the Environmental Allowance Modifier (EAM) was devised. The EAM function (Containment Pressure (EAM) with a setpoint of $\leq$ 0.5 psig) senses the presence of adverse containment conditions (elevated pressure) and enables the Steam Generator Water Level - Low-Low trip setpoint (Adverse) which reflects the increased transmitter uncertainties due to this environment. The EAM allows the use of a lower Steam Generator Water Level - Low-Low (EAM) trip setpoint when these conditions are not present, thus allowing more margin to trip for normal operating conditions.

The Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during early escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of RCS loop ΔT. Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level trip setpoint and the primary side power level. The magnitude of the delays decreases with increasing

Steam Generator Water Level (Cont'd)

primary side power level, up to 50 percent RTP. Above 50 percent RTP there are no time delays for the Low-Low level trips.

In the event of failure of a Steam Generator Water Level channel, it is placed in the trip condition as input to the Solid State Protection System and does not affect either the EAM or TTD setpoint calculations for the remaining operable channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation (T_S) to match the multiple steam generator time delay calculation (T_M) for the affected protection set, through the MMI. Failure of the Containment Pressure (EAM) channel to a protection set also does not affect the EAM setpoint calculations. This results in the requirement that the operator adjust the affected Steam Generator Water Level - Low-Low (EAM) trip setpoints to the same value as the Steam Generator Water Level - Low-Low (Adverse). Failure of the RCS loop ΔT channel input (failure of more than one T_H RTD or failure of a T_C RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50 percent RTP to 0 percent RTP, through the MMI.

The High Containment Pressure ESF trip that generates a safety injection signal and subsequent reactor trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from a feedwater system pipe break inside of containment. IEEE 279 requirements are satisfied by 2/3 logic for protection function actuation, thus allowing for a single failure of a channel and still performing the protection function.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.6 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions on increasing power:

P-6 Enables the manual block of the source range reactor trip (i.e., prevents premature block of source range trip).

P-7
P-13 Defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump undervoltage and underfrequency, pressurizer low pressure, and pressurizer high level.

P-8 Defeats the automatic block of reactor trip on low RCS coolant flow in a single loop.

P-9 Defeats the automatic block of reactor trip on turbine trip.

P-10 Enables the manual block of reactor trip on power range (low setpoint), intermediate range, as a backup block for source range, and intermediate range rod stops (i.e., prevents premature block of the noted functions).

On decreasing power, the opposite function is performed at reset setpoints.

P-4 Reactor-tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows manual block of the automatic reactuation of safety injection.

Reactor not tripped - defeats manual block preventing automatic reactuation of safety injection.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that no anomaly exists such that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The QUADRANT POWER TILT RATIO limit at which corrective action is required provides DNB and linear heat generation protection with x-y plane power tilts. The QUADRANT POWER TILT RATIO limit is reflected by a corresponding peaking augmentation factor which is included in the generation of the AFD limits.

The 2-hour time allowance for operation with the tilt condition greater than 1.02 but less than 1.09, is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the allowable THERMAL POWER by 3 percent for each percent of tilt in excess of 1.02.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

- Determine whether systems important to safety are performing their intended functions.
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release and to determine if a gross breach of a barrier has occurred.

For Sequoyah, the redundant channel capability for Auxiliary Feedwater (AFW) flow consists of a single AFW flow channel for each Steam Generator with the second channel consisting of three AFW valve position indicators (two level control valves for the motor driven AFW flowpath and one level control valve for the turbine drive AFW flowpath) for each steam generator.

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this specification. The analysis of an SGTR event assumes a bounding primary to secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2 "Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves. The main condenser isolates based on an assumed concurrent loss of off-site power.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere depends on the accident and whether there are faulted SGs associated with the accident. For a steamline break (SLB), the maximum primary to secondary leakage under accident conditions is limited to 3.7 gpm from the faulted SG and 0.1 gpm from each of the non-faulted SGs. For other accidents that assume a faulted SG (e.g., feedwater line break), the maximum primary to secondary leakage under accident conditions is limited to 1.0 gpm from the faulted SG and 0.1 gpm from each of the non-faulted SGs. For accidents in which there are no faulted SGs, the primary to secondary leakage is limited to 0.1 gpm from each SG. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), and 10 CFR 100 (Ref. 3).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

REACTOR COOLANT SYSTEM

BASES

LCO (continued)

In the context of this specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.k "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

REACTOR COOLANT SYSTEM

BASES

LCO (continued)

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all American Society of Mechanical Engineers (ASME) Code, Section III, Service Level A (normal operating conditions), and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analyses assumptions are discussed in the Applicable Safety Analyses section. The accident induced leakage rate includes any primary to secondary leakage existing prior to the accident in addition to primary to secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, "Operational Leakage," and limits primary to secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a loss-of-coolant accident (LOCA) or a SLB. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODES 1, 2, 3, or 4.

Reactor coolant system (RCS) conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

ACTIONS

The ACTIONs are modified by a clarifying footnote that Action (a) may be entered independently for each SG tube. This is acceptable because the actions provide appropriate compensatory measures for each affected SG tube. Complying with the actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent action entry, and application of associated actions.

Actions (a) and (b)

Action (a) applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 4.4.5.1. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained until the next refueling outage or SG inspection, Action (a) requires unit shutdown and Action (b) requires the affected tube(s) be plugged.

An allowed time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Action (a) allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to startup following the next refueling outage or SG inspection. This allowed time is acceptable since operation until the next inspection is supported by the operational assessment.

ACTIONS (continued)

If SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours and the affected tube(s) plugged prior to restart (Mode 4).

The action times are reasonable, based on operating experience, to reach the desired plant condition from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 4.4.5.0

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program defines the frequency of SR 4.4.5.0. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.4.5.1

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.k are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of this surveillance ensures that the surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential (i.e., prior to HOT SHUTDOWN following a SG tube inspection).

REACTOR COOLANT SYSTEM

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

REACTOR COOLANT SYSTEM

BASES

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a steam generator tube rupture or a steam line break (SLB) accident. To a lesser extent, other accidents or transients also involve secondary steam release to the atmosphere. The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for steam generator tube rupture (SGTR) assumes the contaminated secondary fluid is released via safety valves for up to 30 minutes. Operator action is taken to isolate the affected steam generator within this time period. The 0.4 gpm operational primary to secondary leakage safety analysis assumption is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes a 3.7 gpm primary to secondary leakage through the affected generator and 0.3 gpm through the non-affected generators as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). The expected leak rate following a steam line rupture is limited to below 3.7 gpm at atmospheric conditions and 70°F in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines.

The RCS operational leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. UNIDENTIFIED LEAKAGE

One gpm of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket

B 3/4.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3/4.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications. In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head). For the RHR subsystem during the injection phase, water is taken from the refueling storage tank (RWST) and injected in the Reactor Coolant System (RCS) through at least two cold legs.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation.

One train of ECCS during the injection phase provides sufficient flow for core cooling, by the centrifugal charging subsystem supplying each of the four cold legs and the RHR subsystem supplying at least two cold legs, to meet the analysis requirements for a credible MODE 4 Loss of Coolant Accident (LOCA.)

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

LCO (continued)

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to the cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.

Either RHR cold leg injection valve FCV-63-93 or FCV-63-94 may be closed when in MODE 4, for testing of the primary/secondary check valves in the injection lines. Closing one of the two cold leg injection flow paths does not make ECCS RHR subsystem inoperable.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. The manual actions necessary to realign the RHR subsystem may include actions to cool the RHR system piping due to the potential for steam voiding in piping or for inadequate NPSH available at the RHR pumps. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.1.4, "Reactor Coolant System Cold Shutdown." MODE 6 core cooling requirements are addressed by LCO 3.9.8.1 “Residual Heat Removal and Coolant Circulation - All Water Levels,” and LCO 3.9.8.2 “Residual Heat Removal and Coolant Circulation - Low Water Level.”

ACTIONS

A Note prohibits the application of LCO 3.0.4b to an inoperable ECCS high head subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS high head subsystem and the provisions of LCO 3.0.4b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A second Note allows the required ECCS RHR subsystem to be inoperable because of surveillance testing of RCS pressure isolation valve leakage (FCV-74-1 and FCV-74-2). This allows testing while RCS pressure is high enough to obtain valid leakage data and following valve closure for RHR decay heat removal path. The condition requiring alternate heat removal methods ensures that the RCS heatup rate can be controlled to prevent MODE 3 entry and thereby ensure that the reduced ECCS operational requirements are maintained. The condition requiring manual realignment capability, FCV-74-1 and FCV-74-2 can be opened from the main control room ensures that in the unlikely event of a design basis accident during the one hour of surveillance testing, the RHR subsystem can be placed in ECCS recirculation mode when required to mitigate the event.

Action a.

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The action time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

ACTIONS (continued)

Action b.

With no ECCS high head subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour action time to restore at least one ECCS high head subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

When Action b cannot be completed within the required action time, within one hour, a controlled shutdown should be initiated. Twenty four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE REQUIREMENTS

SR 4.5.3

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

1. The applicable references from Bases 3.5.2 apply.
2. NRC Safety Evaluation Report, NUREG-0011, Section 1.1, "Introduction," regarding Amendment 49 dated January 6, 1978.

3/4.5.4 BORON INJECTION SYSTEM

This Specification was deleted.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST), as part of the ECCS, ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation-cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses. Additionally, the OPERABILITY of the RWST, as part of the ECCS, ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.5.6 SEAL INJECTION FLOW

BACKGROUND

The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of the NRC Policy Statement.

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection needle valves to provide a total seal injection flow in the acceptable region of Technical Specification Figure 3.5.6-1. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Technical Specification Figure 3.5.6-1 are consistent with the accident analysis.

The limits on seal injection flow must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTION

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this condition, action must be taken to restore the flow to below its limit. The operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The completion time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the completion times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

When the actions cannot be completed within the required completion time, a controlled shutdown must be initiated. The completion time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown from MODE 3, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

Surveillance 4.5.6

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The differential pressure that is above the reference minimum value is established between the charging header (PT 62-92) and the RCS, and total seal injection flow is verified to be within the limits determined in accordance with the ECCS safety analysis (Ref. 3). The seal water injection flow limits are shown in Technical Specification Figure 3.5.6-1. The frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve surveillance frequencies. The frequency has proven to be acceptable through operating experience.

The requirements for charging flow vary widely according to plant status and configuration. When charging flow is adjusted, the positions of the air-operated valves, which control charging flow, are adjusted to balance the flows through the charging header and through the seal injection header to ensure that the seal injection flow to the RCPs is maintained between 8 and 13 gpm per pump. The reference minimum differential pressure across the seal injection needle valves ensures that regardless of the varied settings of the charging flow control valves that are required to support optimum charging flow, a reference test condition can be established to ensure that flows across the needle valves are within the safety analysis. The values in the safety analysis for this reference set of conditions are calculated based on conditions during power operation and they are correlated to the minimum ECCS flow to be maintained under the most limiting accident conditions.

As noted, the surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the surveillance is timely. Performance of this surveillance within the 4-hour allowance is required to maintain compliance with the provisions of Specification 4.0.3.

REFERENCES

1. FSAR, Chapter 6.3 "Emergency Core Cooling System" and Chapter 15.0 "Accident Analysis."
2. 10 CFR 50.46.
3. Westinghouse Electric Company Calculation CN-FSE-99-48

3/4.8.1 AND 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The electrically powered AC safety loads are separated into redundant load groups such that loss of any one load group will not prevent the minimum safety functions from being performed. Specification 3.8.1.1 requires two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System and four separate and independent diesel generator sets to be OPERABLE in MODES 1, 2, 3, and 4. These requirements ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated design basis accident.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident. Minimum required switchyard voltages are determined by evaluation of plant accident loading and the associated voltage drops between the transmission network and these loads. These minimum voltage values are provided to TVA's Transmission Operations for use in system studies to support operation of the transmission network in a manner that will maintain the necessary voltages. Transmission Operations is required to notify SQN Operations if it is determined that the transmission network may not be able to support accident loading or shutdown operations as required by 10 CFR 50, Appendix A, GDC-17. Any offsite power circuits supplied by that transmission network that are not able to support accident loading or shutdown operations are inoperable.

The unit station service transformers (USSTs) utilize auto load tap changers to provide the required voltage response for accident loading. The load tap changer associated with a USST is required to be functional and in "automatic" for the USST to supply power to a 6.9 kV Unit Board.

The inability to supply offsite power to a 6.9 kV Shutdown Board constitutes the failure of only one offsite circuit, as long as offsite power is available to the other load group's Shutdown Boards. Thus, if one or both 6.9 kV Shutdown Boards in a load group do not have an offsite circuit available, then only one offsite circuit would be inoperable. If one or more Shutdown Boards in each load group, or all four Shutdown Boards, do not have an offsite circuit available, then both offsite circuits would be inoperable. An "available" offsite circuit meets the requirements of GDC-17, and is either connected to the 6.9 kV Shutdown Boards or can be connected to the 6.9 kV Shutdown Boards within a few seconds.

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network (beginning at the switchyard) to one load group of Class 1E 6.9 kV Shutdown Boards (ending at the supply side of the normal or alternate supply circuit breaker). Each required offsite circuit is that combination of power sources described below that are normally connected to the Class 1E distribution system, or can be connected to the Class 1E distribution system through automatic transfer at the 6.9 kV Unit Boards.

The following offsite power configurations meet the requirements of LCO 3.8.1.1.a:
(Note that common station service transformer (CSST) B is a spare transformer with two sets of secondary windings that can be used to supply a total of two Start Buses for CSST A and/or CSST C, with each supplied Start Bus on a separate CSST B secondary winding.)

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

1. Two offsite circuits consisting of a AND b (no board transfers required; a loss of either circuit will not prevent the minimum safety functions from being performed):

   a. From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C), and CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

   b. From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

2. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.2)(b), or a.2) to b.1)(a) on a loss of (USSTs) 1A and 1B, OR relies on automatic transfer from alignment a.3) to b.2)(a), or a.4) to b.1)(b) on a loss of USSTs 2A and 2B):

   a. Normal power source alignments

      1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B);

      2) From the 500 kV switchyard through USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);

      3) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B); AND

      4) From the 161 kV switchyard through USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C).

   b. Alternate power source alignments

      1) From the 161 kV transmission network, through:

         (a) CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

         (b) CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); OR

      2) From the 161 kV transmission network, through:

         (a) CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), AND

         (b) CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

3. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 2 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):

   a. Normal power source alignments

      1) From the 161 kV switchyard through USST 2A to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B), and USST 2B to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C);

      2) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

      3) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

   b. Alternate power source alignments

      1) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

      2) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).

4. Two offsite circuits consisting of a AND b (relies on automatic transfer from alignment a.1) to b.1) and b.2) on a loss of the Unit 1 USSTs; a loss of alignment a.2) or a.3) will not prevent the minimum safety functions from being performed):

   a. Normal power source alignments

      1) From the 500 kV switchyard through USST 1A to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B), and USST 1B to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C);

      2) From the 161 kV transmission network, through CSST A (winding Y) to Start Bus 2A to 6.9 kV Shutdown Board 2B-B (through 6.9 kV Unit Board 2C); AND

      3) From the 161 kV transmission network, through CSST C (winding X) to Start Bus 2B to 6.9 kV Shutdown Board 2A-A (through 6.9 kV Unit Board 2B).

   b. Alternate power source alignments

      1) From the 161 kV transmission network, through CSST A (winding X) to Start Bus 1A to 6.9 kV Shutdown Board 1B-B (through 6.9 kV Unit Board 1C); AND

      2) From the 161 kV transmission network, through CSST C (winding Y) to Start Bus 1B to 6.9 kV Shutdown Board 1A-A (through 6.9 kV Unit Board 1B).

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Other offsite configurations are possible using different combinations of available USSTs and CSSTs, as long as the alignments are consistent with the analyzed configurations, and the alignments otherwise comply with the requirements of GDC 17.

For example, to support breaker testing, offsite power to the 6.9 kV Shutdown Boards can be realigned from normal feed to alternate feed. This would result in Shutdown Boards 1A-A and 2A-A being fed from Unit Boards 1A and 2A, respectively, and Shutdown Boards 1B-B and 2B-B being fed from Unit Boards 1D and 2D, respectively. The CSST being utilized as the alternate power source to one load group of Shutdown Boards would also be realigned (normally CSST A available to Shutdown Boards 1B-B and 2B-B or CSST C available to Shutdown Boards 1A-A and 2A-A, would be realigned to CSST A available to Shutdown Boards 1A-A and 2A-A or CSST C available to Shutdown Boards 1B-B and 2B-B).

LCO 3.8.1.1 is modified by Note @ that specifies CSST A and CSST C are required to be available via automatic transfer at the associated 6.9 KV Unit Boards, when USST 2A and USST 2B are being utilized as normal power sources to the offsite circuits. (Note that CSST B can be substituted for CSST A or CSST C.) This offsite power alignment is consistent with Configuration 3, as stated above. Note @ remains in effect until November 30, 2013, or until the USST modifications are implemented on Units 1 and 2, whichever occurs first. (The scheduled startup from the Unit 1 fall 2013 refueling outage is November 2013.) *Following expiration of Note @, Configuration 3 can continue to be used.*

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The footnote for Action b of LCO 3.8.1.1 requires completion of a determination that the OPERABLE diesel generators are not inoperable due to common cause failure or performance of Surveillance 4.8.1.1.2.a.4 if Action b is entered. The intent is that all diesel generator inoperabilities must be investigated for common cause failures regardless of how long the diesel generator inoperability persists.

Action b of LCO 3.8.1.1 is further modified by a second note which precludes making more than one diesel generator inoperable on a pre-planned basis for maintenance, modifications, or surveillance testing. The intent of this footnote is to explicitly exclude the flexibility of removing a diesel generator set from service as a part of a pre-planned activity. While the removal of a diesel generator set (A or B train) is consistent with the initial condition assumptions of the accident analysis, this configuration is judged as imprudent. The term pre-planned is to be taken in the context of those activities which are routinely scheduled and is not relative to conditions which arise as a result of emergent or unforeseen events. As an example, this footnote is not intended to preclude the actions necessary to perform the common mode testing requirements required by Action b. As another example, this footnote is not intended to prevent the required surveillance testing of the diesel generators should one diesel generator maintenance be unexpectedly extended and a second diesel generator fall within its required testing frequency. Thus, application of the note is intended for pre-planned activities.

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

In addition, this footnote is intended to apply only to those actions taken directly on the diesel generator. For those actions taken relative to common support systems (e.g. ERCW), the support function must be evaluated for impact on the diesel generator.

The action to determine that the OPERABLE diesel generators are not inoperable due to common cause failures provides an allowance to avoid unnecessary testing of OPERABLE diesel generators. If it can be determined that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generators, Surveillance Requirement 4.8.1.1.2.a.4 does not have to be performed. If the cause of inoperability exists on other diesel generator(s), the other diesel generator(s) would be declared inoperable upon discovery and Action e of LCO 3.8.1.1 would be entered as applicable. Once the common failure is repaired, the common cause no longer exists, and the action to determine inoperability due to common cause failure is satisfied. If the cause of the initial inoperable diesel generator cannot be confirmed not to exist on the remaining diesel generators, performance of Surveillance 4.8.1.1.2.a.4 suffices to provide assurance of continued OPERABILITY of the other diesel generators.

According to Generic Letter 84-15, 24 hours is reasonable to confirm that the OPERABLE diesel generators are not affected by the same problem as the inoperable diesel generator.

Action f prohibits the application of LCO 3.0.4.b to an inoperable diesel generator. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable diesel generator and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

With the minimum required AC power sources not available, it is required to suspend CORE ALTERATIONS and operations involving positive reactivity additions that could result in loss of required SDM (Mode 5) or boron concentration (Mode 6). Suspending positive reactivity additions that could result in failure to meet minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than or equal to that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

The requirements of Specification 3.8.2.1 provide those actions to be taken for the inoperability of A.C. Distribution Systems. Action a of this specification provides an 8-hour action for the inoperability of one or more A.C. boards. Action b of this specification provides a relaxation of the 8-hour action to 24-hours provided the Vital Instrument Power Board is inoperable solely as a result of one inoperable inverter and the board has been energized within 8 hours. In this condition the requirements of Action a do not have to be applied. Action b is not intended to provide actions for inoperable inverters, which is addressed by the operability requirements for the boards, and is included only for relief from the 8-hour

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

action of Action a when only one inverter is affected. More than one inverter inoperable will result in the inoperability of the associated 120 Volt A.C. Vital Instrument Power Board(s) in accordance with Action a. With more than one inverter inoperable entry into the actions of TS 3.0.3 is not applicable because Action a includes provisions for multiple inoperable inverters as attendant equipment to the boards.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137 "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979. The surveillance requirements for the diesel generator load-run test and the 24-hour endurance and margin test are in accordance with Regulatory Guide 1.9, Revision 3, July 1993, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plant." During the diesel generator endurance and margin surveillance test, momentary transients outside the kw and kvar load ranges do not invalidate the test results. Similarly, during the diesel generator load-run test, momentary transients outside the kw load range do not invalidate the test results.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. 6800 volts is the minimum steady state output voltage and the 10 second transient value. 6800 volts is 98.6% of nominal bus voltage of 6900 volts and is based on the minimum voltage required for the diesel generator supply breaker to close on the 6.9 kV Shutdown Board. The specified maximum steady state output voltage of 7260 volts is based on the degraded over voltage relay setpoint and is equivalent to 110% of the nameplate rating of the 6600 volt motors. The specified minimum and maximum frequencies of the diesel generator are 58.8 Hz and 61.2 Hz, respectively. These values are equal to $\pm$ 2% of the 60 Hz nominal frequency and are derived from the recommendations given in regulatory Guide 1.9.

Where the SRs discuss maximum transient voltages during load rejection testing, the following is applicable. The maximum transient voltage of 8880 volts represents a conservative limit to ensure the resulting voltage will not exceed a level that will cause component damage. It is based on the manufacturer's recommended high potential test voltage of 60% of the original factory high potential test voltage (14.8 kV). The diesel generator manufacturer has determined that the engine and/or generator controls would not experience detrimental effects for transient voltages < 9000 volts. The maximum transient voltage of 8276 volts is retained from the original technical specifications to ensure that the voltage transient following rejection of the single largest load is within the limits originally considered acceptable. It was based on 114% of 7260 volts, which is the Range B service voltage per ANSI-C84.1.

The Surveillance Requirement (SR) to transfer the power supply to each 6.9 kV Unit Board from the normal supply to the alternate supply demonstrates the OPERABILITY of the alternate supply to power the shutdown loads. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This SR is modified by two Notes. The reason for Note # is that, during operation with the reactor critical, performance of this SR for the Unit 2

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Unit Boards could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Note ## specifies that transfer capability is only required to be met for 6.9 kV Unit Boards that require normal and alternate power supplies. When both load groups are being supplied power by the USSTs, only the 6.9 kV Unit Boards associated with one load group are required to have normal and alternate power supplies. Therefore, only one CSST is required to be OPERABLE and available as an alternate power supply. Additionally, manual transfers between the normal supply and the alternate supply are not relied upon to meet the accident analysis. Manual transfer capability is verified to ensure the availability of a backup to the automatic transfer feature.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

The test listed below is a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If the results from this test is within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

tanks. This test is to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the test to exceed 31 days. The test, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel in accordance with D4057-1988 (ref.);

b. Verify in accordance with the test specified in ASTM D975-1990 (Ref.) that the sample has an absolute specific gravity at 60/60 degrees F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60 degrees F of ≥ 27 degrees and ≤ 39 degrees, a kinematic viscosity at 40 degrees C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125 degrees F; and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-1986 (Ref.).

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-1990 (Ref.) are met, except that the analysis for sulfur may be performed in accordance with ASTM D1552-1990 (Ref.) or ASTM D2622-1987 (Ref.). The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on D/G operation. This surveillance ensures availability of high quality fuel oil for the D/Gs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-94, Method A (Ref.). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each of the four interconnected tanks which comprise a 7-day tank must be considered and tested separately.

The frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between frequency intervals.

References:

ASTM Standards

D4057-1988, “Practice for manual sampling of petroleum and petroleum Products.”

D975-1990, “Standard Specifications for Diesel Fuel oils.”

D4176-1986, “Free Water and Particulate Contamination in Distillate Fuels.”

D1552-1990, “Standard Test Method for Sulfur in Petroleum Products (High Temperature Method).”

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

D2622-1987, "Standard Test Method for Sulfur in Petroleum Products (X-Ray Spectrographic Method)."

D2276-1994, "Standard Test Method for Particulate Containment in Aviation Turbine Fuels."

D1298-1985, "Standard Test Method for Density, Specific Gravity, or API Gravity of Crude Petroleum and Liquid Petroleum Products by Hydrometer Method."

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

This specification is deleted.