Enclosure 3
to NG-19-0024

DVD-ROM of the DAEC UFSAR Revision 25
PUBLIC VERSION

Enclosure 4
to NG-19-0024

DAEC Technical Requirements Manual

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T 3.3 INSTRUMENTATION

T 3.3.3 Non-Type A, Non-Category 1 Post Accident Monitoring (PAM) Instrumentation

TLCO 3.3.3 The PAM Instrumentation for each Function in Table T3.3.3-1 shall be OPERABLE.

APPLICABILITY: According to Table T3.3.3-1

ACTIONS

--NOTE--
Separate Condition entry is allowed for each Function. For Function 1, separate Condition entry is allowed for each SV/SRV. For Function 2, separate Condition entry is allowed for each SRV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Function 1 channels inoperable.	A.1 ----------NOTE--------- A.1 and A.2 are only Applicable to S/RVs ---------------------------- Verify Function 2 instruments for the affected S/RV are OPERABLE.	Immediately
	AND	
	A.2. Monitor suppression pool temperature.	Once per 12 hours
	AND	
	A.3 Restore channel to OPERABLE status.	30 days

(continued)

Table T3.3.3-1 (page 1 of 1)
Non-Type A, Non-Category 1 PAM Instrumentation

FUNCTION	Applicable MODES or other specified conditions	Minimum number of OPERABLE instrument channels	Surveillance Requirements
1. Safety/Safety Relief Valve Position Indicator (Primary Detector)	1, 2	2/valve	TSR 3.3.3.2 TSR 3.3.3.3 TSR 3.3.3.6
2. Safety/ Relief Valve Position Indicator (Backup Thermocouple)	1, 2	1/valve	TSR 3.3.3.2 TSR 3.3.3.5
3. Reactor Building Exhaust Stack Extended Range Effluent Radiation Monitor	1, 2	1	TSR 3.3.3.2 TSR 3.3.3.4
4. Turbine Building Exhaust Stack Extended Range Effluent Radiation Monitor	1, 2	1	TSR 3.3.3.2 TSR 3.3.3.4
5. Offgas Stack Extended Range Effluent Radiation Monitor	1, 2	1	TSR 3.3.3.2 TSR 3.3.3.4
6. Containment Water Level Monitor	1, 2	2	TSR 3.3.3.2 TSR 3.3.3.5
7. Drywell and Suppression Chamber – H_2 Analyzer	1, 2	1	TSR 3.3.3.2 TSR 3.3.3.6
8. Drywell and Suppression Chamber – O_2 Analyzer	1, 2	1	TSR 3.3.3.2 TSR 3.3.3.6

T 3.7 PLANT SYSTEMS

T 3.7.3 Structural Integrity

TLCO 3.7.3 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-------------------NOTE------------- Not applicable to component(s) that are isolated from service. --- A. Requirements of the LCO not met for Class 1 or Class 2 component in MODES 1, 2, and 3.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours
B. Requirements of the LCO for Class 1 or Class 2 component(s) not met in other than MODES 1, 2, and 3.	B.1 Restore structural integrity of component(s) to within limits. OR B.2 Isolate affected component(s).	Prior to entering MODE 2 or 3 from MODE 4 Prior to entering MODE 2 or 3 from MODE 4

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Requirements of the LCO not met for Class 3 component(s).	C.1 Initiate action to restore the structural integrity of component(s) to within limits.	Immediately
	OR	
	C.2 Initiate action to isolate affected component(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.3.1	Inservice inspection shall be performed in accordance with the requirements for ASME Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a, and as modified by NRC approved alternate measures.	In accordance with the Inservice Inspection program
TSR 3.7.3.2	The inservice inspection program for piping identified in NRC Generic Letter 88-01 and Supplement 1 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter and supplement, and the NRC Safety Evaluation of BWRVIP-75.	In accordance with the Augmented Inspection Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.11.1.1	Demonstrate the circuitry associated with the detector alarms is Functional.	12 months
TSR 3.11.1.2	Each smoke detector listed in Tables T3.11.1-1 and T3.11.1-2 shall be demonstrated Functional by performance of smoke testing.	12 months

Table 4.0-3a (RPS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.1.1.1	RPS CHANNEL CHECK	RPS Functions: 1a, 2a, 2b, & 4	3.0.0-01 3.0.0-03	12 Hours	No	
3.3.1.1.2	APRM Gain Adjustments	RPS Functions: 2b & 2c	3.0.0-01	24 Hours	**Yes**	
3.3.1.1.3	Scram Contactor Functional Test	RPS Functions: 2b, 2c, 2d, 3, 4, 5, 6, 7a, 7b, 8, & 9	3.3.1.1-22	31 Days	No	AR 2193083
3.3.1.1.4	CHANNEL FUNCTIONAL TEST	RPS Functions: 1a, 1b, & 2a	3.3.1.1-06 3.3.1.1-07 3.3.1.1-25 3.3.1.1-26 3.3.1.1-31 3.3.1.1-35 3.3.1.1-38	7 Days	**Yes**	
3.3.1.1.5	CHANNEL FUNCTIONAL TEST	RPS Functions: 1a* & 1b*	3.3.1.1-06 3.3.1.1-07 3.3.1.1-31	7 Days	No	
3.3.1.1.7	Verify IRM/APRM channel overlap	RPS Functions: 1a & 2a	3.3.1.1-30	7 Days	**Yes**	
3.3.1.1.8	LPRM Calibration	RPS Functions: 2a, 2b & 2c	3.3.1.1-24 3.3.1.1-37	1000 MWD/T	No	

Table 4.0-3h (ECCS Instrumentation)

TS SR #	**Subject**	**TS SSC/Function**	**STP #**	**Frequency**	**SR Frequency Note?**	**AR#**
3.3.5.1.1	CHANNEL CHECK	ECCS Functions: 1a, 2a, 2d, 2g, 2i, 2.j, 3a, 3c, 4a, 4c, 5a, 5c	3.0.0-01	24 Hours	No	AR 2205811
3.3.5.1.2	CHANNEL FUNCTIONAL TEST	ECCS Functions: 2d, 2g, 2h, 2i, 2j	3.3.5.1-01 3.3.5.1-02 3.3.5.1-03 3.3.5.1-04 3.3.5.1-05 3.3.5.1-12 3.3.5.1-13 3.3.5.1-19 3.3.5.1-20 3.3.5.1-21 3.3.5.1-22	92 Days	No	AR 2174652

Table 4.0-3i (RPV WIC Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.5.2.1	CHANNEL CHECK	RPV WIC Functions: 3a, 4a	3.0.0-01 3.0.0-03	12 Hours	No	AR 2205811
3.3.5.2.2	CHANNEL FUNCTIONAL TEST	RPV WIC Functions: 1a, 1b, 2a, 2b, 3a, 4a	3.3.1.1-04 3.3.1.1-05 3.3.5.1-10 3.3.5.1-11 3.3.5.1-35 3.3.5.1-36 3.3.6.1-10 3.3.6.1-11 3.5.1-01 3.5.1-02 3.5.1-11 3.5.1-12	92 Days	No	AR 2205811

Table 4.0-3j (RCIC Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.5.3.1	CHANNEL CHECK	RCIC Function: 1, 2	3.0.0-01	24 Hours	No	
3.3.5.3.2	CHANNEL FUNCTIONAL TEST	RCIC Function: 1, 2, 3	3.3.1.1-04 3.3.1.1-05 3.3.5.1-01 3.3.5.1-02 3.3.5.1-23 3.3.5.1-24	92 Days	No	
3.3.5.3.3	CHANNEL CALIBRATION	RCIC Function: 1, 2	3.3.1.1-05 3.3.5.1-02	12 Months	No	
3.3.5.3.4	CHANNEL CALIBRATION	RCIC Function: 3	3.3.5.1-24	24 Months	No	
3.3.5.3.5	LOGIC SYSTEM FUNCTIONAL TEST	RCIC Function: 1, 2, 3	3.3.1.1-05 3.3.5.1-02 3.3.5.1-24 3.3.5.1-37 3.5.3-04 3.5.3-05	24 Months	No	

Table 4.0-3k (PCIS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.6.1.1	CHANNEL CHECK	PCIS Function: 1a, 1c, 2a, 2b, 5e, 6b	3.0.0-01 3.0.0-03	12 Hours	No	
3.3.6.1.2	CHANNEL CHECK	PCIS Function: 1e, 1f, 2c, 2d, 2e, 3e, 3g, 3h, 3i, 4e, 4g, 4h, 4i, 5a, 5b, 5c, 5f	3.0.0-01 3.3.6.1-38	24 Hours	No	
3.3.6.1.3	CHANNEL FUNCTIONAL TEST	PCIS Function: 7a	3.3.6.1-07 3.3.6.1-08	31 Days	No	

Table 4.0-3k (PCIS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.6.1.4	CHANNEL FUNCTIONAL TEST	PCIS Function: 1a, 1b, 1c, 1d, 1e, 1f, 2a, 2b, 2c, 2d, 2e, 3a, 3b, 3c, 3d, 3e, 3f, 3g, 3h, 3i, 4a, 4b, 4c, 4d, 4e, 4f, 4g, 4h, 4i, 5a, 5b, 5c, 5e, 5f, 6a, 6b, 6c	3.3.1.1-02 3.3.1.1-03 3.3.1.1-04 3.3.1.1-05 3.3.5.1-08 3.3.5.1-09 3.3.6.1-01 3.3.6.1-02 3.3.6.1-03 3.3.6.1-04 3.3.6.1-05 3.3.6.1-10 3.3.6.1-11 3.3.6.1-12 3.3.6.1-13 3.3.6.1-15 3.3.6.1-16 3.3.6.1-17 3.3.6.1-18 3.3.6.1-19 3.3.6.1-20 3.3.6.1-21 3.3.6.1-22 3.3.6.1-23 3.3.6.1-24 3.3.6.1-25 3.3.6.1-26 3.3.6.1-27 3.3.6.1-28 3.3.6.1-29 3.3.6.1-30 3.3.6.1-31 3.3.6.1-32 3.3.6.1-33 3.3.6.1-34 3.3.6.1-40 3.3.6.1-41 3.3.6.1-43 3.3.6.1-44 3.3.6.1-45 3.3.6.1-46 3.3.6.1-47 3.3.6.1-48	92 Days	No	

Table 4.0-3k (PCIS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.6.1.5	CHANNEL CALIBRATION	PCIS Function: 1b, 1c, 6a	3.3.6.1-01 3.3.6.1-02 3.3.6.1-03	92 Days	No	
3.3.6.1.6	CHANNEL CALIBRATION	PCIS Function: 4c	3.3.6.1-31	184 Days	No	
3.3.6.1.7	CHANNEL CALIBRATION	PCIS Function: 1e, 1f, 5e	3.3.6.1-04 3.3.6.1-10 3.3.6.1-15	12 Months	No	
3.3.6.1.8	CHANNEL CALIBRATION	PCIS Function: 1a, 1d, 2a, 2b, 2c, 2d, 2e, 3a, 3b, 3c, 3d, 3e, 3f, 3g, 3h, 3i,4a, 4b, 4d, 4e, 4f, 4g, 4h, 4i, 5a, 5b, 5c, 5f, 6b, 6c, 7a	3.3.1.1-03 3.3.1.1-05 3.3.5.1-08 3.3.6.1-08 3.3.6.1-10 3.3.6.1-13 3.3.6.1-17 3.3.6.1-19 3.3.6.1-21 3.3.6.1-23 3.3.6.1-25 3.3.6.1-27 3.3.6.1-29 3.3.6.1-34 3.3.6.1-40 3.3.6.1-44 3.3.6.1-46 3.3.6.1-48	24 Months	No	

Table 4.0-3k (PCIS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.6.1.9	LOGIC SYSTEM FUNCTIONAL TEST	PCIS Function: 1a, 1b, 1c, 1d, 1e, 1f, 2a, 2b, 2c, 2d, 2e, 3a, 3b, 3c, 3d, 3e, 3f, 3g, 3h, 3i, 4a, 4b, 4c, 4d, 4e, 4f, 4g, 4h, 4i, 5a, 5b, 5c, 5d, 5e, 5f, 6a, 6b, 6c, 7a	3.1.7-02 3.3.1.1-03 3.3.1.1-05 3.3.5.1-08 3.3.5.1-29 3.3.6.1-01 3.3.6.1-02 3.3.6.1-03 3.3.6.1-04 3.3.6.1-06 3.3.6.1-08 3.3.6.1-09 3.3.6.1-10 3.3.6.1-13 3.3.6.1-14 3.3.6.1-15 3.3.6.1-17 3.3.6.1-19 3.3.6.1-21 3.3.6.1-23 3.3.6.1-25 3.3.6.1-27 3.3.6.1-29 3.3.6.1-31 3.3.6.1-34 3.3.6.1-40 3.3.6.1-42 3.3.6.1-44 3.3.6.1-46 3.3.6.1-48 3.3.6.1-49 3.3.6.1-50 3.3.6.1-51 3.3.6.1-52 3.6.1.3-06	24 Months	No	

Table 4.0-31 (SCIS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.6.2.1	CHANNEL CHECK	SCIS Function: 1	3.0.0-01 3.0.0-03	12 Hours	No	
3.3.6.2.2	CHANNEL CHECK	SCIS Function: 3, 4	3.0.0-01 3.0.0-03 3.3.6.1-38	24 Hours	No	
3.3.6.2.3	CHANNEL FUNCTIONAL TEST	SCIS Function: 1, 2, 3, 4	3.3.1.1-02 3.3.1.1-03 3.3.1.1-04 3.3.1.1-05 3.3.6.1-21 3.3.6.1-22 3.3.6.1-23 3.3.6.1-24	92 Days	No	
3.3.6.2.4	CHANNEL CALIBRATION	SCIS Function: 1, 2, 3, 4	3.3.1.1-03 3.3.1.1-05 3.3.6.1-21 3.3.6.1-23	24 Months	No	
3.3.6.2.5	LOGIC SYSTEM FUNCTIONAL TEST	SCIS Function: 1, 2, 3, 4	3.3.1.1-03 3.3.1.1-05 3.3.6.1-21 3.3.6.1-23 3.3.6.1-50 3.3.6.1-51	24 Months	No	

Table 4.0-3m (LLS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.6.3.1	CHANNEL FUNCTIONAL TEST (outside DW)	LLS Function: 3	3.3.6.3-01 3.3.6.3-02	92 Days	No	
3.3.6.3.2	CHANNEL FUNCTIONAL TEST	LLS Function: 1, 2	3.3.1.1-01 3.3.6.3-03 3.3.6.3-04	92 Days	No	
3.3.6.3.3	CHANNEL CALIBRATION	LLS Function: 1	3.3.1.1-01	92 Days	No	
3.3.6.3.4	CHANNEL CALIBRATION	LLS Function: 2	3.3.6.3-04	184 Days	No	
3.3.6.3.5	CHANNEL CALIBRATION	LLS Function: 3	3.3.6.3-02	24 Months	No	
3.3.6.3.6	LOGIC SYSTEM FUNCTIONAL TEST	LLS Function: 1, 2, 3	3.3.1.1-01 3.3.6.3-02 3.3.6.3-04 3.3.6.3-05	24 Months	No	

Table 4.0-3n (SFU Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.7.1.1	CHANNEL CHECK	SFU Rad Monitors	3.3.6.1-38	24 Hours	No	
3.3.7.1.2	CHANNEL FUNCTIONAL TEST	SFU Rad Monitors	3.3.7.1-01 3.3.7.1-02	92 Days	No	
3.3.7.1.3	CHANNEL CALIBRATION	SFU Rad Monitors	3.3.7.1-02	24 Months	No	
3.3.7.1.4	LOGIC SYSTEM FUNCTIONAL TEST	SFU Rad Monitors	3.3.7.1-02 3.7.4-01	24 Months	No	

Table 4.0-3o (LOP Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.8.1.1	CHANNEL FUNCTIONAL TEST	LOP Function: 2a, 2b	3.3.8.1-01 3.3.8.1-02	31 Days	No	
3.3.8.1.2	CHANNEL FUNCTIONAL TEST	LOP Function: 1a, 3	3.3.8.1-03 3.3.8.1-04 3.3.8.1-05	12 Months	No	
3.3.8.1.3	CHANNEL CALIBRATION	LOP Function: 2a, 2b, 3	3.3.8.1-02 3.3.8.1-05	12 Months	No	
3.3.8.1.4	CHANNEL CALIBRATION	LOP Function: 1a	3.3.8.1-04	24 Months	No	
3.3.8.1.5	LOGIC SYSTEM FUNCTIONAL TEST	LOP Function: 1a, 2a, 2b, 3	3.3.8.1-04 3.3.8.1-05 3.3.8.1-06	24 Months	No	AR 2065602 AR 2133658
			3.7.2-01	24 Months Staggered Test Basis		

Table 4.0-3p(RPS-EPA Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.8.2.1	CHANNEL FUNCTIONAL TEST	RPS-EPA	3.3.8.2-01 3.3.8.2-02	184 Days	**Yes**	
3.3.8.2.2	CHANNEL CALIBRATION	RPS-EPA	3.3.8.2-01	24 Months	No	
3.3.8.2.3	System Functional Test	RPS-EPA	3.3.8.2-01	24 Months	No	

Table 4.0-4 (Reactor Coolant System)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.4.9.7	RPV Flange Temperature Check – ≤100°F	RCS P/T Limits	3.0.0-01 3.0.0-03	12 Hours	**Yes**	
3.4.10.1	Steam Dome Pressure Check	Reactor Steam Dome Pressure	3.0.0-01	12 Hours	No	

3.5.1.1	Piping locations susceptible to gas accumulation are sufficiently full of water	HPCI, Core Spray, LPCI	3.5.1-13 3.5.1-14 3.5.1-15	31 Days	No	
3.5.1.2	Valve Position Checks	HPCI, Core Spray, LPCI	3.0.0-02	31 Days	**Yes**	
3.5.1.3	Accumulator Supply Check	ADS	3.0.0-04	31 Days	No	
3.5.1.6	Pump Flowrate Check – Low Pressure	HPCI	3.5.1-06	24 Months	**Yes**	
3.5.1.7	Simulated Auto Actuation Test	HPCI, Core Spray, LPCI	3.3.5.1-15 3.3.5.1-29 3.3.5.1-30 3.3.5.1-37 3.5.1-03 3.5.1-04 3.5.1-05 3.5.1-06 3.5.1-07 3.5.1-10 3.6.1.3-06 3.8.7-01	24 Months	No	AR 2065602
3.5.1.8	Simulated Auto Actuation Test	ADS	3.3.5.1-16	24 Months	No	
3.5.1.9	Valves Manually Open Check	ADS	3.4.3-03	24 Months	No	

Table 4.0-5 (ECCS, RPV WIC, and RCIC)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.5.2.1	DRAIN TIME ≥ 36 hours	RPV WIC	3.0.0-01 3.0.0-03	12 Hours	No	AR 2205811
3.5.2.2	Suppression Pool Level Check	RPV WIC - LPCI	3.0.0-01 3.0.0-03	12 Hours	No	
3.5.2.3	CST & Suppression Pool Level Check	RPV WIC - Core Spray	3.0.0-01 3.0.0-03	12 Hours	No	
3.5.2.4	Piping locations susceptible to gas accumulation are sufficiently full of water	RPV WIC - Core Spray, LPCI	3.0.0-01 3.5.1-13 3.5.1-14 3.5.1-15	31 Days	No	
3.5.2.5	Valve Position Checks	RPV WIC - Core Spray, LPCI	3.0.0-02	31 Days	**Yes**	
3.5.2.6	Operate required ECCS subsystem through recirculation line	RPV WIC - Core Spray, LPCI	3.5.1-01 3.5.1-02 3.5.1-11 3.5.1-12	92 Days	No	AR 2205811
3.5.2.7	Simulated Auto Actuation Test	RPV WIC isolation valves	3.3.6.1-09 3.3.6.1-14 3.6.1.3-06	24 Months	No	AR 2205811
3.5.2.8	Manually operate required ECCS subsystem	RPV WIC - Core Spray, LPCI	3.5.1-01 3.5.1-02 3.5.1-11 3.5.1-12	24 Months	**Yes**	AR 2205811

Table 4.0-5 (ECCS, RPV WIC, and RCIC)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.5.3.1	Piping locations susceptible to gas accumulation are sufficiently full of water	RCIC	3.5.3-08	31 Days	No	
3.5.3.2	Valve Position Checks	RCIC	3.0.0-02	31 Days	**Yes**	
3.5.3.4	Pump Flowrate Check – Low Pressure	RCIC	3.5.3-03	24 Months	**Yes**	
3.5.3.5	Simulated Auto Actuation Test	RCIC	3.5.3-02 3.5.3-03 3.5.3-04 3.5.3-05 3.5.3-07	24 Months	No	

TB 3.3 INSTRUMENTATION

TB 3.3.3 Non-Type A, Non-Category 1 Post Accident Monitoring (PAM) Instrumentation

BASES

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. The PAM instrumentation supplements existing instrumentation that was designed to monitor primarily the normal operational ranges of these parameters.

Each of the Safety/Safety Relief Valve Position Indicator (primary detector) instrument channels is comprised of three instruments (pressure switches) which are arranged in a "two out of three" logic. When a channel of Safety/Relief Valve Position Indication is inoperable, the suppression pool water temperature shall be monitored to observe any unexplained temperature increase which might be indicative of an open safety/relief valve.

The Extended Range Effluent Radiation Monitor instrumentation channels consist of the local microcomputer cabinet indicator and the data recording device. To be considered OPERABLE, channel data must be retrievable from the EMS software on the plant process computer, directly from the plant process computer itself, or from an Effluent Monitoring System recorder. These monitors shall be calibrated by means of a built-in check source or a known radioactive source. The requirements listed apply only to the high range monitors.

With the change in regulations (Reference 1), a combustible gas mixture is no longer postulated to occur as part of any design basis accident, but only as a result of severe accident scenarios. Consequently, the functional requirements of the hydrogen and oxygen monitoring instrumentation are downgraded from their original Reg. Guide 1.97, Category 1 qualification to non-safety-related instrumentation. Thus, only one channel of instrumentation is required for each monitoring function. However, both the drywell and suppression chamber atmospheres are required to be capable of being monitored. The normal condition for the drywell and suppression chamber hydrogen and oxygen monitors is in the Standby mode, but the instruments should be capable of being made functional shortly after the initiation of the event (as directed by the Emergency Response Organization) and capable of continuously monitoring the containment atmosphere. Instrument OPERABILITY is demonstrated by routine CHANNEL CHECKS and periodic CHANNEL CALBRATIONS.

(continued)

TB 3.7 PLANT SYSTEMS

TB 3.7.3 Structural Integrity

BASES

A pre-service inspection of Nuclear Class I components was conducted to ensure freedom from defects greater than code allowance; in addition, this served as a reference base for future inspections. Prior to operation, the Reactor Coolant System (RCS) as described in Article IS-120 of Section XI of the ASME Boiler and Pressure Vessel Code was inspected to provide assurance that the system was free of gross defects. In addition, the facility was designed such that gross defects should not occur throughout plant life. The pre-service inspection program was based on the 1970 Section XI of the ASME Code for in-service inspection. This inspection plan was designed to reveal problem areas (should they occur) before a leak in the coolant system could develop. The program was established to provide reasonable assurance that no Loss of Coolant Accident (LOCA) would occur at the DAEC as a result of leakage or breach of pressure-containing components and piping of the RCS, portions of the Emergency Core Cooling System (ECCS), and portions of the reactor coolant associated auxiliary systems.

A pre-service inspection was not performed on Nuclear Class II components since it was not required at that stage of DAEC construction when it would have been used. For these components, shop and in-plant examination records of components and welds will be used as a basis for comparison with in-service inspection data.

Visual examinations for leaks will be made periodically on ASME Section XI Class 1, 2, and 3 systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the ASME Section XI boundaries.

The type of examination planned for each component depends on the location, accessibility, and type of potential defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface examinations are planned where practical, and where added sensitivity is required. Ultrasonic examinations or radiography shall be used where defects can occur in concealed surfaces. Section 5.2.4 of the UFSAR provides details of the inservice inspection program.

An augmented inspection program was implemented to address concerns relating to Intergranular Stress Corrosion Cracking (IGSCC) in reactor coolant piping made of austenitic stainless steel. The augmented inspection program conforms to the NRC Staff's positions set forth in Generic Letter 88-01 and Supplement 1, NUREG 0313, Revision 2 and the NRC Safety Evaluation of BWRVIP-75 for inspection schedule, inspection methods and personnel, and inspection sample expansion.

(continued)

BASES (continued)

Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

References for Bases Section TB 3.7.3

1. ASME Boiler and Pressure Vessel Code, Section XI, 1970 and 1980 Editions.
2. Generic Letter 88-01 and Supplement 1.
3. NUREG 0313, Revision 2.
4. UFSAR Section 3.1.2.2.6, Criterion 15, Rx Coolant System Design.
5. UFSAR Section 5.2, Integrity of RCPB.
6. UFSAR Section 3.8.4.5, Structural Acceptance Criteria.
7. NRC Safety Evaluation of BWRVIP-75.

BASES (continued)

and the status of the fire protection water system including fire pump running, fire pump trouble, and low fire water system pressure.

NFPA 805, 3.8.2 Requirement:
If automatic fire detection is required to meet the performance or deterministic requirements of Chapter 4, then these devices shall be installed in accordance with NFPA 72, National Fire Alarm Code, and its applicable appendixes.

NFPA 805, 3.8.2 Basis:
The adequacy of detector placement and spacing is evaluated in code evaluations FPE-S00-001, FPE-S00-002, FPE-S02-001, and FPE-S02-002. Note: NFPA 72 did not exist at the time fire detection was installed. The governing code was NFPA 72E-1974.

SURVEILLANCE REQUIREMENTS

Smoke tests of smoke detectors are performed using guidance from NFPA, and insurance or manufacturers recommendations. Smoke detectors are tested "in-place" using inert gas or by use of chemical smoke generators. Based on EPRI TR 1006756 "Fire Protection Equipment Surveillance Optimization and Maintenance Guide," Section C.4, a fire detector's rated sensitivity is not a reliable measure of the detector's performance under real-world fire scenarios. Sensitivity testing is of limited value in assuring adequate performance. Functional smoke testing is an adequate test of the detector and associated circuitry. Industry OE indicates other plants have eliminated or have never performed sensitivity testing except after cleaning the detector. Periodic sensitivity testing may still be performed on those detectors that actuate suppression or other plant systems to prevent nuisance system actuations, but will not be used to demonstrate functionality. Periodic sensitivity testing has been removed as a means of demonstrating functional performance of TRM and non-TRM required fire detectors.

Circuit checks by initiation of end of the line or end of the branch detectors will more thoroughly test the parallel circuits than testing on a rotating detector basis. This test is not a detector test, but is a test to simulate the effect of electrical supervision as defined by NFPA 72.

(continued)

BASES (continued)

References for Bases Section TB 3.11.1

1. NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition.
2. License Amendment Request, August 5, 2011, Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition (ML11221A280)
3. FP-AB-100 Fire Protection Plan
4. FPE-S00-001 Rev. 1 - Evaluation of Smoke Detector Installation in Fire Zone 10A (Battery Corridor)
5. FPE-S00-002 Rev. 1 - Evaluation of Smoke Detector Spacing and Location in the Control Room Back-Panel Area (Fire Zone 12A)
6. FPE-S02-001 Rev. 3 - Fire Detection Code Compliance Evaluation for Fire Plan Required and Fire PRA Higher Risk Areas
7. FPE-S02-002 Rev. 3 - Fire Detection Code Compliance Evaluation for Lower Risk and Non-Fire Plan Required Areas Inside Protected Area

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B 3.8-48	08/15/17	174	B 3.9-19	04/09/04	044A	B 3.10-33	08/01/98	223
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Technical Specifications Bases

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with the UFSAR (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, reactivity anomaly is used as a measure of the predicted versus measured (i.e., monitored) core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, and whatever neutron poisons (mainly xenon and samarium) that are

(continued)

BASES

BACKGROUND (continued)	present in the fuel. The predicted core reactivity, as represented by core k_{eff}, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The monitored core k_{eff} is calculated by the core monitoring system at actual plant conditions and is compared to the predicted value at the same cycle exposure.
APPLICABLE SAFETY ANALYSES	Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity. The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted core k_{eff} for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict core k_{eff} may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured core k_{eff} from the predicted core k_{eff} that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred. Reactivity anomalies satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted core k_{eff} of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, reactivity anomaly is not required during these conditions.

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally

(continued)

BASES

ACTIONS

A.1 (continued)

reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted core k_{eff} is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The Plant Process Computer calculates the core k_{eff} for the reactor conditions obtained from plant instrumentation. A comparison of the monitored core k_{eff} to the predicted core k_{eff} at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.2.1 (continued) subsequent replacement with a new control rod or a control rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted core k_{eff} can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. Additionally, the Reactor Engineer or individual fulfilling this role will normally be involved with determining equilibrium xenon conditions. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.
REFERENCES	1. UFSAR, Sections 3.1.2.3.7, 3.1.2.3.9, and 3.1.2.3.10. 2. UFSAR, Chapter 15.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 26.2 gpm at a discharge pressure ≥ 1150 psig when pumping demineralized water to the test tank ensures that pump performance has not degraded below design values during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

(continued)

BASES (continued)

ACTIONS (continued)

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. .

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position. This SR is modified by a Note that allows the SR to be met for OPERABLE valves that are temporarily closed while performing the testing required by SR 3.1.8.2. The Note is necessary to avoid potential conflicts between the two SRs created by SR 3.0.1.

The Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The Frequency is based on operating experience and takes into account the level of redundancy in the system design as well as being in accordance with the INSERVICE TESTING PROGRAM.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.4.1, "Recirculation Loops Operating," allows the APRMs to be reading greater than actual THERMAL POWER to effectively lower the APRM Flow Biased High setpoints by 6.3% for single recirculation loop operation. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated power plus 6.3%. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 21.7% RTP is provided that requires the SR to be met only at ≥ 21.7% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 21.7% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 21.7% RTP, the Surveillance is required to have been satisfactorily performed within the previous Frequency, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 21.7% if the Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 21.7% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

There are four pairs of RPS automatic scram contactors (i.e., K14 relay contacts) with each pair associated with an automatic scram logic (A1, A2, B1, and B2). The automatic scram contactors can be functionally tested without the necessity of using an automatic scram

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

function trip. This functional test can be accomplished by placing the associated RPS Test Switch in the trip position, which will deenergize a pair of the automatic scram contactors and in turn, trip the associated RPS logic. The RPS Test Switches were not specifically credited in the accident analysis and thus, do not have any OPERABILITY requirements of their own. However, because the Manual Scram pushbuttons at the DAEC are not configured the same as the generic model (i.e., they are in a separate RPS logic - A3 and B3), the RPS Test Switches have been found to be functionally equivalent to the Manual Scram pushbuttons in the generic model for performing the functional test of the automatic scram contactors. If an RPS Test Switch(es) is (are) not available for performing this test, it is permissible to take credit for a CHANNEL FUNCTIONAL TEST of an automatic RPS trip function (i.e., SR 3.3.1.1.9), if performed within the required Frequency for this Surveillance, as it will also test the K14 relay contacts.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

As noted, SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9 and SR 3.3.1.1.13 (continued)

are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Frequency of SR 3.3.1.1.13 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at this Frequency.

SR 3.3.1.1.10

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.1.11, SR 3.3.1.1.12 and SR 3.3.1.1.14

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The CHANNEL CALIBRATION for Functions 5 and 8 shall consist of the physical inspection and actuation of these position switches.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

(Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR greater than or equal to the limit specified in the COLR, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR greater than or equal to the limit specified in the COLR, no RWE event will result in exceeding the MCPR SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE.

2. Rod Worth Minimizer

The RWM enforces a rod pattern which is consistent with the Banked Position Withdrawal Sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, 7, and 11. The standard BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 11) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 11 control rod insertion sequence for shutdown, the Rod Worth Minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion process, or it can be bypassed if it is not programmed to reflect the optional BPWS shutdown sequence, as perfmitted by the Applicability Note for the RWM in Table 3.3.2.1-1.

The RWM Function satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

Since the RWM is a hardwired system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod

(continued)

TABLE B 3.3.3.2-1 (PAGE 1 OF 2)

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

FUNCTION	REQUIRED NUMBER OF CHANNELS
<u>Instrument Parameter</u>	
1. Reactor Pressure Vessel Pressure Control	
a) Reactor Pressure	1
2. Reactor Pressure Vessel Inventory Control	
a) Reactor Level Wide Range	1
b) Reactor Level (Floodup)	1
c) B Core Spray Discharge Flow	1
3. Decay Heat Removal	
a) B RHR Loop Flow	1
b) B RHRSW Loop Flow	1
c) B RHR Heat Exchanger Shell Pressure	1
4. Safety Support Systems	
a) Torus Level	1
b) Torus Temperature	1
<u>Transfer/Control Functions</u>	
1. Reactor Pressure Vessel Pressure Control	
a) SRVs (Manual and Self-actuating)	2
b) SRVs (Self-actuating only)	(a)
c) SRVs (Self-actuating only)	1
d) Reactor Pressure Instrumentation	(b)
e) Emergency Fuses	(c)
2. Reactor Pressure Vessel Inventory Control	
a) B Core Spray MOVs	3
b) B Core Spray Pump	1
c) RCIC RPV High Level Shutdown/Restart	(d)
d) Outboard MSIVs	4
e) Closes Outboard MSL Drain MO-4424	1
f) HPCI RPV High Level Shutdown/Restart	1
g) Reactor Level Floodup Instrumentation	(b)
h) Reactor Level Wide Range and B Core Spray Discharge Flow Instrumentation	(e)
i) Emergency Fuses	(c)
3. Decay Heat Removal	
a) B RHR MOVs	7
b) B RHR Pumps	1
c) B RHRSW MOV and Closes RHR Radwaste Isolation MO-1937	1
d) B RHRSW Pumps	1
e) Closes RHR Cross-tie MO-2010	(d)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the ECCS are explicitly assumed in the safety analyses of References 1, 2 and 3. The ECCS is initiated to preserve the integrity of the fuel cladding by limiting the post LOCA peak cladding temperature to less than the 10 CFR 50.46 limits.

ECCS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the ECCS instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Table 3.3.5.1-1 is modified by a footnote which is added to show that certain ECCS instrumentation Functions also perform DG initiation.

Allowable Values are specified for each ECCS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel waterlevel), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., on-off sensor or bi-stable trip circuit) changes state. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a Reactor Vessel Water Level - Low Low Low (continued)

Reactor Vessel Water Level — Low Low Low signals are initiated from four level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Reactor Vessel Water Level — Low Low Low Allowable Value is chosen to allow time for the low pressure core flooding LCO, and systems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level — Low Low Low Function are only required to be OPERABLE when the ECCS are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation.

1.b, 2.b. Drywell Pressure — High

High pressure in the drywell could indicate a break in the Reactor Coolant Pressure Boundary (RCPB). The low pressure ECCS and associated DGs are initiated upon receipt of the Drywell Pressure — High Function in order to minimize the possibility of fuel damage. The Drywell Pressure — High Function, along with the Reactor Water Level — Low Low Low Function, is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c, 2.c. Reactor Steam Dome Pressure — Low (Injection Permissive) (continued)

Four channels of Reactor Steam Dome Pressure — Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems.

1.d, 2.f. Core Spray and Low Pressure Coolant Injection Pump Discharge Flow — Low (Bypass)

The minimum flow instruments are provided to protect the associated low pressure ECCS pump(s) from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valves (normally open for the CS System and normally closed for the LPCI System) receive an open signal when low flow is sensed, and automatically close when the flow rate is adequate to protect the associated pump. The LPCI and CS Pump Discharge Flow — Low Functions are assumed to be OPERABLE and capable of closing the minimum flow valves to ensure that the low pressure ECCS flow rates assumed during the transients and accidents analyzed in References 1, 2, and 3 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow switch per CS pump and one differential pressure switch for the two RHR pumps in each division are used to detect the associated subsystems' flow rates. The logic is arranged such that each differential pressure switch or flow switch causes its associated minimum flow valve to receive an open signal. The logic will close the minimum flow valve once the closure setpoint is exceeded. The LPCI minimum flow valves are time delayed such that the valves will not open for 10 seconds after the switches detect low flow. The time delay is provided by design to limit reactor vessel inventory loss during the startup of the RHR shutdown cooling mode although, typically, the minimum flow valves are prevented from opening when operating in the shutdown cooling mode. The Pump Discharge Flow — Low Allowable Values are high enough to ensure that the pump flow rate is sufficient to protect the pump, yet low enough to ensure

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d, 2.f. Core Spray and Low Pressure Coolant Injection Pump Discharge Flow-Low (Bypass) (continued)

that the closure of the minimum flow valve is initiated to allow the assumed flow into the core. Each channel of Pump Discharge Flow — Low Function (two CS channels, one per pump and two LPCI channels, one per loop) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude the ECCS function.* Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems.

2.d. Reactor Vessel Shroud Level — Low

The Reactor Vessel Shroud Level-Low Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes with a LPCI initiation signal still present. This function ensures: 1) that the permissive is removed prior to reaching two thirds core height when vessel level is decreasing, and 2) that the permissive is not restored until two thirds core height is reached when vessel level is increasing. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures to allow containment cooling/spray regardless of the level present in the shroud.

Reactor Vessel Shroud Level — Low signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Reactor Vessel Shroud Level — Low Allowable Value is chosen to allow the low pressure core flooding systems to activate and provide adequate cooling before allowing a manual transfer.

Four channels of the Reactor Vessel Shroud Level — Low Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the specified initiation time of the LPCI subsystems is not assumed, and other administrative controls are adequate to control the valves that this Function isolates (since the containment cooling mode of RHR is not required to be OPERABLE in MODES 4 and 5 and is normally not used).

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.e, 2.e Core Spray Pump Startup - Time Delay Relay and Low Pressure Coolant Injection Pump Start - Time Delay Relay

The purpose of these time delay relays is to stagger the start of the CS pumps and LPCI pumps that are in each of Divisions 1 and 2, thus limiting the starting transients on the 4.16 kV emergency buses. This Function is only necessary when power is being supplied from the standby power sources (DG). However, since the time delay does not degrade ECCS operation, it remains in the pump start logic at all times. The CS Pump and LPCI Pump Start — Time Delay Relays are assumed to be OPERABLE in the accident and transient analyses requiring ECCS initiation. That is, the analyses assume that the pumps will initiate when required and excess loading will not cause failure of the standby power sources.

There are two CS Pump Start - Time Delay Relays and four LPCI Pump Start - Time Delay Relays, one in each of the CS and RHR pump start logic circuits. While each time delay relay is dedicated to a single pump start logic, a single failure of a CS Pump Start - Time Delay Relay or of a LPCI Pump Start - Time Delay Relay could result in the failure of the three low pressure ECCS pumps, powered from the same emergency bus if the emergency bus is being powered by its associated DG, to perform their intended function within the assumed time (e.g., as in the case where two or more ECCS pumps on one emergency bus start simultaneously due to an inoperable time delay relay which, in turn cause the associated DG output breaker to trip open due to undervoltage conditions). This still leaves three of the six low pressure ECCS pumps OPERABLE; thus, the single failure criterion is met (i.e., loss of one instrument does not preclude ECCS initiation). The Allowable Value for the CS Pump Start-Time Delay Relay and each LPCI Pump Start — Time Delay Relay is chosen to be long enough so that most of the starting transient of one pump is complete before starting another pump on the same 4.16 kV emergency bus and short enough so that ECCS operation is not degraded.

Each CS Pump Start - Time Delay Relay and each LPCI Pump Start - Time Delay Relay Function is required to be OPERABLE only when the associated CS or LPCI System is required to be OPERABLE. Refer to LCO 3.5.1 for Applicability Bases for the CS and LPCI Systems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.f 2.k 4.16 kV Emergency Bus Sequential Loading Relay

An undervoltage condition on an emergency bus indicates that sufficient power (from either the offsite or onsite sources) is not available to allow starting of the low pressure ECCS pumps. Therefore, if this condition exists, the start permissive signal is withheld from the circuits that start both the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) pumps during accident conditions, and the CS and LPCI pumps powered from the respective emergency bus are prevented from starting. This ensures that the low pressure ECCS pumps are not started during accident conditions unless adequate power is available.

Each emergency bus is monitored by a single relay, which inputs into a one-out-of-two once logic for each division. Each logic channel supports one CS and two LPCI pumps. An instrument channel consists of the common bus monitoring relay and the associated relay contacts for each ECCS pump.

The 4.16 kV Emergency Bus Sequential Loading Relay Allowable Values are low enough to prevent low pressure ECCS pump starting unless adequate power is available, but high enough so that low pressure ECCS pump starting is not unnecessarily prohibited or delayed and is within the maximum adjustable range of the relay.

To ensure that no single failure can prevent successful operation of the combined low pressure ECCS, two channels of the 4.16 kV Emergency Bus Sequential Loading Relay Function are required to be OPERABLE whenever the LPCI System is required to be OPERABLE (i.e., MODES 1, 2, and 3). Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the CS and LPCI Systems.

(continued)

BASES

ACTIONS

B.1, B.2 and B.3 (continued)

For Required Action B.2, automatic initiation capability is lost if certain combinations of Function 3.a or Function 3.b channels are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the feature(s) associated with the inoperable, untripped channels must be declared inoperable within 1 hour.

Notes are provided (the Note to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed. Required Action B.1 (the Required Action for certain inoperable channels in the low pressure ECCS subsystems) is not applicable to Function 2.d, since this Function provides backup to administrative controls ensuring that operators do not divert LPCI flow from injecting into the core when needed. Thus, a total loss of Function 2.d capability for 24 hours is allowed, since the LPCI System remains capable of performing its intended function.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." For Required Action B.1, the Completion Time only begins upon discovery that a redundant feature in two or more low pressure ECCS subsystems (i.e., both CS subsystems or either CS subsystem in combination with the LPCI subsystem) cannot automatically initiate their supported features due to inoperable, untripped channels within the same Function as described in the paragraph above.

(continued)

BASES

ACTIONS

C.1, C.2 and C.3 (continued)

(d) two or more Function 2.i channels for the same recirculation pump are inoperable in both trip systems, (e) two Function 2.j channels are inoperable in both trip systems, or (f) two or more Function 1.e or 2.e channels are inoperable in different divisions. In these situations (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.3 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action C.1 or C.2, as appropriate, applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for either both divisions of LPCI loop select logic or for two or more low pressure ECCS subsystems are inoperable (e.g., both CS subsystems or either CS subsystem in combination with the LPCI subsystem), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For Functions 1.c, 1.e, and 2.e, the affected portions are the associated low pressure ECCS pumps. For Functions 2.g, 2.h, 2.i, and 2.j, the affected portion is the LPCI subsystem.

The Note to Required Action C.1 states that Required Action C.1 is only applicable for Functions 1.c, 2.c, 1.e, and 2.e. The Note to Required Action C.2 states that Required Action C.2 is only applicable for Functions 2.g, 2.h, 2.i, and 2.j. Required Actions C.1 and C.2 are not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable), since the loss of one channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference 5 and considered acceptable for the 24 hours allowed by Required Action C.3.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the Core Spray and Low Pressure Coolant Injection Pump Discharge Flow — Low Bypass Functions result in redundant automatic initiation capability being lost for the feature(s). For Required Action E.1, the features would be those that are initiated by Functions 1.d and 2.f (e.g., low pressure ECCS). Redundant automatic initiation capability is lost if either the automatic opening or closing function for two or more low pressure ECCS minimum flow valves is inoperable. Since each of the four minimum flow valves is initiated by a corresponding instrument channel, redundant automatic initiation capability is lost if any two of the four Function 1.d and 2.f channels are inoperable. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected low pressure ECCS pump(s) to be declared inoperable. However, since channels for two or more minimum flow valves in the low pressure ECCS subsystems are inoperable, and the Completion Times started concurrently for the channels of the low pressure ECCS minimum flow valves, this results in the affected low pressure ECCS pump(s) being concurrently declared inoperable.

In this situation (loss of redundant automatic initiation capability), the 7 day allowance of Required Action E.2 is not appropriate and the system or subsystem associated with each inoperable channel must be declared inoperable within 1 hour. A Note is also provided (the Note to Required Action E.1) to delineate that Required Action E.1 is only applicable to low pressure ECCS Functions. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 5 and considered acceptable for the 7 days allowed by Required Action E.2.

(continued)

BASES

ACTIONS (continued)

F.1

With a Function 1.f (2.k) channel inoperable, the 4.16 kV Emergency Bus Sequential Loading Relay is not cable of providing a start permissive signal for the low pressure ECCS pumps in the affected division, and the associated low pressure ECCS are not capable of performing their intended functions. Placing a channel in the tripped condition makes the associated ECCS pump(s) inoperable, since the pump(s) is prevented from automatically starting. In fact, tripping the bus power monitor relay will cause the associated DG to start, as it is common to the bus undervoltage logic. Consequently, one hour is provided to restore OPERABILITY of the channel. Otherwise, the affected low pressure ECCS subystems are declared inoperable immediately. This requires entry into LCO 3.5.1, which provides appropriate actions for inoperable low pressure ECCS Systems and subsystems.

G.1 and G.2

Required Action G.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within similar ADS trip logic A and B Functions result in redundant automatic initiation capability being lost for the ADS. Redundant automatic initiation capability is lost if either (a) one Function 4.a channel and one Function 5.a channel are inoperable and untripped, or (b) one Function 4.c channel and one Function 5.c channel are inoperable and untripped.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action G.2 is not appropriate and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock."

(continued)

B 3.3 INSTRUMENTATION

B 3.3.5.2 Reactor Pressure Vessel (RPV) Water Inventory Control Instrumentation

BASES

BACKGROUND

The RPV contains penetrations below the TS 2.1.1.3 Safety Limit that have the potential to drain the reactor coolant inventory to below the TS 2.1.1.3 Safety Limit. If the water level should drop below the TS 2.1.1.3 Safety Limit, the ability to remove decay heat is reduced, which could lead to elevated cladding temperatures and clad perforation. Safety Limit 2.1.1.3 requires the RPV water level to be above the top of the active irradiated fuel at all times to prevent such elevated cladding temperatures.

Technical Specifications are required by 10 CFR 50.36 to include limiting safety system settings (LSSS) for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur. The actual settings for the automatic isolation channels are the same as those established for the same functions in MODES 1, 2, and 3 in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," or LCO 3.3.6.1, "Primary Containment Isolation Instrumentation".

With the unit in MODE 4 or 5, RPV water inventory control is not required to mitigate any events or accidents evaluated in the safety analyses. RPV water inventory control is required in MODES 4 and 5 to protect Safety Limit 2.1.1.3 and the fuel cladding barrier to prevent the release of radioactive material should a draining event occur. Under the definition of DRAIN TIME, some penetration flow paths may be excluded from the DRAIN TIME calculation if they will be isolated by valves that will close automatically without offsite power prior to the RPV water level being equal to the TS 2.1.1.3

(continued)

BASES

BACKGROUND (continued)	Safety Limit when actuated by RPV water level isolation instrumentation. The purpose of the RPV Water Inventory Control Instrumentation is to support the requirements of LCO 3.5.2, "Reactor Pressure Vessel (RPV) Water Inventory Control," and the definition of DRAIN TIME. There are functions that are required for manual initiation or operation of the ECCS injection/spray subsystem required to be OPERABLE by LCO 3.5.2 and other functions that support automatic isolation of Residual Heat Removal subsystem and Reactor Water Cleanup system penetration flow path(s) on low RPV water level. The RPV Water Inventory Control Instrumentation supports operation of core spray (CS) and low pressure coolant injection (LPCI). The equipment involved with each of these systems is described in the Bases for LCO 3.5.2.
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	With the unit in MODE 4 or 5, RPV water inventory control is not required to mitigate any events or accidents evaluated in the safety analyses. RPV water inventory control is required in MODES 4 and 5 to protect and Safety Limit 2.1.1.3 and the fuel cladding barrier to prevent the release of radioactive material should a draining event occur. A double-ended guillotine break of the Reactor Coolant System (RCS) is not postulated in MODES 4 and 5 due to the reduced RCS pressure, reduced piping stresses, and ductile piping systems. Instead, an event is postulated in which a single operator error or initiating event allows draining of the RPV water inventory through a single penetration flow path with the highest flow rate, or the sum of the drain rates through multiple penetration flow paths susceptible to a common mode failure (e.g., seismic event, loss of normal power, single human error). It is assumed, based on engineering judgment, that while in MODES 4 and 5, one low pressure ECCS injection/spray subsystem can be manually initiated to maintain adequate reactor vessel water level. As discussed in References 1, 2, 3, 4, and 5, operating experience has shown RPV water inventory to be significant to public health and safety. Therefore, RPV Water Inventory Control satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Permissive and interlock setpoints are generally considered as nominal values without regard to measurement accuracy.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Steam Dome Pressure - Low (Injection Permissive)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS injection/spray subsystem manual injection functions. This function ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. While it is assured during MODES 4 and 5 that the reactor steam dome pressure will be below the ECCS maximum design pressure, the Reactor Steam Dome Pressure - Low signals are assumed to be OPERABLE and capable of permitting initiation of the ECCS.

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure switches that sense the reactor dome pressure. The pressure switches are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

The Allowable Value is low enough to prevent overpressuring the equipment in the low pressure ECCS.

The four channels of Reactor Steam Dome Pressure – Low Function are required to be OPERABLE in MODES 4 and 5.

1.b, 2.b. Core Spray and Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)

The minimum flow instruments are provided to protect the associated low pressure ECCS pump from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valve is opened when low flow is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump.

One flow transmitter for each Core Spray subsystem and each LPCI subsystem is used to detect the associated subsystems' flow rates. The logic is arranged such that each transmitter causes its associated minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded. The LPCI minimum flow valves are time delayed such that the valves will not open for 10 seconds after the switches detect low flow. The time delay is provided to limit reactor vessel inventory loss during the startup of the Residual Heat Removal (RHR) shutdown cooling mode.

The Pump Discharge Flow - Low Allowable Values are high enough to ensure that the pump flow rate is sufficient to protect the pump, yet low enough to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core.

One channel of the Pump Discharge Flow - Low Function is required to be OPERABLE in MODES 4 and 5 when the associated Core Spray or LPCI pump is required to be OPERABLE by LCO 3.5.2 to ensure the subsystem is capable of injecting into the Reactor Pressure Vessel when manually initiated.

RHR System Isolation

3.a - Reactor Vessel Water Level – Low

The definition of Drain Time allows crediting the closing of penetration flow paths that are capable of being isolated by valves that will close automatically without offsite power prior to the RPV water level being equal to the TS 2.1.1.3 Safety Limit when actuated by RPV water level isolation instrumentation. The Reactor Vessel Water Level - Low Function associated with RHR System isolation may be credited for automatic isolation of penetration flow paths associated with the RHR System.

The Reactor Vessel Water Level — Low signals are initiated from four level indicating switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. While four channels (two channels per trip system) of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Reactor Vessel Water Level – Low Function are available, only two channels (all in the same trip system) are required to be OPERABLE.

The Reactor Vessel Water Level - Low Allowable Value was chosen to be the same as the Primary Containment Isolation Instrumentation Reactor Vessel Water Level - Low Allowable Value (LCO 3.3.6.1), since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level - Low Function is only required to be OPERABLE when automatic isolation of the associated penetration flow path is credited in calculating DRAIN TIME.

This Function isolates the Group 4 valves.

Reactor Water Cleanup (RWCU) System Isolation

4.a - Reactor Vessel Water level – Low Low
The definition of Drain Time allows crediting the closing of penetration flow paths that are capable of being isolated by valves that will close automatically without offsite power prior to the RPV water level being equal to the TS 2.1.1.3 Safety Limit when actuated by RPV water level isolation instrumentation. The Reactor Vessel Water Level - Low Low Function associated with RWCU System isolation may be credited for automatic isolation of penetration flow paths associated with the RWCU System.

Reactor Vessel Water Level - Low Low signals are initiated from four level indicating switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. While four channels (two channels per trip system) of the Reactor Vessel Water Level – Low Function are available, only two channels (all in the same trip system) are required to be OPERABLE.

The Reactor Vessel Water Level – Low Low Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Allowable Value (LCO 3.3.5.1), since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level – Low Low Function is only required to be OPERABLE when automatic isolation of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	associated penetration flow path is credited in calculating DRAIN TIME. This Function isolates Group 5 valves.
ACTIONS	A Note has been provided to modify the ACTIONS related to RPV Water Inventory Control instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPV Water Inventory Control instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable RPV Water Inventory Control instrumentation channel.

(continued)

BASES

ACTIONS (continued)

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.2-1. The applicable Condition referenced in the Table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

RHR System Isolation, Reactor Vessel Water Level - Low and Reactor Water Cleanup System, Reactor Vessel Water Level - Low Low functions are applicable when automatic isolation of the associated penetration flow path is credited in calculating Drain Time. If the instrumentation is inoperable, Required Action B.1 directs an immediate declaration that the associated penetration flow path(s) are incapable of automatic isolation. Required Action B.2 directs calculation of DRAIN TIME. The calculation cannot credit automatic isolation of the affected penetration flow paths.

C.1

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS injection/spray subsystem manual injection functions. If the permissive is inoperable, manual initiation of ECCS is prohibited. Therefore, the permissive must be placed in the trip condition within 1 hour. With the permissive in the trip condition, manual initiation may be performed.

The Completion Time of 1 hour is intended to allow the operator time to evaluate any discovered inoperabilities and to place the channel in trip.

D.1

If a Core Spray or Low Pressure Coolant Injection Pump Discharge Flow – Low bypass function is inoperable, there is a risk that the associated low pressure ECCS pump could overheat when the pump is operating and the associated injection valve is not fully open. In this condition, the operator can take manual control of the pump and the injection valve to ensure the pump does not overheat.

The 24 hour Completion Time was chosen to allow time for the operator to evaluate and repair any discovered inoperabilities.

(continued)

BASES

ACTIONS (continued)

D.1 (continued)

The Completion Time is appropriate given the ability to manually start the ECCS pumps and open the injection valves and to manually ensure the pump does not overheat.

E.1

With the Required Action and associated Completion Time of Condition C or D not met, the associated low pressure ECCS injection/spray subsystem may be incapable of performing the intended function, and must be declared inoperable immediately.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each RCIC System instrumentation Function are found in the SRs column of Table 3.3.5.2-1.

SR 3.3.5.2.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK guarantees that undetected outright channel failure is limited; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.2.2 (continued)

Any setpoint adjustment chall be consistent with the assumptions of the current plant specific methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

REFERENCES

1. Information Notice 84-81 "Inadvertent Reduction in Primary Coolant Inventory in Boiling Water Reactors During Shutdown and Startup," November 1984.

2. Information Notice 86-74, "Reduction of Reactor Coolant Inventory Because of Misalignment of RHR Valves," August 1986.

3. Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(F)," August 1992.

4. NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," May 1993.

5. Information Notice 94-52, "Inadvertent Containment Spray and Reactor Vessel Draindown at Millstone 1," July 1994.

B 3.3 INSTRUMENTATION

B 3.3.5.3 Reactor Core Isolation Cooling (RCIC) System Instrumentation

BASES

BACKGROUND

The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is unavailable, such that initiation of the low pressure Emergency Core Cooling Systems (ECCS) pumps does not occur. A more complete discussion of RCIC System operation is provided in the Bases of LCO 3.5.3, "RCIC System."

The RCIC System may be initiated by either automatic or manual means, although manual initiation requires manipulation of individual component control switches. Automatic initiation occurs for conditions of reactor vessel low low water level. The variable is monitored by four level switches that are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic arrangement. Once initiated, the RCIC logic seals in and can be reset by the operator only when the reactor vessel water level signals have cleared.

The RCIC test return valve is closed on a RCIC initiation signal to allow full system flow to the Reactor Pressure Vessel (RPV). However, this design feature is not assumed in any transient analyses. Transient analyses are performed assuming the RCIC System is in the standby readiness condition when the transient occurs.

The RCIC System also monitors the water levels in the Condensate Storage Tank (CST) since this is the preferred source of water for RCIC operation. Upon receipt of a RCIC initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless the pump suction from the suppression pool valves is open. If the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close (i.e., one-out-of-two once logic).

(continued)

BASES

BACKGROUND (continued)

To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The RCIC System provides makeup water to the reactor until the reactor vessel water level reaches the RPV high water level trip (two-out-of-two once logic), at which time the RCIC steam supply valve closes (the injection valve also closes due to the closure of the steam supply valve). The RCIC System restarts if vessel level again drops to the low low level initiation point.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The function of the RCIC System, to provide makeup coolant to the reactor, is used to respond to transient events. The RCIC System is not an ECCS System, although the RCIC System Operation is credited during Loss of Feedwater events. Based on its contribution to the reduction of overall plant risk, however, the system, and therefore its instrumentation meets Criterion 4 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the RCIC System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.3-1. Each Function must have a required number of OPERABLE channels with their setpoints within the specified Allowable Values. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RCIC System instrumentation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip Setpoints are those predetermined values of output at which an action should take place.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., on-off sensor, bi-stable trip circuit, or trip unit) changes state. Margin is provided to allow for process, calibration (i.e., M&TE) and some instrument uncertainties (e.g., drift). Allowable Values derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The individual Functions are required to be OPERABLE in MODE 1, and in MODES 2 and 3 with reactor steam dome pressure > 150 psig since this is when RCIC is required to be OPERABLE. (Refer to LCO 3.5.3 for Applicability Bases for the RCIC System.)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level — Low Low

Low RPV water level indicates that normal feedwater flow is insufficient to maintain reactor vessel water level and that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the RCIC System is initiated at Reactor Vessel Water Level - Low Low to assist in maintaining water level above the top of the active fuel.

Reactor Vessel Water Level — Low Low signals are initiated from four level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level — Low Low Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with the High Pressure Coolant Injection assumed to fail will be sufficient to maintain adequate core cooling. However, the prevention of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level — Low Low (continued)

initiation of low pressure ECCS at Reactor Vessel Water Level - Low Low Low is not assured.

Four channels of Reactor Vessel Water Level — Low Low Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

2. Reactor Vessel Water Level — High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level - High signal is used to close the RCIC steam supply valve to prevent overflow into the Main Steam Lines (MSLs). (The injection valve also closes due to the closure of the steam supply valve.)

Reactor Vessel Water Level — High signals for RCIC are initiated from two level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level — High Allowable Value is high enough to preclude isolating the injection valve of the RCIC during normal operation, yet low enough to trip the RCIC System prior to water overflowing into the MSLs.

Two channels of Reactor Vessel Water Level — High Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

3. Condensate Storage Tank Level — Low

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this preferred source. Normally, the suction valve between the RCIC pump and the CST is open and, upon receiving a RCIC initiation signal,

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Condensate Storage Tank Level — Low (continued)

water for RCIC injection would be taken from the CST. However, if the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes.

Two level switches are used to detect low water level in the CSTs. The Condensate Storage Tank Level — Low Function Allowable Value is set high enough to ensure adequate pump suction head while water is being taken from the CST.

Two channels of Condensate Storage Tank Level — Low Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC swap to suppression pool source. Refer to LCO 3.5.3 for RCIC Applicability Bases.

ACTIONS

A Note has been provided to modify the ACTIONS related to RCIC System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RCIC System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, the Note has been provided to allow separate Condition entry for each inoperable RCIC System instrumentation channel.

(continued)

BASES

ACTIONS (continued)

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.3-1. The applicable Condition referenced in the Table is Function dependent. Each time a channel is discovered to be inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic initiation capability for the RCIC System. In this case, automatic initiation capability is lost if certain combinations of two Function 1 channels are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock". For Required Action B.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically initiated due to the required combination of two inoperable, untripped Reactor Vessel Water Level — Low Low channels. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status.

(continued)

ACTIONS

B.1 and B.2 (continued)

If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Action taken.

C.1

A risk based analysis was performed and determined that an allowable out of service time of 24 hours (Ref. 1) is acceptable to permit restoration of any inoperable channel to OPERABLE status (Required Action C.1). A Required Action (similar to Required Action B.1) limiting the allowable out of service time, if a loss of automatic RCIC initiation capability exists, is not required. This Condition applies to the Reactor Vessel Water Level — High Function whose logic is arranged such that any inoperable channel will result in a loss of automatic RCIC shutdown on high level capability. As stated above, this loss of automatic RCIC shutdown capability was analyzed and determined to be acceptable.

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in automatic suction transfer capability being lost. The automatic suction transfer capability is lost if two Function 3 channels are inoperable and untripped. In this situation (loss of automatic suction transfer), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour from discovery of loss of RCIC suction transfer capability. As noted, Required Action D.1 is only applicable if the RCIC pump suction is not aligned to the suppression pool since, if aligned, the Function is already performed.

(continued)

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of suction transfer capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide suction transfer signals and the fact that the RCIC System is not assumed in any accident analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1, which performs the intended function of the channel (shifting the suction source to the suppression pool). Alternatively, Required Action D.2.2 allows the manual alignment of the RCIC suction to the suppression pool, which also performs the intended function. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the RCIC System piping remains filled with water. If it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the RCIC discharge piping), Condition E must be entered and its Required Action taken.

E.1

With any Required Action and associated Completion Time not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each RCIC System instrumentation Function are found in the SRs column of Table 3.3.5.3-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 2 and 3; and (b) for up to 6 hours for Function 1, provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 1) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RCIC will initiate when necessary. Because the Ref. 1 analysis made no assumptions regarding the elapsed time between testing of consecutive channels in the same logic, it is not necessary to remove jumpers/relay blocks or reconnect lifted leads used to prevent actuation of the trip logic during testing of logic channels with instruments in series solely for the purpose of administering the AOT clocks, provided that the AOT allowance is not exceeded on a per instrument channel basis.

SR 3.3.5.3.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a parameter on other similar channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.3.1 (continued)

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.3.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extenstions.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on the reliability analysis of Reference 1.

SR 3.3.5.3.3 and SR 3.3.5.3.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based upon the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.3.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.3 overlaps this Surveillance to provide complete testing of the safety function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at this Frequency.

REFERENCES

1. GENE-770-06-2, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6.a. Reactor Steam Dome Pressure – High (continued)

present), yet high enough to preclude spurious isolations of shutdown cooling during system startup and operation and to provide sufficient overlap with the low pressure isolations of the HPCI and RCIC turbines to allow the transition to shutdown cooling during plant shutdowns.

This Function isolates the Group 4 valves.

6.b. Reactor Vessel Water Level — Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to begin isolating the potential sources of a break. The Reactor Vessel Water Level — Low Function associated with RHR Shutdown Cooling System isolation is not directly assumed in safety analyses because a break of the RHR Shutdown Cooling System is bounded by breaks of the Recirculation Suction and MSL. The RHR Shutdown Cooling System isolation on Reactor Vessel Water Level-Low supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System when the system is in operation (i.e., the shutdown cooling suction valves are automatically isolated, and if both of the RHR shutdown cooling suction valves are not fully closed and reactor steam dome pressure is less than 135 psig (nominal), then the two inboard LPCI injection valves are also automatically isolated if a low reactor vessel water level signal is received).

Reactor Vessel Water Level — Low signals are initiated from four level indicating switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels (two channels per trip system) of the Reactor Vessel Water Level — Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6.b. Reactor Vessel Water Level — Low (continued)

The Reactor Vessel Water Level — Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level — Low Allowable Value (LCO 3.3.1.1), since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level — Low Function is only required to be OPERABLE in MODE 3, to prevent this potential flow path from lowering the reactor vessel level to the top of the fuel. In MODES 1 and 2, another isolation (i.e., Reactor Steam Dome Pressure — High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

This Function isolates the Group 4 valves.

6.c. Drywell Pressure - High

High drywell pressure indicates that the RHR Shutdown Cooling System piping downstream of the inboard isolation valve located in the drywell may have experienced a break. In order to prevent the level in the RPV from dropping below the top of active fuel if this were to occur, this Function will cause the RHR Shutdown Cooling System to isolate if the system is in use (i.e., the shutdown cooling suction valves are automatically isolated, and if both of the RHR shutdown cooling suction valves are not fully closed and reactor steam dome pressure is less than 135 psig (nominal), then the two inboard LPCI injection valves are also automatically isolated if a high drywell pressure signal is received). The Drywell Pressure - High Function associated with the RHR Shutdown Cooling System isolation is not directly assumed in safety analyses because a break of the RHR Shutdown Cooling system is bounded by breaks of the Recirculation System and MSL.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level — Low (continued)

release. The Reactor Vessel Water Level — Low Function is one of the Functions assumed to be OPERABLE and capable of providing isolation and initiation signals. The isolation and initiation systems on and Reactor Vessel Water Level — Low support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

Reactor Vessel Water Level — Low signals are initiated from level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level-Low Allowable Value was chosen to be the same as the Reactor Protection System Reactor Vessel Level-Low Allowable Value (LCO 3.3.1.1), since this provides an early indication that the capability to cool the fuel is being threatened.

The Reactor Vessel Water Level — Low Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required.

2. Drywell Pressure — High

High drywell pressure can indicate a break in the Reactor Coolant Pressure Boundary (RCPB). An isolation of the secondary containment and actuation of the SBGT System are initiated in order to minimize the potential of an offsite dose release.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Control Building Intake Area Radiation — High (continued)

The Control Building Intake Area Radiation — High Function is required to be OPERABLE in MODES 1, 2, and 3 and during CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., CORE ALTERATIONS), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

ACTIONS

A Note has been provided to modify the ACTIONS related to SFU System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable SFU System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable SFU System instrumentation channel.

A.1, and A.2

With a Control Building Intake Area Radiation-High Channel inoperable, the initiation capability of the associated SFU subsystem is lost. Therefore, the associated SFU subsystem(s) must be declared inoperable within 1 hour per Required Action A.1, or must be placed in the isolation mode of operation, (i.e., the SFU in operation and the Control Building Ventilation System isolated, within 1 hour per Required Action A.2. Placing the subsystem in the isolation mode ensures that control room personnel will be protected in the event of a Design Basis Accident. The method used to place the SFU subsystem(s) in operation must provide for automatically re-initiating the subsystem(s) upon restoration of power following a loss of power to the SFU subsystem(s).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the SRVs and SVs will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the SRV and SV lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the INSERVICE TESTING PROGRAM. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The SRV and SV setpoints are ± 3% for OPERABILITY; however the valves are reset to ± 1% during the Surveillance to allow for drift.

The Surveillance Frequency is in accordance with the INSERVICE TESTING PROGRAM requirements contained in the ASME Code. This Surveillance must be performed during shutdown conditions.

SR 3.4.3.2

The actuator of each dual function safety/relief valves (S/RVs) is stroked to verify that the pilot valve strokes when manually actuated. The actuator test is performed by energizing a solenoid that pneumatically actuates a plunger. The plunger is connected to the second stage disc located within the main valve body. When steam pressure actuates the plunger during plant operation, this allows pressure to be vented from the top of the main valve piston, allowing reactor pressure to lift the main valve piston, which opens the main valve disc. The test will verify movement of the plunger in accordance with vendor recommendations. However, since this test is performed prior to establishing the reactor pressure needed to overcome main valve closure forces, the main valve disc will not stroke during the test.

This SR, together with the valve testing performed as required by the ASME Code for pressure relieving devices (ASME OM Code - 2001 through 2003 Addenda), verify the capability of each relief valve to perform its function.

Valve testing will be performed at a steam test facility, where the valve (i.e., main valve and pilot valve) and an actuator representative of the actuator used at the plant will be installed on a steam header in the same orientation as the plant installation. The test conditions in the test facility will be similar to those in the plant installation, including ambient temperature, valve insulation, and steam conditions. The valve will then be leak tested, functionally tested to ensure the valve is capable of opening and

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

closing (including stroke time), and leak tested a final time. Valve seat tightness will be verified by a cold bar test, and if not free of fog, leakage will be measured and verified to be below design limits. In addition, for the safety mode of S/RVs, an as-found setpoint verification and as-found leak check are performed, followed by verification of set pressure, and delay time. The valve will then be shipped to the plant without any disassembly or alteration of the main valve or pilot valve components.

The combination of the valve testing and the valve actuator testing provide a complete check of the capability of the valves to open and close, such that full functionality is demonstrated through overlapping tests, without cycling the valves.

If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the SRV is not considered inoperable.

This SR is not applicable to the SVs, due to their design which does not include the manual relief capability, nor do they have a discharge line that can become blocked.

The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

REFERENCES

1. UFSAR, Section 5.2.2.2.1.
2. UFSAR, Section 15.1.2.
3. ASME Code for Operation and Maintenance of Nuclear Power Plants.
4. NUREG 1482, Guidelines for Inservice Testing at Nuclear Power Plants.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.1 ECCS — Operating

BASES

BACKGROUND

The ECCS is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a Loss of Coolant Accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network consists of the High Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the Condensate Storage Tank (CST), it is capable of providing a source of water for the HPCI and CS systems.

On receipt of an initiation signal, after the appropriate time delays for the Diesel Generators (DGs) to start and provide power to the 4160 VAC bus, assuming the concurrent Loss of Offsite Power (LOOP), ECCS pumps automatically start; the system aligns and the pumps inject water, taken either from the CST or suppression pool, into the Reactor Coolant System (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. Although the system is initiated, ADS action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCI pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the vessel to cool the core. If the break is small, the HPCI System will maintain coolant inventory as well as vessel level while the RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with LPCI and CS. In this event, the ADS timed sequence would be allowed to time out and open the selected Safety Relief Valves (SRVs) depressurizing the RCS, thus allowing the LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly precluding HPCI from injecting to the vessel, and the LPCI and CS subsystems cool the core.

(continued)

BASES

LCO

Each ECCS injection/spray subsystem and four ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the LPCI System, and one HPCI System. The low pressure ECCS subsystems are defined as the two CS subsystems and the LPCI System. Management of gas voids is important to ECCS injection/spray subsystem OPERABILITY.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

The LPCI System may be considered OPERABLE during alignment and operation for decay heat removal (i.e.- Shutdown Cooling) when below the actual RHR Shutdown Cooling interlock pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary. In addition, the risk of a LOCA during the transition from the RHR interlock pressure to cold shutdown is minimal.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, HPCI is not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. In MODES 2 and 3, when reactor steam dome pressure is ≤ 100 psig, ADS is not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "RPV Water Inventory Control."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to manual valves unless the valves are being manipulated to serve as all or part of a system vent flow path opened under administrative control, as described in the SR Note (and Bases paragraph below). In this case, the SR Note allows the licensee to credit administratively controlled manual action to close the system vent flow path in order to maintain system Operability during system venting and performance of the gas accumulation SR. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the HPCI System, this SR also includes the steam flow path for the turbine and the flow controller position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of this SR was derived from the INSERVICE TESTING PROGRAM requirements for performing valve testing. The Frequency is further justified because the valves are operated under procedural control and because improper valve position would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include a stationing of a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

In Mode 3 with reactor steam dome pressure less than the actual RHR interlock pressure, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by Note 1, which allows the LPCI System to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Manual alignment of the RHR cross tie valves (MO2010 and V-19-48) may only be credited when the valves are soft seated per normal operating procedure in order to prevent inoperability due to thermal binding. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. At the low

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

pressures and decay heat loads associated with operation in Mode 3 with reactor steam dome pressure less than the RHR interlock pressure, a reduced complement of low pressure ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling, when necessary.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification that a 30 day supply of nitrogen exists for each ADS accumulator ensures adequate nitrogen pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that following a failure of the pneumatic supply to the accumulator, each ADS valve can be actuated at least 5 times up to 30 days following a LOCA (Reference 4). This SR can be met by either: 1) verifying that the drywell nitrogen header supply pressure is ≥ 90 psig, or 2) when drywell nitrogen header supply pressure is < 90 psig, using the actual accumulator check valve leakage rates obtained from the most-recent tests to determine, analytically, that a 30 day supply of nitrogen exists for each accumulator. The results of this analysis can also be used to determine when the 30 day supply of nitrogen will no longer exist for individual ADS accumulators, and when each ADS valve would subsequently be required to be declared inoperable, assuming the drywell nitrogen supply pressure is not restored to ≥ 90 psig. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency takes into consideration administrative controls over operation of the nitrogen system and alarms for low nitrogen pressure.

SR 3.5.1.4, SR 3.5.1.5, and SR 3.5.1.6

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established during preoperational testing or by analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4, SR 3.5.1.5, and SR 3.5.1.6 (continued)

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 940 psig to perform SR 3.5.1.5, the high pressure test, and ≤ 160 psig to perform SR 3.5.1.6, the low pressure test. Adequate steam flow is represented by approximately 0.5 turbine bypass valves open. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable.

Therefore, SR 3.5.1.5 and SR 3.5.1.6 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hour allowance to reach the required pressure and flow is sufficient to achieve stable conditions for testing and provide a reasonable time to complete the SRs.

The Frequency for SR 3.5.1.4 and SR 3.5.1.5 is in accordance with the INSERVICE TESTING PROGRAM requirements. The Surveillance Frequency for SR 3.5.1.6 is controlled under the Surveillance Frequency Control Program. The Frequency for SR 3.5.1.6 is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at this Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.9 (continued)

The plunger is connected to the second stage disc located within the main valve body. When steam pressure actuates the plunger during plant operation, this allows pressure to be vented from the top of the main valve piston, allowing reactor pressure to lift the main valve piston, which opens the main valve disc. The test will verify movement of the plunger in accordance with vendor recommendations. However, since this test is performed prior to establishing the reactor pressure needed to overcome main valve closure forces, the main valve disc will not stroke during the test.

This SR, together with the valve testing performed as required by the ASME Code for pressure relieving devices (ASME OM Code - 2001 through 2003 Addenda), verify the capability of each relief valve to perform its function.

Valve testing will be performed at a steam test facility, where the valve (i.e., main valve and pilot valve) and an actuator representative of the actuator used at the plant will be installed on a steam header in the same orientation as the plant installation. The test conditions in the test facility will be similar to those in the plant installation, including ambient temperature, valve insulation, and steam conditions. The valve will then be leak tested, functionally tested to ensure the valve is capable of opening and closing (including stroke time), and leak tested a final time. Valve seat tightness will be verified by a cold bar test, and if not free of fog, leakage will be measured and verified to be below design limits. In addition, for the safety mode of S/RVs, an as-found setpoint verification and as-found leak check are performed, followed by verification of set pressure, and delay time. The valve will then be shipped to the plant without any disassembly or alteration of the main valve or pilot valve components.

The combination of the valve testing and the valve actuator testing provide a complete check of the capability of the valves to open and close, such that full functionality is demonstrated through overlapping tests, without cycling the valves.

SR 3.5.1.8 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 Reactor Pressure Vessel (RPV) Water Inventory Control

BASES

BACKGROUND

The RPV contains penetrations below the TS 2.1.1.3 Safety Limit that have the potential to drain the reactor coolant inventory below the TS 2.1.1.3 Safety Limit. If the water level should drop below the TS 2.1.1.3 Safety Limit, the ability to remove decay heat is reduced, which could lead to elevated cladding temperatures and clad perforation. Safety Limit 2.1.1.3 requires the RPV water level to be above the top of the active irradiated fuel at all times to prevent such elevated cladding temperatures.

APPLICABLE SAFETY ANALYSES

With the unit in MODE 4 or 5, RPV water inventory control is not required to mitigate any events or accidents evaluated in the safety analyses. RPV water inventory control is required in MODES 4 and 5 to protect Safety Limit 2.1.1.3 and the fuel cladding barrier to prevent the release of radioactive material to the environment should an unexpected draining event occur.

A double-ended guillotine break of the Reactor Coolant System (RCS) is not postulation in MODES 4 and 5 due to the reduced RCS pressure, reduced piping stresses, and ductile piping systems. Instead, an event is considered in which single operator error or initiating event allows draining of the RPV water inventory through a single penetration flow path with the highest flow rate, or the sum of the drain rates through multiple penetration flow paths susceptible to a common mode failure (e.g., seismic event, loss of normal power, single human error). It is assumed, based on engineering judgement, that while in MODES 4 and 5, one low pressure ECCS injection/spray subsystem can maintain adequate reactor vessel water level.

As discussed in References 1, 2, 3, 4, and 5, operating experience has shown RPV water inventory to be significant to public health and safety. Therefore, RPV Water Inventory Control satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

LCO

The RPV water level must be controlled in Modes 4 and 5 to ensure that if an unexpected draining event should occur, the reactor coolant water level remains above the top of the active irradiated fuel as required by SAFETY Limit 2.1.1.3.

The Limiting Condition for Operation (LCO) requires the DRAIN TIME of RPV water inventory to the TS 2.1.1.3 Safety Limit to be ≥ 36 hours. A DRAIN TIME of 36 hours is considered reasonable to identify and initiate action to mitigate unexpected draining of reactor coolant. An event that could cause loss of RPV water inventory and result in the RPV water level reaching the TS 2.1.1.3 Safety Limit in greater than 36 hours does not represent a significant challenge to Safety Limit 2.1.1.3 and can be managed as part of normal plant operation.

One low pressure ECCS injection/spray subsystem is required to be OPERABLE and capable of being manually started to provide defense-in-depth should an unexpected draining event occur. A low pressure ECCS injection/spray subsystem consists of either one Core Spray (CS) subsystem or one Low Pressure Coolant Injection (LPCI) subsystem. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or condensate storage tank (CST) to the RPV. Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. In MODES 4 and 5, the RHR System cross tie valve is not required to be closed.

The LCO is modified by a Note which allows a required LPCI subsystem to be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Because of the restrictions on DRAIN TIME, sufficient time will be available following an unexpected draining event to manually align and initiate LPCI subsystem operation to maintain RPV water inventory prior to the RPV water level reaching the TS 2.1.1.3 Safety Limit.

(continued)

BASES

APPLICABILITY

RPV water inventory control is required in MODES 4 and 5. Requirements on water inventory control in other MODES are contained in LCOs in Section 3.3, Instrumentation, and other LCOs in Section 3.5, ECCS, RCIC, and RPV Water Inventory Control. RPV water inventory control is required to protect Safety Limit 2.1.1.3 which is applicable whenever irradiated fuel is in the reactor vessel.

ACTIONS

A.1 and B.1

If the required low pressure ECCS injection/spray subsystem is inoperable, it must be restored to OPERABLE status within 4 hours. In this Condition, the LCO controls on DRAIN TIME minimize the possibility that an unexpected draining event could necessitate the use of the ECCS injection/spray subsystem, however the defense-in-depth provided by the ECCS injection/spray subsystem is lost. The 4 hour Completion Time for restoring the required low pressure ECCS injection/spray subsystem to OPERABLE status is based on engineering judgement that considers the LCO controls on DRAIN TIME and the low probability of an unexpected draining event that would result in loss of RPV water inventory.

If the inoperable ECCS injection/spray subsystem is not restored to OPERABLE status within the required Completion Time, action must be initiated immediately to establish a method of water injection capable of operating without offsite electrical power. The method of water injection includes the necessary instrumentation and controls, water sources, and pumps and valves needed to add water to the RPV or refueling cavity should an unexpected draining event occur. The method of water injection may be manually initiated and may consist of one or more systems or subsystems, and must be able to access water inventory capable of maintaining the RPV water level above the TS 2.1.1.3 Safety Limit for ≥ 36 hours. If recirculation of injected water would occur, it may be credited in determining the necessary water volume.

C.1, C.2, and C.3

With the DRAIN TIME less than 36 hours but greater than or equal to 8 hours, compensatory measures should be taken to ensure the ability to implement mitigating actions should an unexpected draining event occur. Should a draining event lower the reactor coolant level below the TS 2.1.1.3 Safety Limit,

(continued)

ACTIONS

C.1, C.2, and C.3 (continued)

there is potential for damage to the reactor fuel cladding and release of radioactive material. Additional actions are taken to ensure that radioactive material will be contained, diluted, and processed prior to being released to the environment.

The secondary containment provides a controlled volume in which fission products can be contained, diluted, and processed prior to release to the environment. Required Action C.1 requires verification of the capability to establish the secondary containment boundary in less than the DRAIN TIME. The required verification confirms actions to establish the secondary containment boundary are preplanned and necessary materials are available. The secondary containment boundary is considered established when one Standby Gas Treatment (SBGT) subsystem is capable of maintaining a negative pressure in the secondary containment with respect to the environment.

Verification that the secondary containment boundary can be established must be performed within 4 hours. The required verification is an administrative activity and does not require manipulation or testing of equipment. Secondary containment penetration flow paths form a part of the secondary containment boundary. Required Action C.2 requires verification of the capability to isolate each secondary containment penetration flow path in less than the DRAIN TIME. The required verification confirms actions to isolate the secondary containment penetration flow paths are preplanned and necessary materials are available. Power operated valves are not required to receive automatic isolation signals if they can be closed manually within the required time. Verification that the secondary containment penetration flow paths can be isolated must be performed within 4 hours. The required verification is an administrative activity and does not require manipulation or testing of equipment.

One SBGT subsystem is capable of maintaining the secondary containment at a negative pressure with respect to the environment and filter gaseous releases. Required Action C.3 requires verification of the capability to place one SBGT subsystem in operation in less that the DRAIN TIME. The required verification confirms actions to place SBGT subsystem in operation are preplanned and necessary materials are available. Verification that a SBGT subsystem can be placed in operation must be performed within 4 hours. The required verification is an administrative activity and does not require manipulation or testing of equipment.

(continued)

BASES

ACTIONS (continued)

D.1, D.2, D.3, and D.4

With the DRAIN TIME less than 8 hours, mitigating actions are implemented in case an unexpected draining event should occur. Note that if the DRAIN TIME is less than 1 hour, Required Action E.1 is also applicable.

Required Action D.1 requires immediate action to establish an additional method of water injection augmenting the ECCS injection/spray subsystem required by the LCO. The additional method of water injection includes the necessary instrumentation and controls, water sources, and pumps and valves needed to add water to the RPV or refueling cavity should an unexpected draining event occur. The Note to Required Action D.1 states that either the ECCS injection/spray subsystem or the additional method of water injection must be capable of operating without offsite electrical power. The additional method of water injection may be manually initiated and may consist of one or more systems or subsystems. The additional method of water injection must be able to access water inventory capable of being injected to maintain the RPV water level above the TS 2.1.1.3 Safety Limit for ≥ 36 hours. The additional method of water injection and the ECCS injection/spray subsystem may share all or part of the same water sources. If recirculation of injected water would occur, it may be credited in determining the required water volume.

Should a draining event lower the reactor coolant level to below the TS 2.1.1.3 Safety Limit, there is potential for damage to the reactor fuel cladding and release of radioactive material. Additional actions are taken to ensure that radioactive material will be contained, diluted, and processed prior to being released to the environment.

The secondary containment provides a control volume in which fission products can be contained, diluted, and processed prior to release to the environment. Required Action D.2 requires that actions be immediately initiated to establish the secondary containment boundary. With the secondary containment boundary established, one SBGT subsystem is capable of maintaining a negative pressure in the secondary containment with respect to the environment.

The secondary containment penetrations form a part of the

(continued)

BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

secondary containment boundary. Required Action D.3 requires that actions be immediately initiated to verify that each secondary containment penetration flow path is isolated or to verify that it can be manually isolated from the control room.

One SBGT subsystem is capable of maintaining the secondary containment at a negative pressure with respect to the environment and filter gaseous releases. Required Action D.4 requires that actions be immediately initiated to verify that at least on SBGT subsystem is capable of being placed in operation. The required verification is an administrative activity and does not require manipulation of testing of equipment.

E.1

If the Required Actions and associated Completion times of Conditions C or D are not met or if the DRAIN TIME is less than 1 hour, actions must be initiated immediately to restore the DRAIN TIME to ≥ 36 hours. In this condition, there may be insufficient time to respond to an unexpected draining event to prevent the RPV water inventory from reaching the TS 2.1.1.3 Safety Limit. Note that Required Actions D.1, D.2, D.3, and D.4 are also applicable when DRAIN TIME is less than 1 hour.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

This Surveillance verifies that the DRAIN TIME of RPV water inventory to the TS 2.1.1.3 Safety Limit is ≥ 36 hours. The period of 36 hours is considered reasonable to identify and initiate action to mitigate draining of reactor coolant. Loss of RPV water inventory that would result in the RPV water level reaching the TAF in greater than 36 hours does not represent a significant challenge to Safety Limit 2.1.1.3 and can be managed as part of normal plant operation.

The definition of DRAIN TIME states that realistic cross-sectional areas and drain rates are used in the calculation. A realistic drain rate may be determined using a single, step-wise, or integrated calculation considering the changing RPV water level during a draining event. For a Control Rod RPV penetration flow path with the Control Rod Drive Mechanism removed and not replaced with

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

a blank flange, the realistic cross-sectional area is based on the control rod blade seated in the control rod guide tube. If the control rod blade will be raised from the penetration to adjust or verify seating of the blade, the exposed cross-sectional area of the RPV penetration flow path is used..

The definition of DRAIN TIME excludes from the calculation those penetration flow paths connected to an intact closed system, or isolated by manual or automatic valves that are locked, sealed, or otherwise secured in the closed position, blank flanges, or other devices that prevent flow of reactor coolant through the penetration flow paths. A blank flange or other bolted device must be connected with a sufficient number of bolts to prevent draining in the event of an Operating Basis Earthquake. Normal or expected leakage from closed systems or past isolation devices is permitted. Determination that a system is intact and closed or isolated must consider the status of branch lines and ongoing plant maintenance and testing activities.

The Residual Heat Removal (RHR) Shutdown Cooling System is only considered an intact closed system when misalignment issues (Reference 6) have been precluded by functional valve interlocks or by isolation devices, such that redirection of RPV water out of an RHR subsystem is precluded. Further, RHR Shutdown Cooling System is only considered an intact closed system if its controls have not been transferred to Remote Shutdown, which disables the interlocks and isolation signals.

The exclusion of penetration flow paths from the determination of DRAIN TIME must consider the potential effects of a single operator error or initiating event on items supporting maintenance and testing (rigging, scaffolding, temporary shielding, piping plugs, snubber removal, freeze seals, etc.). If failure of such items could result and would cause a draining event from a closed system or between the RPV and the isolation device, the penetration flow path may not be excluded from the DRAIN TIME calculation.

Surveillance Requirement 3.0.1 requires SRs to be met between performances. Therefore, any changes in plant conditions that would change the DRAIN TIME requires that a new DRAIN TIME be determined.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.2 and SR 3.5.2.3

The minimum water level of 7.0 ft required for the suppression pool is periodically verified to ensure that the suppression will provide adequate Net Positive Section Head (NPSH) for the LPCI subsystem pump, recirculation volume, and vortex prevention. With the suppression pool water level less that the required limit, the required LPCI subsystem is inoperable.

When suppression pool level is < 8.0 ft, the required CS subsystem is considered OPERABLE only if it can take suction from the CST, and the CST water level is sufficient to provide the required NPSH for the CS pump. Therefore, a verification that either the suppression pool water level is ≥ 8.0 ft or that a required CS subsystem is aligned to take suction from the CSTs and the CSTs contain ≥ 75,000 gallons of water, equivalent to 11 ft in one CST or ≥ 7 ft in both CSTs, ensures that the required CS subsystem can supply at least 75,000 gallons of makeup water to the RPV.

This Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.4

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the required ECCS injection/spray subsystems full of water ensures that the ECCS subsystem will perform properly. This may also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring that the lines are full is to vent at the high points.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5

Verifying the correct alignment for power operated and automatic valves in the required ECCS subsystem flow path provides assurance that the proper flow paths will be available for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to manual valves unless the valves are being manipulated to serve as all or part of a system vent flow path opened under administrative control, as described in the SR Note (and Bases paragraph below). In this case, the SR Note allows the licensee to credit administratively controlled manual action to close the system vent flow path in order to maintain system Operability during system venting and performance of the gas accumulation SR. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is appropriate because the valves are operated under procedural control and the probability of their being mispositioned during this time period is low.

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include a stationing of a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

In Modes 4 and 5, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by a Note that allows a required LPCI subsystem to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. Because of the restrictions on

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.5 (continued)

DRAIN TIME, sufficient time will be available following an unexpected draining event to manually align and initiate LPCI subsystem operation to maintain RPV water inventory prior to the RPV water level reaching the TS 2.1.1.3 Safety Limit.

SR 3.5.2.6

Verifying that the required ECCS injection/spray subsystem can be manually started and operate for at least 10 minutes demonstrates that the subsystem is available to mitigate a draining event. Testing the ECCS injection/spray subsystem through the recirculation line is necessary to avoid overfilling the refueling cavity. The minimum operating time of 10 minutes was based on engineering judgement. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.7

Verifying that each valve credited for automatically isolating a penetration flow path actuates to the isolation position on an actual or simulated RPV water level isolation signal is required to prevent RPV water inventory from dropping below the TS 2.1.1.3 Safety Limit should an unexpected draining event occur. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.8

The required ECCS subsystem must be capable of being manually operated. This Surveillance verifies that the required ECCS injection/spray subsystem, including the associated pump and valves, can be manually operated.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

(continued)

BASES

REFERENCES

1. Information Notice 84-81, "Inadvertent Reduction in Primary Coolant Inventory in Boiling Water Reactors During Shutdown and Startup," November 1984.

2. Information Notice 86-74, "Reduction of Reactor Coolant Inventory Because of Misalignment of RHR Valves," August 1986.

3. Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," August 1992.

4. NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," May 1993.

5. Information Notice 94-52, "Inadvertent Containment Spray and Reactor Vessel Draindown at Millstone 1," July 1994.

6. General Electric Service Information Letter No. 388, "RHR Valve Misalignment During Shutdown Cooling Operation for BWR 3/4/5/6," February 1983.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following Reactor Pressure Vessel (RPV) isolation accompanied by a loss of coolant flow from the Feedwater System to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC Systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the Feedwater System line, where the coolant is distributed within the RPV through the Feedwater sparger. Suction piping is provided from the Condensate Storage Tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard Main Steamline Isolation Valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures 150 psig to 1120 psig. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

BACKGROUND (continued)

The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge piping is kept full of water. The RCIC System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for RCIC is such that the water in the feedwater lines keeps the remaining portion of the RCIC discharge line full of water. Therefore, RCIC does not require a "keep fill" system when its suction is aligned to the CST. When RCIC suction is aligned to the suppression pool and the system is not in operation, an alternate means of keeping the discharge piping full is required to support system OPERABILITY.

APPLICABLE SAFETY ANALYSES

The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. The RCIC System is not an ECCS System, although the RCIC System Operation is credited during Loss of Feedwater events. Based on its contribution to the reduction of overall plant risk, however, the system satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of the RCIC System provides adequate core cooling in the event of RPV isolation accompanied by a loss of Feedwater flow. The RCIC System has sufficient capacity for maintaining RPV inventory during an isolation event. Management of gas voids is important to RCIC System OPERABILITY.

APPLICABILITY

The RCIC System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, the low pressure ECCS injection/spray subsystems can provide sufficient flow to the RPV. In MODES 4 and 5, RCIC is not required to be OPERABLE since RPV water inventory control is required by LCO 3.5.2, “RPV Water Level Inventory Control.”

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1 (continued)

may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RCIC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, plant configuration, or personnel safety. For these locations, alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

SR 3.5.3.2

Verifying the correct alignment for power operated and automatic valves in the RCIC flow path provides assurance that the proper flow path will exist for RCIC operation. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to manual valves unless the valves are being manipulated to serve as all or part of a system vent flow path opened under administrative control, as described in the SR Note (and Bases paragraph below). In this case, the SR Note allows the licensee to credit administratively controlled manual action to close the system vent flow path in order to maintain system Operability during system venting and performance of the gas accumulation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

SR. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of this SR was derived from the INSERVICE TESTING PROGRAM requirements for performing valve testing. The Frequency is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include a stationing of a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≥ 940 psig to perform SR 3.5.3.3, the high pressure test, and ≤ 160 psig to perform SR 3.5.3.4, the low pressure test. Adequate steam flow is represented by approximately 0.4 turbine bypass valves open. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable.

Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hour allowance to reach the required pressure and flow is sufficient to achieve stable conditions for testing and provide a reasonable time to complete the SRs.

The Frequency for SR 3.5.3.3 is in accordance with the INSERVICE TESTING PROGRAM. The Surveillance Frequency for SR3.5.3.4 is controlled under the Surveillance Frequency Control Program. The Frequency for SR 3.5.3.4 is based on the need to perform the Surveillance under conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at this Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.5

The RCIC System is required to actuate automatically in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the RCIC System will cause the system to operate as designed, including actuation of the system throughout its emergency operating sequence; that is, automatic pump startup and actuation of all automatic valves to their required positions. This test also ensures the RCIC System will automatically restart on an RPV low water level signal received subsequent to an RPV high water level trip and that the suction is automatically transferred from the CST to the suppression pool on a CST Low Level signal. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed design function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at this Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance

(continued)

BASES

LCO (continued)	provided in the INSERVICE TESTING PROGRAM. Purge valves with resilient seals, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing. This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the primary containment boundary during accidents.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE in MODES 4 and 5. Shutdown Cooling System Isolation valves, however, are required to be OPERABLE in Modes 4 and 5 to prevent inadvertent reactor vessel draindown. These valves are only required to be OPERABLE for those functions required OPERABLE by the associated instrumentation per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." Per Table 3.3.6.1, if RHR Shutdown Cooling System integrity is maintained (i.e., no OPDRVs in progress within the RHR Shutdown Cooling System boundary) only one trip system and its associated PCIVs are required to be OPERABLE in Modes 4 and 5. Specifically, either an inboard trip system and its associated PCIVs (MO-1908 and MO-2003) or the outboard trip system and its associated PCIVs (MO-1909 and MO-1905) must be OPERABLE. (This does not include the valves that isolate the associated instrumentation.)
ACTIONS	The ACTIONS are modified by a Note allowing penetration flow path(s) to be unisolated intermittently under administrative controls. For valves requiring local operation, these controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated. For valves that can be operated remotely from the control room, the valve hand

(continued)

ACTIONS (continued)

F.1 and F.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR ensures that the primary containment purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable.

If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is modified by a Note stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for

(continued)

BASES

SUREVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 (continued)

inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surevillances that require the valves to be open. The 18 inch purge valves are capable of closing in the enviornment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency was chosen to provide added assurance that the purge valves are in the correct position.

SR 3.6.1.3.2

The traversing incore probe (TIP) shear isolation valves are actuated by explosive charges. Surveillance of explosive charge continuity provides assurance that TIP valves will actuate when required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.6.1.3.3

Verifying the isolation time of each power operated automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.5. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the requirements of the INSERVICE TESTING PROGRAM.

SR 3.6.1.3.4

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 3), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.4 (continued)

corresponding Surveillance Frequency. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The purge system isolation valves are tested in three groups, by penetration: drywell purge exhaust group (CV-4302 and CV-4303), torus purge exhaust group (CV-4300 and CV-4301), and drywell/torus purge supply group (CV-4307, CV-4308 and CV-4306). If the results of a combined leak rate or pressure drop test indicate excessive leakage, credit can be taken for one of the purge valves to satisfy Required Action E.1, if it can be reasonably determined that the purge valve to be credited for isolation is not leaking excessively.

Additionally, this SR must be performed once within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that which occurs to a valve that has not been opened.) Thus, decreasing the normal Surveillance interval is a prudent measure after a valve has been opened.

SR 3.6.1.3.5

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR 50.67 limits and that the core remains covered. The Frequency of this SR is in accordance with the requirements of the INSERVICE TESTING PROGRAM.

SR 3.6.1.3.6

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. A Note has been added for the MSIVs, that allows this SR to be met by any series of sequential, overlapping, or total steps so that proper operation of the MSIVs on receipt of an actual or simulated isolation signal is verified. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this Surveillance when performed at this Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.7

This SR requires a demonstration that a representative sample of reactor instrumentation line Excess Flow Check Valves (EFCVs) are OPERABLE by verifying that the valves cause a marked decrease in flow rate on a simulated instrument line break. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event evaluated in Reference 5. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). The nominal 10 year interval is based on other performance-based testing programs, such as inservice testing (snubbers) and Option B to 10 CFR 50, Appendix J. EFCV test failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 10).

SR 3.6.1.3.8

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. Other administrative controls, such as those that limit the shelf life of the explosive charges, must also be followed. The Frequency of this SR is in accordance with the requirements of the INSERVICE TESTING PROGRAM.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1 (continued)

limits. In addition, for the safety mode of S/RVs, an as-found setpoint verification and as-found leak check are performed, followed by verification of set pressure, and delay time. The valve will then be shipped to the plant without any disassembly or alteration of the main valve or pilot valve components.

The combination of the valve testing and the valve actuator testing provide a complete check of the capability of the valves to open and close, such that full functionality is demonstrated through overlapping tests, without cycling the valves.

The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.1.5.2

The LLS designated SRVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the LLS function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.3, "Low-Low Set (LLS) Instrumentation," overlaps this SR to provide complete testing of the safety function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at this Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

REFERENCES

1. UFSAR, Section 5.4.13.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
3. NEDE-30021-P, Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for DAEC, January 1983.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

breaker assembly valve be periodically verified to be closed by visual inspection. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker assembly valve status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor building-to-suppression chamber vacuum breaker assembly valves opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breaker assembly valves are controlled by plant procedures and do not represent inoperable vacuum breaker assembly valves. The second Note is included to clarify that vacuum breaker assembly valves open due to an actual differential pressure are not considered as failing this SR.

SR 3.6.1.6.2

Each vacuum breaker assembly valve must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of this SR was developed based upon INSERVICE TESTING PROGRAM requirements for performing valve testing.

SR 3.6.1.6.3

Demonstration of vacuum breaker assembly valve opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker assembly valve full open differential pressure of ≤ 0.614 psid is valid. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based upon the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 6.2.1.1.2.5.

BASES

ACTIONS

C.1 and C.2 (continued)

must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1

Each vacuum breaker is verified closed (except when performing its intended function as stated in LCO 3.6.1.7) to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by verifying that a differential pressure of 0.5 psid between the suppression chamber and drywell is maintained for 1 hour without makeup. Each vacuum breaker is equipped with two closed position indicators. One position indicator indicating closed is sufficient to verify the vacuum breaker is closed. However, if one closed position indicator is found to be inoperable, actions should be initiated to restore it to OPERABLE status, if possible. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

A Note is added to this SR which allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

SR 3.6.1.7.2

Each required vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This ensures that the safety analysis assumptions are valid. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of this SR was developed, based on INSERVICE TESTING PROGRAM requirements

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to SRV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 remains valid. Suppression Pool Water Level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).
LCO	A limit that suppression pool water level be ≥ 10.11 ft and ≤ 10.43 ft is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation. The level requirements also ensure that downcomer submergence is sufficient to ensure condensation effectiveness and prevent steam bypass to the suppression chamber air space and that loads and structural integrity are acceptable.
APPLICABILITY	In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirements for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "RPV Water Inventory Control."

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

Verifying by administrative means the correct alignment for manual, power operated and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to manual valves or to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each RHR pump develops a flow rate $\geq$ 4800 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that the primary containment peak pressure and temperature and the local suppression pool temperature can be maintained below design limits. This test also verifies that pump performance has not degraded during the surveillance interval. Flow is a normal test of centrifugal pump performance required by ASME Code (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice testing confirms component OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.2.3.3

RHR Suppression Pool Cooling System piping and components

(continued)

BASES (continued)

LCO	An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.
APPLICABILITY	In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY. In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume.
ACTIONS	<u>A.1</u> If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

Verifying that secondary containment equipment hatches (e.g., the Refueling Floor roof hatch and the HPCI/RCIC room roof hatches) are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur and provides adequate assurance that exfiltration from the secondary containment will not occur. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.2

Verifying that one secondary containment access door in each access opening is closed provides adequate assurance that exfiltration from the secondary containment will not occur. An access opening contains at least one inner and one outer door. In some cases, secondary containment access openings are shared such that there are multiple inner or outer doors. The intent is to not breach secondary containment, which is achieved by maintaining the inner and outer portion of the barrier closed except when the access opening is being used for entry and exit.

SR 3.6.4.1.2 is modified by a Note that applies to doors located in high radiation areas and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these doors, once they have been verified to be in the proper position, is low.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.4.1.3

The SBGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. SR 3.6.4.1.3 demonstrates that one SBGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge under calm wind conditions (i.e., less than 15 mph wind speed) at a flow rate ≤ 4000 cfm. This cannot be accomplished if the secondary containment boundary is not intact. Therefore, this test is used to ensure secondary containment boundary integrity. Since this SR is a secondary containment test, it need not be performed with each SBGT subsystem. The SBGT subsystems are tested on an alternating basis, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SBGT subsystem will perform this test, and also to ensure that the secondary containment remains sufficiently leak tight, even with a worst case single failure present (i.e., a lockout relay failure that results in either all of the inboard or all of the outboard SCIV/Ds failing to close). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Operating experience has shown these components usually pass the Surveillance when performed at this Frequency.

Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIV/Ds is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIV/Ds OPERABLE is not required in MODE 4 or 5.

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. For isolation devices requiring local operation, these controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. For isolation devices that can be operated remotely from the control room, the isolation device handswitch is tagged per plant procedures, identifying that the isolation device is open under administrative control and must be closed should an isolation signal occur. In the event of an isolation signal, plant procedures direct control room operators to verify all automatic actions occur, and to manually initiate those automatic actions that should have occurred but did not. This will ensure the control room operators verify any isolation devices open under administrative control close in response to an isolation signal. If any of the open isolation devices are unable or fail to close automatically, the control room operators will manually close them.

Note 1 also expands upon the allowance of LCO 3.0.5, which would only allow the penetration to be opened for testing, by allowing the penetration to be opened for other operational reasons, such as draining, venting, etc. The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition

(continued)

BASES (continued)

ACTIONS

C.1 and C.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

Verifying that the isolation time of each power operated automatic SCIV/D is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV/D will isolate in a time period less than or equal to that assumed in the safety analyses. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.4.2.2

Verifying that each automatic SCIV/D closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or which are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment. This SR ensures that each automatic SCIV/D will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based

(continued)

BASES (continued)

LCO

Following a DBA, a minimum of one SBGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SBGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SBGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SBGT System in OPERABLE status is not required in MODE 4 or 5.

ACTIONS

A.1

With one SBGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this Condition, the remaining OPERABLE SBGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SBGT subsystem and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the SBGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If both SBGT subsystems are inoperable in MODE 1, 2, or 3, the SBGT System may not be capable of supporting the required radioactivity release control function. Therefore, actions are required to enter LCO 3.0.3 Immediately.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SBGT subsystem ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage or fan or motor failure, can be detected for corrective action. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system, however these components are not the most-limiting for overall system reliability at this SR Frequency.

SR 3.6.4.3.2

This SR verifies that the required SBGT filter testing is performed in accordance with Specification 5.5.7, Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, system flow capability, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

LCO (continued)

In order for the SFU subsystems to be considered OPERABLE, the CBE boundary must be maintained such that the CBE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs. In the event of an inoperable CBE boundary in MODES 1, 2, or 3, mitigating actions are required to ensure CBE occupants are protected from hazardous chemicals and smoke.

DAEC does not have automatic SFU actuation for hazardous chemicals or smoke. Current practices at DAEC do not utilize chemicals in sufficient quantity to present a chemical hazard to the CBE. Smoke is not considered in the DAEC safety analysis. Therefore, there are no specific limits at DAEC for hazardous chemicals or smoke.

The LCO is modified by a Note allowing the CBE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CBE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through the doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CBE. This individual will have a method to rapidly close the opening and to restore the CBE boundary to a condition equivalent to the design condition when a need for CBE isolation is indicated.

APPLICABILITY

In MODES 1, 2, and 3, the SFU System must be OPERABLE to ensure that the CBE will remain habitable during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the SFU System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

(continued)

BASES

APPLICABILITY (continued)

a. During CORE ALTERATIONS; and

b. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one SFU subsystem inoperable, for reasons other than an inoperable CBE boundary, the inoperable SFU subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE SFU subsystem is adequate to perform the CBE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE subsystem could result in loss of the SFU System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CBE boundary and into the CBE can result in CBE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), the CBE boundary is inoperable. As discussed in the Applicable Safety Analyses section, the DAEC licensing basis identifies that CBE inleakage limits for hazardous chemicals and smoke are not necessary to protect the CBE occupants. Allowing verification by administrative means for hazardous chemicals and smoke is considered acceptable, since the limit established for radiological events is the limiting value for determining entry into Condition B for an inoperable CBE boundary. These administrative controls consist of the following:

- Verification that the periodic check of onsite and offsite hazardous chemical sources has been performed within the last year; and

- Verification that the smoke analysis of Reference 7 remains valid and current.

(continued)

ACTIONS (continued)

C.1 and C.2 (continued)

conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1, D.2.1, and D.2.2

LCO 3.0.3 is not applicable in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment or during CORE ALTERATIONS, if the inoperable SFU subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE SFU subsystem may be placed in the isolation mode (i.e., one SFU subsystem in operation with the control building isolated). This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected. An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the CBE. This places the unit in a condition that minimizes the accident risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position.

E.1

If both SFU subsystems are inoperable in MODE 1, 2, or 3 for reasons other than an inoperable CBE boundary (i.e., Condition B), the SFU System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

F.1 and F.2

LCO 3.0.3 is not applicable in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition F are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment or during CORE ALTERATIONS, with two SFU subsystems inoperable, or with one or more SFU subsystems inoperable due to an inoperable CBE boundary, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the CBE. This places the unit in a condition that minimizes the accident risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

Operating each SFU subsystem for ≥ 15 minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage or fan or motor failure, can be detected for corrective action. Since the SFU charcoal is tested at a Relative Humidity ≥ 95%, extended operation of the electric heaters is not required. Thus, each subsystem need only be operated for ≥ 15 minutes to demonstrate the function of each subsystem. The function of the SFU electric heaters is to pre-heat incoming air to above 40°F to ensure adsorption occurs within the temperature range that charcoal testing is performed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

BASES (continued)

LCO

Two independent and redundant subsystems of the CBC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The CBC System is considered OPERABLE when the individual components necessary to maintain the control building temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, chillers, compressors, ductwork, dampers, and associated instrumentation and controls. A CBC is considered inoperable if it trips and cannot be promptly restarted. Therefore, a CBC that spuriously trips and can subsequently be restarted in a reasonable period of time, is not considered inoperable. In addition, during conditions in MODES other than MODES 1, 2, and 3 when the CBC System is required to be OPERABLE (e.g., during CORE ALTERATIONS), the necessary portions of the ESW System, RWS System, and the Ultimate Heat Sink are also required as part of the OPERABILITY requirements covered by this LCO.

APPLICABILITY

In MODE 1, 2, or 3, the CBC System must be OPERABLE to ensure that the control building temperature will not exceed equipment OPERABILITY limits.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident (DBA) are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CBC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During CORE ALTERATIONS; and

b. During movement of irradiated fuel assemblies in the secondary containment.

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1 and D.2.2

LCO 3.0.3 is not applicable in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment or during CORE ALTERATIONS, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CBC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position.

E.1 and E.2

LCO 3.0.3 is not applicable in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

During movement of irradiated fuel assemblies in the secondary containment or during CORE ALTERATIONS, if Required Actions B.1 and B.2 cannot be met within the required Completion Times, action must be taken to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control building. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR verifies that the heat removal capability of the system is sufficient to remove available control building heat load. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is appropriate since significant degradation of the CBC System is not expected over this time period.

REFERENCES

1. UFSAR, Section 9.4.4.2.

BASES (continued)

SUREVEILLANCE REQUIREMENTS

SR 3.7.9.1

Operating each CB/SBGT Instrument Air compressor for ≥ 20 minutes allows the oil and other components to reach their operating temperature. This periodic operation removes condensation which may cause rusting in the cylinders, if it were to accumulate. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency and the operating time are based on vendor recommendations.

SR 3.7.9.2

This SR verifies that each CB/SBGT Instrument Air subsystem has the capability to deliver suffcient quantity of compressed air to support the SBGT, SFU, CBC, and containment isolation functions. This SR takes into account both the compressor capacity and the integrity of the distribution system.

This SR also verifies the automatic start capability of the CB/SBGT Instrument Air compressor in each subsystem. This is demonstrated by the use of an actual or simulated initiation signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is consistent with the Frequency for pump testing in accordance with the INSERVICE TESTING PROGRAM requirements. Therefore, this Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.3.1.2.1.
2. UFSAR, Section 6.2.4.
3. UFSAR, Section 6.2.5.
4. UFSAR, Section 6.5.3.3.
5. UFSAR, Section 6.4.2.
6. UFSAR, Section 9.4.4.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources — Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred and alternate preferred), and the onsite standby power sources (Diesel Generators (DGs) 1G-31 and 1G-21). As discussed in UFSAR Section 3.1.2.2.8 (Ref. 1), the design of the AC Electrical Power System provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) Systems via essential buses 1A3 and 1A4.

The Class 1E AC Distribution System is divided into redundant load groups, so loss of any one group does not prevent the minimum safety functions from being performed. Each load group has connections to two preferred offsite power supplies and a single DG.

Offsite power is supplied to the 161 kV and 345 kV switchyards from the transmission network by six transmission lines. The 345 kV switchyard and the 161 kV switchyard are connected via the autotransformer, and both sections of the switchyard are connected to the transmission grid by at least two independent lines. From the 161 kV switchyard preferred power source breakers CB5550 and/or CB5560, a single overhead transmission line feeds the startup transformer. From the startup transformer, dual isolated secondary windings provide feeds to the 4160 volt essential buses, 1A3 and 1A4, through separate bus supply lines and circuit breakers. The startup transformer is sized to supply all plant power (both essential and non-essential loads) during unit startup. From the 161 kV switchyard alternate preferred power source breaker CB8490, a single 34.5 kV underground line feeds the standby transformer. From the standby transformer, a single 4160 volt line feeds both essential buses through separate bus supply circuit breakers. A detailed description of the offsite power network and circuits to the onsite Class 1E essential buses is found in the UFSAR, Sections 8.2.1.3 and 8.3.1.1.5 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls

(continued)

BASES

LCO (continued)

(1G-31 and 1G-21) 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 DBA. Qualified offsite circuits are those that are described in the UFSAR, and are part of the licensing basis for the unit.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the essential buses. In accordance with commitments made in response to Generic Letter 2006-02 (Ref. 4), Condition C is entered whenever the grid operator (e.g., Midwest Independent System Operator (MISO)) determines that offsite power grid conditions are such that a trip of the DAEC turbine/generator would lead directly to voltages in the DAEC switchyard below the trip setpoints for Loss of Power (LOP) Instrumentation (LCO 3.3.8.1). The two offsite circuits consist of: 1) either the incoming circuit breaker (1400) and disconnects (1401 and 1402) or incoming circuit breaker (2690) and disconnects (2691 and 2692), the overhead 161 kV buswork, circuit breaker (1590) and disconnect (1591), the Reserve Auxiliary Transformer (1X5), circuit breaker (8490) and disconnect (8491), the underground 34.5 kV line, the standby transformer (1X4), the 4160 volt supply line and the two supply circuit breakers (1A301 and 1A401) to essential buses 1A3 and 1A4, respectively, and 2) either the incoming circuit breaker (5550) and disconnects (5551 and 5552) or incoming circuit breaker (5560) and disconnects (5553 and 5555), the overhead 161 kV line, the startup transformer (1X3), the two 4160 volt supply lines and the two supply circuit breakers (1A302 and 1A402) to essential buses 1A3 and 1A4, respectively.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective essential bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the essential buses. Proper sequencing of loads, including non-essential load shedding capability, is a required function for DG OPERABILITY.

The AC sources must be separate and independent (to the extent possible) of other AC sources. For the DGs, the separation and independence are complete. For the offsite AC sources, the separation and independence are to the extent practical. A circuit may be connected to more than

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and the capability to reject the largest single load and return to the required voltage and frequency (i.e. - voltage $\geq$ 3744 V and $\leq$ 4576 V and frequency $\geq$ 59.5 Hz and $\leq$ 60.5 Hz) within predetermined periods of time (i.e., 1.3 seconds for voltage and 3.9 seconds for frequency) while maintaining an acceptable margin to the overspeed trip. The largest single load for each DG is a core spray pump motor (700 hp). This Surveillance may be accomplished by tripping its associated single largest post-accident load with the DG solely supplying the bus.

As specified by IEEE-308 (Ref. 14), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower. For both DGs, this represents 64.5 Hz, equivalent to 75% of the difference between nominal speed and the overspeed trip setpoint.

The time, voltage, and frequency tolerances specified in the Bases for this SR are derived from UFSAR Table 8.3-1 (Ref. 16) recommendations for response during load sequence intervals. The voltage and frequency are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are the steady state voltage and frequency to which the system must recover following load rejection within a predetermined time period. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the Electrical Distribution Systems that could challenge continued steady state operation and, as a result, plant safety systems. Credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.12

Under either LOCA conditions or during a loss of offsite power, loads are sequentially connected to the bus by a timed logic sequence using individual time delay relays. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. Verifying the load sequence time interval is greater than or equal to 2 seconds ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load. The Allowable Values for the Core Spray and Low Pressure Coolant Injection Pump Start - Time Delay Relays, Table 3.3.5.1-1, Functions 1.e and 2.e, ensure this time interval is maintained as well as ensuring that safety analysis assumptions regarding ESF equipment time delays are not violated. Allowances for instrument inaccuracies in the load sequence time interval are also accounted for by the Pump Start - Time Delay Relay Allowable Value.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources — Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources — Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shutdown the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and corresponding stresses result in the probabilities of occurrences significantly reduced or eliminated, and minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, and 3, various deviations from the analysis assumptions and design requirements are allowed within the

(continued)

BASES

LCO (continued)

manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents). Automatic initiation of the required DG during shutdown conditions is specified in LCO 3.3.8.1, "LOP Instrumentation".

The qualified offsite circuit(s) must be capable of maintaining rated frequency and voltage while connected to their respective essential bus(es), and of accepting required loads during an accident. Qualified offsite circuits are those that are described in the UFSAR and are part of the licensing basis for the unit. The required offsite circuit consists of either: 1) either the incoming circuit breaker (1400) and disconnects (1401 and 1402) or incoming circuit breaker (2690) and disconnects (2691 and 2692), the overhead 161 kV buswork, circuit breaker (1590) and disconnect (1591), the Reserve Auxiliary Transformer (1X5), circuit breaker (8490) and disconnect (8491), the underground 34.5 kV line, the standby transformer (1X4), the 4160 volt supply line and the two supply circuit breakers (1A301 and 1A401) to essential buses 1A3 and 1A4, respectively, or 2) the incoming circuit breaker (5550 or 5560) and disconnect (5551 or 5555, respectively), the overhead 161 kV line, the startup transformer (1X3), one of the two 4160 volt supply lines and one of the two supply circuit breakers (1A302 or 1A402) to essential buses 1A3 or 1A4, respectively, if required by LCO 3.8.8.

The required DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective essential bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the essential buses. The necessary portions of Emergency Service Water are also required to provide appropriate cooling to each required DG.

Proper sequencing of loads, including non-essential load shedding capability, is a required function for DG OPERABILITY.

In addition, proper timed logic sequence operation, is an integral part of offsite circuit OPERABILITY since its inoperability could impact the ability to start and maintain energized loads required OPERABLE by LCO 3.8.8. No automatic transfer capability is required for offsite circuits to be considered OPERABLE during shutdown conditions.

(continued)

BASES (continued)

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment to provide assurance that:

a. Systems that provide core cooling are available;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Actions have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1

An offsite circuit is considered inoperable if it is not available to supply power to either of the essential buses. If both essential 4.16 kV buses are required per LCO 3.8.8, one division with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare required

(continued)

BASES (continued)

ACTIONS

A.1 (continued)

features inoperable that are not powered from offsite power, appropriate restrictions can be implemented in accordance with the affected required feature(s) LCOs' ACTIONS.

Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action.

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3,

With the required offsite circuit not available to either division, the option still exists to declare all affected required features inoperable per required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to Immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of Immediately is consistent with the required times for ACTIONS requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3, (continued)

during which the plant safety systems may be without sufficient power. Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A have been modified by a Note to indicate that when Condition A is entered with no AC power to any required essential bus, ACTIONS for LCO 3.8.8 must be Immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a division is de-energized. LCO 3.8.8 provides the appropriate restrictions for the situation involving a de-energized division.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, and 3. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

This SR is modified by two Notes. The reason for Note 1 is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during the performance of SRs, and to preclude deenergizing a required 4160 V essential bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE.

Note 2 states that SR 3.8.1.13 is considered to be met without the ECCS initiation signals OPERABLE when the ECCS initiation signals are not required to be OPERABLE per Table 3.3.5.1-1. This SR demonstrates the DG response to an ECCS signal (either alone or in conjunction with a loss-of-power signal). This is consistent with the ECCS instrumentation requirements of Table 3.3.5.1-1 that do not require the ECCS signals to be OPERABLE in MODES 4 and 5 when ECCS is not required to be OPERABLE.

REFERENCES

None.

BASES (continued)

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the 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 DBA. Because stored diesel fuel oil, lube oil, and starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG for Conditions B, E, and F. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) governed by separate Condition entry and application of associated Required Actions.

A.1

In this Condition, the 7 day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. The fuel oil level equivalent to a 6 day supply is 32,689 gallons. These circumstances may be caused by events such as:

a. Full load operation required for an inadvertent start while at minimum required level; or

b. Feed and bleed operations that may be necessitated by increasing particulate levels or any number of other oil quality degradations.

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tank to support a single DG's operation for 7 days at full load. The fuel oil level equivalent to a 7 day supply is 37,967 gallons when calculated in accordance with References 2 and 3. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel output, and the corresponding fuel consumption rate, the onsite fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lubricating oil inventory is available to support at least 7 days of full load operation for each DG. The lube oil inventory equivalent to a 7 day supply is 257 gallons and is based on the DG manufacturer's consumption values for the run time of the DG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from the lube oil makeup tank to the DG. The requirement is considered to be fulfilled by observing that the DG lube oil sump level is maintained in the normal band by the lube oil sump level controller.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. A Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the plant staff.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.3 (continued)

inspection of cell condition and rack integrity.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of inter-cell, inter-rack, inter-tier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection.

The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR, provided visible corrosion is removed during performance of this Surveillance. The connection resistance limits for this SR must be no more than 20% above the resistance as measured during installation, or not above the ceiling value established by the manufacturer. The resulting limits are 5.0 E-5 ohms for inter-cell connections and 1.4 E-4 ohms for inter-rack connections, inter-tier connections and terminal connections.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of these SRs is consistent with the intent of IEEE-450 (Ref. 7), which recommends detailed visual inspection of cell condition and inspection of cell to cell and terminal connection resistance.

SR 3.8.4.6

Battery charger capability requirements are based on the design capacity of the chargers (Ref. 3). According to the recommendations of Regulatory Guide 1.32 (Ref. 8), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurences. The minimum required amperes ensures that these requirements can be satisfied. The minimum required voltage ensures that the requirements are satisfied at the minimum float voltage that the battery chargers are normally set for. Charger operation at an output greater than the minimum required voltage and amperage, such as during equalize mode,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.6 (continued)

demonstrates the battery charger meets, and exceeds, the Surveillance Requirements.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance on a required battery charger would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. This Note does not preclude performance of this SR on the "spare" battery charger (i.e., a charger not in-service or "required"). This Note also acknowledges that credit may be taken for unplanned events that satisfy the Surveillance.

SR 3.8.4.7

A battery service test is a special test of the battery's capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements as specified in Reference 4. The voltage of each cell shall be determined after the discharge. Following discharge, battery cell parameters must be restored in accordance with LCO 3.8.6. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is consistent with the maximum length of an operating cycle.

This SR is modified by two Notes. Note 1 allows the performance of a performance discharge test in lieu of a service test.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources — Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources — Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 15 (Ref. 1), assume that Engineered Safety Feature Systems are OPERABLE. The 125 VDC Electrical Power System provides normal and emergency DC electrical power for the Diesel Generators (DGs), emergency auxiliaries, and control and switching during all MODES of operation and during movement of irradiated fuel assemblies in the Secondary Containment.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The DC electrical power subsystems — with: Division I and Division II 125 VDC subsystems each consisting of one 125 V battery, the associated battery charger or the swing battery charger and the corresponding control equipment and interconnecting cabling supply power to the associated distribution system; and, the 250 VDC subsystem consisting of the 250V battery, one of the two battery chargers and the corresponding control equipment and interconnecting cabling sufficient to provide electrical power to the outboard RHR-SDC suction isolation valve (MO-1909), are required to be OPERABLE to support required DC distribution subsystems required OPERABLE by LCO 3.8.8, "Distribution Systems —Shutdown." This requirement ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:

a. Required features to provide core cooling are available;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

(continued)

BASES (continued)

ACTIONS

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Actions have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, and A.2.3

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC electrical power subsystems remaining OPERABLE with one or more DC electrical power subsystems inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowance of the option to declare required features inoperable with associated DC electrical power subsystems inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. However, in many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems — Shutdown

BASES

BACKGROUND	A description of the AC and DC Electrical Power Distribution System is provided in the Bases for LCO 3.8.7, "Distribution Systems — Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 15 (Ref. 1), assume Engineered Safety Feature (ESF) Systems are OPERABLE. The AC and DC Electrical Power Distribution Systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF Systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC and DC Electrical Power Distribution System is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5, and during movement of irradiated fuel assemblies in the secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the Electrical Distribution System necessary to support OPERABILITY of Technical Specifications required systems, equipment, and components — both specifically addressed by their own LCO, and implicitly required by the definition of OPERABILITY. In addition, it is acceptable for required buses to be cross-tied during shutdown conditions, permitting a single source to supply multiple redundant buses, provided the source is capable of maintaining proper frequency (if required) and voltage.

Maintaining these portions of the Distribution System energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:

a. Systems that provide core cooling;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

(continued)

BASES (continued)

APPLICABILITY (continued)

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Actions have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

Although redundant required features may require redundant Divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made, (i.e., to suspend CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment).

(continued)

BASES (continued)

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

Not withstanding performance of the above conservative Required Actions, a required Residual Heat Removal-Shutdown Cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.3 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS would not be entered. Therefore, Required Action A.2.4 is provided to direct declaring RHR-SDC inoperable, which results in taking the appropriate RHR-SDC ACTIONS.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

these requirements will conservatively limit radiation releases to the environment.

In the unlikely event of any primary system leak that could result in draining the RPV, the reactor vessel would rapidly depressurize. The makeup capability required in MODE 4 by LCO 3.5.2, "RPV Water Inventory Control," would be more than adequate to keep the RPV water level above the TS 2.1.1.3 Safety Limit under this low decay heat load condition. Small system leaks would be detected by leakage inspections before significant inventory loss occurred.

For the purposes of this test, the protection provided by normally required MODE 4 applicable LCOs, in addition to the secondary containment requirements required to be met by this Special Operations LCO, will ensure acceptable consequences during normal hydrostatic test conditions and during postulated accident conditions.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Operation at reactor coolant temperatures > 212°F can be in accordance with Table 1.1-1 for MODE 3 operation without meeting this Special Operations LCO or its ACTIONS. This option may be required due to P/T limits, however, which require testing at temperatures > 212°F, while some system leakage or hydrostatic testing may require the safety/relief valves to be gagged, preventing their OPERABILITY. Additionally, even with required minimum reactor coolant temperature ≤ 212°F, RCS temperatures may drift above 212°F during the performance of system leakage and hydrostatic testing or during subsequent control rod scram time testing, which is typically performed in conjunction with a system leakage or hydrostatic test. While this Special Operations LCO is provided for system leakage and hydrostatic testing, and for scram time testing initiated in conjunction with a system leakage or

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