NEXTera ENERGY DUANE ARNOLD

Enclosure 2 Contains ~~Proprietary Information~~

~~Withhold Enclosure 2 from Public Disclosure in Accordance with 10 CFR 2.390~~

April 28, 2017

NG-17-0086
10 CFR 50.71(e)
10 CFR 54.37(b)

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

Duane Arnold Energy Center
Docket 50-331
Renewed Op. License No: DPR-49

Transmittal Of Revision 24 Of The DAEC Updated Final Safety Analysis Report And The Current DAEC Technical Specifications Bases

This letter transmits the latest revision (No. 24) of the Updated Final Safety Analysis Report (UFSAR) for the Duane Arnold Energy Center (DAEC) as required by 10 CFR Section 50.71(e) and reflects changes made to the UFSAR since the previous submittal in May, 2015. Enclosure 1 to this letter summarizes the changes included in this revision. Enclosure 2 includes 1 DVD-ROM containing the non-public version of the DAEC UFSAR in electronic format and is being submitted in its entirety, constituting a total replacement copy. Enclosure 2 contains security related information as defined by 10 CFR 2.390(d) and should be withheld from public disclosure.

Enclosure 3 to this letter includes 1 DVD-ROM containing the public version of the DAEC UFSAR.

The DAEC Technical Requirements Manual (TRM) is incorporated by reference into UFSAR Chapter 16. Therefore, Enclosure 4 contains the TRM changed pages and represents all changes made to the TRM since the previous submittal in May, 2015.

Enclosure 2 transmitted herewith contains ~~Proprietary Information.~~
~~**When separated from Enclosure 2, this document is decontrolled.**~~

A053
NRR

The DAEC Technical Specifications Bases Control Program requires that changes to the Technical Specifications Bases (TSB) implemented without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Therefore, in accordance with Section 5.5.10.d of the DAEC Technical Specifications, Enclosure 5 contains the TSB changed pages and represents all changes made to the TSB since the previous submittal in May, 2015.

NextEra Energy Duane Arnold conducted a review of the DAEC plant changes for 10 CFR 54.37(b) applicability. No components were determined to meet the criteria for newly identified components as clarified by Regulatory Issue Summary (RIS) 2007-16, Revision 1, "Implementation of the Requirements of 10 CFR 54.34(b) for Holders of Renewed Licenses."

This letter contains no new commitments.

Please call Michael Davis at (319) 851-7032 with any questions regarding this submittal.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 28, 2017.

Dean Curtland

Dean Curtland
Site Director
NextEra Energy Duane Arnold, LLC

Enclosures (5)

cc Administrator Region III, USNRC
Project Manager, DAEC, USNRC
Resident Inspector, DAEC, USNRC

ENCLOSURE 1
To NG-17-0086

SUMMARY OF CHANGES

Revision 24 of the Duane Arnold Energy Center Updated Final Safety Analysis (UFSAR), excluding editorial changes, is summarized below.

Change Request Number	Description
2014-013	Add Discussion Regarding Non-Safety-Related-Start of Standby Generator System
2015-001	Revise the Recommended Power Grid Operating Voltage Per EC283160
2015-007	Incorporate Amendment 285 Regarding Standby Gas Treatment Heaters
2015-008	Revise Discussion of Ground Monitoring Wells in Accordance with EC283717
2015-009	Reflect Organizational Changes
2015-010	Revise Radwaste Dilution Structure Description per EC283505
2015-011	Reflect FLEX Modifications in Accordance with EC280489
2015-012	Add Description of Spent Fuel Pool Level Probes Installed in Accordance with NRC Order EA-12-051 Per EC283472
2015-013	Correct Editorial Error Regarding Residual Heat Removal – Service Water Pump Flow
2015-014	Revise Remote Shutdown Panel Instrumentation Description Per EC283522
2015-015	Clarify Discussion of Residual Heat Removal – Service Water and Emergency Service Water Pump House Ventilation
2016-001	Update Discussion of Voice-Over-Internet Communications
2016-002	Revise Internal Flooding Discussion Per EC285633
2016-003	Reflect Hardened Containment Vent In Accordance with EC281991
2016-004	Revise Discussion of Instrument Shop Location
2016-005	Incorporate Amendment 292 Regarding Emergency Diesel Generator Fuel Oil
2016-006	Clarify Dose Consequence Calculation Regarding Use of RADTRAD Computer Code
2016-007	Revise Discussion Regarding Location of Steam Leak Detection Temperature Elements
2016-008	Clarify Discussion of Residual Heat Removal Heat Exchanger
2016-009	Reflect Removal of Radwaste Trash Compactor in Accordance with EC286818
2016-010	Revise Discussion of Ground Monitoring and Extraction Wells in Accordance with EC285995
2016-011	Remove Discussion of System Dispatch Control Center Capability per EC287263
2016-014	Revise Discussion of Residual Heat Removal and Core Spray Pressure Switches

2016-015	Reflect ITC Midwest's Transmission System Upgrades
2017-001	Reflect Cycle 26 Core Modification Package Per EC284397
2017-002	Reflect Organization Changes

Per the guidance of NEI 98-03, Revision 1, the following changes constitute UFSAR deletions. A description of those UFSAR deletions and the basis for change is provided below.

<u>Change Request Number</u>	<u>Description and Basis for Removal</u>
2016-004	Description: This change request removed discussion of the location of the Instrument Shop. Basis: 10 CFR 50.59 was applied to this change and determined that removal of the information did not adversely affect a design function.
2016-009	Description: This change request removed discussion of the Radwaste Trash Compactor. Basis: 10 CFR 50.59 was applied to this change and determined that removal of the information did not adversely affect a design function.
2016-011	Description: This change request removed discussion System Dispatch Control Center Capability. Basis: 10 CFR 50.59 was applied to this change and determined that removal of the information did not adversely affect a design function.
2017-001	Description: This change request removed GE14 fuel legacy information in accordance with the Cycle 26 Core Modification Package. Basis: 10 CFR 50.59 was applied to this change and determined that removal of the information did not adversely affect a design function.

Additional UFSAR deletions resulted from plant or program modifications which were evaluated under 10 CFR 50.59, License Amendments, NRC Safety Evaluation Reports issued to DAEC, or constituted removal of redundant information which remains located elsewhere in the UFSAR.

Enclosure 2
to NG-17-0086

DVD-ROM of the DAEC UFSAR Revision 24
NON-PUBLIC VERSION

Enclosure 2 transmitted herewith contains ~~Proprietary Information.~~
~~When separated from Enclosure 2, this document is decontrolled~~.

Enclosure 3
to NG-17-0086

DVD-ROM of the DAEC UFSAR Revision 24
PUBLIC VERSION

Enclosure 4
to NG-17-0086

DAEC Technical Requirements Manual

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TB 3.4-2	10/20/99	2
TB 3.5-1	10/20/99	2
TB 3.5-2	04/18/05	8
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T 1.0 USE AND APPLICATION

The Technical Requirements Manual (TRM) contains TRM Limiting Conditions for Operation (TLCOs) and operational conveniences, such as lists, cross references, acceptance criteria, and drawings.

The TLCOs are contained in Section 3.0 and include operational requirements, TRM Surveillance Requirements (TSRs), and Required Actions for inoperable equipment. Instructions for the use and application of TLCOs are included in Section 1.0.

Operational conveniences provide a ready reference to lists and other helpful tools described in plant procedures and programs.

The TRM is a licensing document and changes to this manual are governed by Administrative Control Procedure ACP-102.19, Preparation, Revision, and Processing of Technical Requirements Manual Change Requests. While the TLCOs are to be treated like Technical Specifications from an implementation viewpoint, the TLCOs are essentially procedures. Therefore, unless specifically stated in the TLCO, entry into or violation of a TRM Required Action, or violation of a TRM Surveillance Requirement is not specifically reportable per 10 CFR 50.72 or 10 CFR 50.73. Likewise, power reductions and/or plant shutdowns required to comply with TRM ACTIONS are not specifically reportable per 10 CFR 50.72(b)(1)(i)(A) or 10 CFR 50.73(a)(2)(i)(A)or (a)(2)(i)(B). Failure to comply with TLCO requirements shall be treated as a failure to follow procedure and entered into the Action Request (AR) system, as appropriate.

(continued)

3.0 SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TSR 3.0.1

TSRs shall be met during the MODES or other specified conditions in the Applicability for individual TLCOs, unless otherwise stated in the TSR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the TLCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the TLCO except as provided in TSR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

TSR 3.0.2

The specified Frequency for each TSR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this TSR are stated in the individual TLCOs.

TSR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the TLCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the TLCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

T 3.5 DRYWELL SPRAY SYSTEM AND ES COMPARTMENT COOLING AND VENTILATION

T 3.5.2 Engineered Safeguards (ES) Compartment Cooling and Ventilation

TLCO 3.5.2 The following ECCS and RCIC unit coolers shall be OPERABLE:

a. One RCIC room unit cooler;

b. One HPCI room unit cooler; and

c. Two CS/RHR room unit coolers.

APPLICABILITY: When the associated pumps are required to be OPERABLE.

ACTIONS

-----NOTES-----

1. Separate condition entry is allowed for each unit cooler.

2. TLCO 3.0.4.b is N/A for HPCI and RCIC Room Coolers.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required unit room cooler automatic start feature inoperable.	A.1 Manually start the affected unit room cooler. <u>AND</u> A.2 Restore unit room cooler automatic start feature to OPERABLE status.	Immediately 7 days

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
B.	Required Action and associated Completion Time for Condition A not met. OR Required unit room cooler inoperable.	B.1	Declare the associated pump(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.5.2.1	-------------------------NOTE---------------------------- Only required to be performed in conjunction with Surveillances of the associated pump. -- Verify forced air circulation by the required unit coolers.	92 days
TSR 3.5.2.2	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.11.2.1	Verify the each valve in the flow path is in its correct position.	1 month
TSR 3.11.2.2	Perform flush of yard header.	6 months
TSR 3.11.2.3	Conduct hydraulic performance test of the system by starting the motor driven fire pump and directing flow around the yard header. Verify flow and pressure at the pump discharge of at least 2615 gpm at 89 psig or greater.	36 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.11.3.1	Start the diesel-driven fire pump and operate for at least 30 minutes.	1 week
TSR 3.11.3.2	Start the motor-driven fire pump and operate for at least 15 minutes on recirculation flow.	1 month
TSR 3.11.3.3	Verify the diesel day tank contains fuel for two hours of operation.	1 month
TSR 3.11.3.4	Verify that a sample of diesel fuel from the storage tank, obtained in accordance with ASTM-D4057-95, is within the acceptable limits specified in Table 1 of ASTM-D975-77 with respect to viscosity, water content and sediment.	3 months
TSR 3.11.3.5	Verify each fire pump develops at least 2615 gpm with a discharge pressure of at least 89 psig.	12 months
TSR 3.11.3.6	Verify the diesel starts from ambient conditions on the auto-start signal and operates for 30 minutes while loading with the fire pump.	12 months

Table 4.0-3h (ECCS Instrumentation)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.3.5.1.9	LOGIC SYSTEM FUNCTIONAL TEST	ECCS Functions: 1a, 1b, 1c, 1d, 1e, 1f, 2a, 2b, 2c, 2d, 2e, 2f, 2g, 2h, 2i, 2j, 2k, 3a, 3b, 3c, 3d, 3e, 3f, 4a, 4b, 4c, 4d, 4e, 5a, 5b, 5c, 5d, 5e	3.3.1.1-05 3.3.5.1-02 3.3.5.1-05 3.3.5.1-06 3.3.5.1-08 3.3.5.1-10 3.3.5.1-11 3.3.5.1-13 3.3.5.1-14 3.3.5.1-15 3.3.5.1-16 3.3.5.1-18 3.3.5.1-20 3.3.5.1-22 3.3.5.1-24 3.3.5.1-26 3.3.5.1-28 3.3.5.1-29 3.3.5.1-30 3.3.5.1-32 3.3.5.1-33 3.3.5.1-37 3.5.1-01 3.5.1-02 3.5.1-03 3.5.1-04 3.5.1-05 3.5.1-07 3.5.1-10 3.5.1-11 3.5.1-12 3.5.3-05	24 Months	No	AR 2065602

Table 4.0-3j (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 4.0-3n (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		

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 & RCIC)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.5.2.1	Suppression Pool Level Check	LPCI	3.0.0-01 3.0.0-03	12 Hours	No	
3.5.2.2	CST & Suppression Pool Level Check	Core Spray	3.0.0-01 3.0.0-03	12 Hours	No	
3.5.2.3	Piping locations susceptible to gas accumulation are sufficiently full of water	Core Spray, LPCI	3.0.0-01 3.5.1-13 3.5.1-14 3.5.1-15	31 Days	No	
3.5.2.4	Valve Position Checks	Core Spray, LPCI	3.0.0-02	31 Days	**Yes**	
3.5.2.6	Simulated Auto Actuation Test	Core Spray, LPCI	3.3.5.1-15 3.3.5.1-29 3.3.5.1-37 3.5.1-03 3.5.1-04 3.6.1.3-06 3.8.7-01	24 Months	No	AR 2065602
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	

Table 4.0-6 (Containment)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.6.1.1.2	Suppression Chamber Leak Check	Primary Containment	3.6.1.1-05	24 Months	No	
3.6.1.2.2	Air Lock Door Interlock Check	Primary Containment	3.6.1.2-02	24 Months	No	
3.6.1.3.1	Verify Valves are Closed	Purge Valves	3.0.0-02	31 Days	No	
3.6.1.3.2	Explosive Valve Continuity Check	TIP Shear Valve	3.0.0-02	31 Days	No	
3.6.1.3.4	T-seal Leakage Check	Purge Valves	3.6.1.3-01	184 Days	No	
3.6.1.3.6	Simulated Auto Actuation Test	PCIVs	3.3.5.1-29 3.3.5.1-30 3.3.6.1-06 3.3.6.1-09 3.3.6.1-14 3.3.6.1-42 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	
3.6.1.3.7	Simulated Auto Actuation Test	EFCVs	3.6.1.3-04 3.6.1.3-07 3.6.1.3-08 3.6.1.3-09 3.6.1.3-10	24 Months	No	
3.6.1.4.1	Drywell Air Temperature Check	Primary Containment	3.0.0-01	24 Hours	No	
3.6.1.5.1	Valves Manually Open Check	LLS Valves	3.4.3-03	24 Months	No	
3.6.1.5.2	Simulated Auto Actuation Test	LLS Valves	3.3.6.3-05	24 Months	No	
3.6.1.6.1	Verify Valves are Closed	Rx Bldg – Suppression Chamber VBs	3.0.0-04	14 Days	No	
3.6.1.6.2	Valve Assembly Functional Test	Rx Bldg – Suppression Chamber VBs	3.6.1.6-01	92 Days	No	

Table 4.0-7 (Plant Systems)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.7.1.1	Verify Valve Positions	RHRSW	3.0.0-02	31 Days	No	
3.7.2.1	Verify River Water Level	UHS	3.0.0-01	24 Hours	No	
3.7.2.2	Verify River Water Temperature	UHS	3.0.0-01	24 Hours	No	
3.7.2.3	Verify River Water Depth	UHS	3.7.2-02	7 Days	**Yes**	
3.7.2.4	Verify Valve Positions	RWS	3.0.0-02	31 Days	No	
3.7.2.5	Verify River Water Depth	UHS	3.7.2-02	92 Days	No	
3.7.2.6	Simulated Auto Actuation Test	RWS	3.7.2-01 3.8.1-07	24 Months Staggered Test Basis	No	AR 2065602 AR 2133658
3.7.3.1	Verify Valve Positions	ESW	3.0.0-04	31 Days	No	
3.7.3.2	Simulated Auto Actuation Test	ESW	3.8.1-04	24 Months	No	AR 2065602
3.7.4.1	15 Minute Operational Run	SFU	3.7.4-05	31 Days	No	
3.7.4.3	Simulated Auto Actuation Test	SFU	3.7.4-01	24 Months	No	
3.7.5.1	Verify Heat Removal Capability	Chillers	3.7.5-01 3.7.5-03	92 Days	No	
3.7.6.1	Verify Gross Gamma Activity	Offgas	3.7.6-01	31 Days	**Yes**	
3.7.7.1	Fully Cycle Each Valve	Turbine Bypass Valves	3.7.7-01 3.7.7-02	92 Days	No	
3.7.7.2	Perform System Functional Test	Turbine Bypass Valves	3.7.7-02	24 Months	No	
3.7.7.3	Verify System Response Time	Turbine Bypass Valves	3.7.7-02	24 Months	No	
3.7.8.1	Verify Pool Water Level	Spent Fuel Pool	3.0.0-03 3.0.0-04	7 Days	No	
3.7.9.1	20 Minute Operational Test	Instrument Air System	3.7.9-01 3.7.9-02	31 Days	No	
3.7.9.2	Simulated Auto Actuation Test	Instrument Air System	3.7.9-03	92 Days	No	

Table 4.0-8 (Electrical Systems)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.8.1.1	Verify Breaker Alignment & Power Availability	Offsite AC Sources	3.8.1-01	7 Days	No	
3.8.1.2	Verify Start (Slow) - Output Voltage & Frequency	EDGs	3.8.1-03 3.8.1-04 3.8.1-05 3.8.1-06	31 Days	No	
3.8.1.3	60 Minute Load Test	EDGs	3.8.1-04 3.8.1-05 3.8.1-06	31 Days	**Yes**	
3.8.1.4	Verify Day Tank Volume	EDGs	3.8.1-04 3.8.1-05 3.8.1-06	31 Days	No	
3.8.1.5	Day Tank Presence of Water Check	EDGs	3.8.1-08 3.8.1-09	31 Days	No	
3.8.1.6	Fuel Oil Transfer Operational Test	EDGs	3.8.1-04 3.8.1-05 3.8.1-06	31 Days	No	
3.8.1.7	Timed (Fast) Start – Voltage & Frequency Check	EDGs	3.8.1-06	184 Days	**Yes**	
3.8.1.8	Verify Slow Transfer Between Start-up & Stand-by Transformers	Offsite AC Sources	3.8.1-07	24 Months Staggered Test Basis	**Yes**	AR 2065602
3.8.1.9	Load Reject Test	EDGs	3.8.1-07	24 Months Staggered Test Basis	**Yes**	AR 2065602

Table 4.0-8 (Electrical Systems)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.8.1.10	Verify Automatic Trip Bypass	EDGs	3.8.1-07	24 Months Staggered Test Basis	**Yes**	AR 2065602
3.8.1.11	Verify Manual Control for Transferring Loads to Offsite Power	EDGs	3.8.1-07	24 Months Staggered Test Basis	**Yes**	AR 2065602
3.8.1.12	Verify Load Sequence Separation	EDGs	3.8.1-07	24 Months Staggered Test Basis	**Yes**	AR 2065602
3.8.1.13	LOOP-LOCA Test	EDGs	3.7.5-03	24 Months	**Yes**	AR 2065602
			3.8.1-07	24 Months Staggered Test Basis		
3.8.3.1	Verify Volume	Fuel Oil	3.0.0-03 3.0.0-04 3.8.1-04 3.8.1-05 3.8.1-06	31 Days	No	
3.8.3.2	Verify Lube Oil Volume	LubeOil	3.0.0-03 3.0.0-04 3.8.1-04 3.8.1-05 3.8.1-06	31 Days	No	

Table 4.0-8 (Electrical Systems)

TS SR #	Subject	TS SSC/Function	STP #	Frequency	SR Frequency Note?	AR#
3.8.3.4	Verify Receiver Air Pressure	Starting Air	3.8.1-04 3.8.1-05 3.8.1-06 3.8.1-11	31 Days	No	
3.8.3.5	Underground Fuel Oil Tank Presence of Water Check	Fuel Oil	3.8.1-08 3.8.1-09	31 Days	No	
3.8.4.1	Verify Terminal Voltage	DC Sources	3.8.4-01	7 Days	No	
3.8.4.2	Battery Terminals Corrosion Check	DC Sources	3.8.4-02	92 Days	No	
3.8.4.3	Battery Condition Visual Checks	DC Sources	3.8.4-05	12 Months	No	
3.8.4.4	Corrosion Removal & Connection Coating Checks	DC Sources	3.8.4-05	12 Months	No	
3.8.4.5	Verify Connection Resistance	DC Sources	3.8.4-05	12 Months	No	
3.8.4.6	Verify Charger Output Voltage & Amperage	DC Source	3.8.4-06 3.8.4-09	24 Months	**Yes**	
3.8.4.7	Battery Service Test	DC Sources	3.8.4-03 3.8.4-07	24 Months	**Yes**	
3.8.4.8.	Battery Modified Performance Discharge Test	DC Sources	3.8.4-04 3.8.4-08	60 Months	**Yes**	
3.8.6.1	Verify Category A Limits	Batteries	3.8.4-01	7 Days	No	
3.8.6.2	Verify Category B Limits	Batteries	3.8.4-02	92 Days	No	
3.8.6.3	Verify Electrolyte Temperature	Batteries	3.8.4-02	92 Days	No	
3.8.7.1	Verify Breaker Alignment & Power Availability	AC & DC Distribution Panels	3.8.7-02	7 Days	No	
3.8.7.2	Verify Breaker Coordination – LPCI Swing Bus	LPCI Swing Bus	3.8.7-01	24 Months	No	
3.8.8.1	Verify Breaker Alignment & Power Availability	AC & DC Distribution Panels	3.8.7-02	7 Days	No	

BASES (continued)

NFPA 805 3.5.16 Basis:
DAEC complies via both Exceptions.

Exception No. 1: The fire protection water system is identified as one of several alternate means of injection when normal injection systems are inadequate or unavailable for the purposes of restoring RPV level, flooding the RPV, proving a makeup water source to the SFP, spraying the primary containment atmosphere or flooding the primary containment. The fire water supply is provided by two fire pumps. One fire pump is sufficient for fire suppression demand flow. The second fire pump is sufficient to provide the demand flow for injection.

Exception No. 2: Water supply to the fire pumps is obtained from a 400,000 gallon wet pit in the pumphouse. The wet pit also provides a common suction to the circulating service water system and the general service water system. The wet pit is supplied by gravity drain from the cooling tower basins in the circulating water system. Based on water level, makeup is provided to the wet pit from up to four 6,000 gpm River Water pumps. The well water system is a backup to the River Water system. The 400,000 gallon storage capacity of the wet pit exceeds the capacity required for 2 hour operation of the largest sprinkler demand (sprinkler system #4 at 2,115 gpm) plus 500 gpm for hose streams.

References for Bases Section TB 3.11.2

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. AIP 404 Rev. 10 - Injection with Fire Water
5. CAL-465-M-006 Rev. 2 - Hydraulic Calculation for Fire Protection System #4
6. CAL-465-M-007 Rev. 0 - Hydraulic Calculation for Fire Protection System #4 – Additional Cases
7. CAL-M16-001 Rev. 0 – Fire Pump Minimum Flow and Pressure Requirement to Supply Most Hydraulically Demanding Suppression System with Hose Stream Allowance.

BASES (continued)

pumps are spaced about 10 feet apart, separated by a divisional valve with additional valves arranged to isolate either connection.

NFPA 805 3.5.8 Requirement:
A method of automatic pressure maintenance of the fire protection water system shall be provided independent of the fire pumps.

NFPA 805 3.5.8 Basis:
Pressure in the fire main system is automatically maintained by a small jockey pump that is supplied by the well water system.

NFPA 805 3.5.9 Requirement:
Means shall be provided to immediately notify the control room, or other suitable constantly attended location, of operation of fire pumps.

NFPA 805 3.5.9 Basis:
Pump supervisory signals are annunciated in the control room.

References for Bases Section TB 3.11.3

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. CAL-465-M-006 Rev. 2 - Hydraulic Calculation for Fire Protection System #4
5. CAL-465-M-007 Rev. 0 - Hydraulic Calculation for Fire Protection System #4 – Additional Cases
6. FPE-S08-002 Rev. 0 - Electric and Diesel Fire Pumps NFPA 20-1970 Code Compliance Evaluation
7. FPE-S08-003 Rev. 0 - Outside Fire Protection NFPA 24-1969 Code Compliance Evaluation
8. IE-77-138 - IELP letter from Liu to Rusche dated 1-18-1977 (4006006324)
9. IE-78-270 - IELP Letter to NRC, Liu to Lear
10. SER No. 1 [section 4.3.1 b.] - NRC Safety Evaluation Report dated June 1, 1978
11. BECH-M133 [Sheets 1 thru 5] - P&ID Fire Protection
12. AIP 404 Rev. 8 - Injection with Fire Water
13. CAL-M16-001 Rev. 0 – Fire Pump Minimum Flow and Pressure Requirement to Supply Most Hydraulically Demanding Suppression System with Hose Stream Allowance.

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B 3.9-26B	05/11/15	146
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B 3.9-28	05/11/15	146
B 3.9-29	03/17/99	019
B 3.9-30	08/01/98	223
B 3.9-31	05/11/15	146
B 3.9-32	05/11/15	146
B 3.10-1	02/16/07	078
B 3.10-2	02/16/07	078
B 3.10-3	02/16/07	078
B 3.10-4	02/16/07	078
B 3.10-5	02/16/07	078
B 3.10-5A	02/16/07	078
B 3.10-6	08/01/98	223
B 3.10-7	08/01/98	223
B 3.10-8	08/01/98	223
B 3.10-9	08/01/98	223
B 3.10-10	04/27/12	120
B 3.10-11	08/01/98	223
B 3.10-12	08/01/98	223
B 3.10-13	08/01/98	223
B 3.10-14	08/01/98	223
B 3.10-15	04/27/12	120
B 3.10-16	08/01/98	223
B 3.10-17	08/01/98	223
B 3.10-18	08/01/98	223
B 3.10-19	08/01/98	223
B 3.10-20	04/27/12	120
B 3.10-21	08/01/98	223
B 3.10-22	08/01/98	223
B 3.10-23	08/01/98	223
B 3.10-24	08/01/98	223
B 3.10-25	04/27/12	120
B 3.10-26	08/01/98	223
B 3.10-27	08/01/98	223
B 3.10-28	04/27/12	120
B 3.10-29	08/01/98	223
B 3.10-30	08/01/98	223
B 3.10-31	08/01/98	223
B 3.10-32	08/01/98	223
B 3.10-33	08/01/98	223
B 3.10-34	08/01/98	223
B 3.10-35	08/01/98	223
B 3.10-36	08/01/98	223
B 3.10-37	04/27/12	120
B 3.10-38	04/27/12	120

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of ECCS initiation setpoints higher than the SL provides margin to the SL but is independent of the SL.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and Abnormal Operational Transients. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 686 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 43% RTP. Thus, a THERMAL POWER limit of 21.7% RTP for reactor pressure < 686 psig is conservative.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which transition boiling is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid transition boiling, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

For SLO, the SLMCPR is greater to account for the increased uncertainties.

(continued)

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Surveillance may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level — Low Low Low

Low Reactor Pressure Vessel (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 the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level — Low Low Low Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level — Low Low Low Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Reactor Vessel Water Level - Low Low Low supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level 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. Four channels of Reactor Vessel Water Level — Low Low 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 Low Low Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Low Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential Loss of Coolant Accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure — Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure — Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 686 psig, which results in a scram due to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e and 1.f. Area Temperature — High

Area temperature is provided to detect a leak in the RCPB and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the UFSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Area temperature signals are initiated from Resistance Temperature Detectors (RTDs) located in the area being monitored. Sixteen channels of Main Steam Tunnel Temperature — High Function are available and 8 channels (2 per main steam line) are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function for a break of the size for which protection is necessary. Eight channels of Turbine Building Area Temperature — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each channel consists of a RTD and its corresponding temperature monitor.

The Area Temperature-High Allowable Value is set far enough above the temperature expected during operations at rated power to avoid spurious isolation, yet low enough to provide early indication of a steamline break.

These Functions isolate the Group 1 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.d., 4.d. Drywell Pressure-High (continued)

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Two channels of Drywell Pressure — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure — High Allowable Value (LCO 3.3.5.1), since this is indicative of a LOCA inside primary containment.

This Function isolates the Group 9 valves.

3.e., 3.g., 3.h., 3.i., 4.e., 4.g., 4.h., 4.i. Area and Differential Temperature — High

Area and differential temperatures are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Functions are not assumed in any UFSAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

Area and Ambient Temperature — High signals are initiated from temperature elements that are located in two areas for HPCI and in two areas for RCIC to protect each system. Two instruments monitor each area. Two channels for each HPCI and RCIC Area, and Suppression Pool Area Ambient Temperature — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

Eight temperature elements provide input to the HPCI and Suppression Pool Area Ventilation Differential Temperature - High Functions and eight temperature elements provide input to the RCIC and Suppression Pool Area Ventilation Differential Temperature - High Functions. The output of these temperature elements is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from two temperature elements for a total of four available channels (two for RCIC and two for HPCI). The HPCI Room Differential Temperature Channel temperature elements are located in the inlet and outlet of the area ventilation system. The RCIC Room Differential Temperature Channel inlet

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.e., 3.g., 3.h., 3.i., 4.e., 4.g., 4.h., 4.i. Area and Differential Temperature — High (continued)

temperature elements are located in the HPCI/RCIC room hallway. The RCIC exhaust temperature elements are located in the outlet of the area ventilation system.

The Allowable Values are set low enough to detect a leak equivalent to 5 gpm. These Functions isolate the Group 6A and 6B valves, as appropriate.

3.f., and 4.f. HPCI and RCIC Leak Detection Time Delay

The HPCI Leak Detection and RCIC Leak Detection Time Delay Functions are provided to allow all the other systems that may be leaking into the pool area (as indicated by the high temperature) to be isolated before HPCI and/or RCIC are automatically isolated. This ensures maximum HPCI and RCIC System availability by preventing isolations due to leaks in other systems. These Functions are not assumed in any UFSAR transient or accident analysis.

There are four time delay relays (two for HPCI and two for RCIC). Two channels each for both HPCI and RCIC Leakage Detection Time Delay Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

No Allowable Values are specified in Table 3.3.6.1-1 for these Functions, as Analytical Limits are not established for this variable, only Nominal Trip Setpoints (NTSPs), which are controlled by plant procedures. The NTSPs are based on ensuring that the HPCI and RCIC systems are isolated in a sufficiently short time after a steam leak is detected so as to ensure that any potential release from such a leak remains bounded by the larger releases of the Design Basis Accidents (DBAs). In addition, the HPCI and RCIC time delays are staggered to allow the operator to determine which of these two systems is the source of leakage, allowing it to isolate, while maintaining the other system for core cooling. The time delay for the HPCI isolation is set shorter than that for RCIC as it has the larger steam line and hence the larger reactor vessel inventory loss if broken. Margin has been added to the NTSPs to account for instrument drift to specify acceptable as-found tolerances. These Functions isolate the Group 6A and 6B valves, as appropriate.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety Relief Valves (SRVs) and Safety Valves (SVs)

BASES

BACKGROUND

The ASME Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of SRVs and SVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the Reactor Coolant Pressure Boundary (RCPB).

The SRVs and SVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs can actuate by either of two modes: the safety mode or the relief mode. However, for the purpose of this LCO, only the safety mode is required. The SVs actuate only in the safety mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. The safety mode function of both SRVs and SVs satisfies the Code requirement. A power generation design basis function of the SRVs is also to prevent opening of the SVs during normal plant isolations and load rejections.

Each SRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool while the SVs discharge directly to the drywell airspace. The SRVs that provide the relief mode are the Low-Low Set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.5, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS — Operating."

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all Main Steam Isolation Valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 6 valves (any combination of SRVs and SVs) are assumed to operate in the

(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)

BASES

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 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Reference 7) contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. The heatup curve provides limits for both heatup and criticality.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the Reactor Coolant Pressure Boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

(continued)

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T Limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The elements of this LCO are:

a. RCS pressure, temperature and heatup or cooldown rates are within the limits specified in the PTLR;

b. The temperature difference between the reactor vessel bottom head coolant and the Reactor Pressure Vessel (RPV) coolant is within the limits specified in the PTLR during recirculation pump startup;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is within the limits specified in the PTLR during recirculation pump startup;

d. RCS pressure and temperature are within the criticality limits specified in the PTLR, prior to achieving criticality; and

e. The temperatures at the reactor vessel head flange and the shell adjacent to the head flange are within the limits specified in the PTLR when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

(continued)

BASES

LCO (continued)

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

<u>A.1 and A.2</u>

Operation outside the P/T limits in the PTLR while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

Verification that operation is within limits is required periodically when RCS pressure and temperature conditions are undergoing planned changes. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, this Frequency permits a reasonable time for assessment and correction of minor deviations.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be initiated and discontinued when the criteria given in the relevant plant procedure for starting and ending the activity are satisfied. During heatups and cooldowns, the temperatures at the reactor vessel shell adjacent to the shell flange, the reactor vessel bottom drain, recirculation loops A and B, and the reactor vessel bottom head shall be monitored. During inservice hydrostatic or leak testing, the reactor vessel metal temperatures at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to the shell flange shall be monitored.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation pump (Ref. 8) are satisfied.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3 and SR 3.4.9.4 (continued)

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

For SR 3.4.9.3, an acceptable means of measuring Reactor Pressure Vessel (RPV) coolant temperature is by using the saturation temperature corresponding to reactor steam dome pressure.

Acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 include but are not limited to comparing the temperatures of the operating recirculation loop and the idle loop. The idle loop and RPV coolant temperature using saturation temperature corresponding to reactor steam dome pressure, or the idle loop and the bottom head coolant temperature with flow through the bottom head drain are acceptable means.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4 during a recirculation pump startup, since this is when the stresses occur. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required.

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7

Limits on temperature at the reactor vessel head flange and the shell adjacent to the head flange are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

SR 3.4.9.5 requires that temperatures at the reactor vessel head flange and the shell adjacent to the head flange must be verified to be above the limits within the Surveillance Frequency before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix G, December 1995
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR), see Appendix A of the Technical Requirements Manual.
8. UFSAR, Section 15.1.5.1.

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)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification that a 100 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 100 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 100 day supply of nitrogen exists for each accumulator. The results of this analysis can also be used to determine when the 100 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 3.5.2.4 (continued)

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 one 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 low pressure and low temperature conditions in Modes 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core coverage prior to postulated fuel uncovery. This will ensure adequate core cooling if an inadvertent RPV draindown should occur.

REFERENCES

1. UFSAR, Section 15.2.1.1.

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)

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

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 (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 ≥ 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

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.3 (continued)

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.

DAEC's nine-month response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat removal, and Containment Spray Systems," stated that RHR Suppression Pool Cooling System piping was not susceptible to gas intrusion when in standby readiness. Due to the lack of susceptibility to gas intrusion, SR 3.6.2.3.3 is not performed on a regularly scheduled basis. However, DAEC's response also stated that, based on DAEC specific operating experience, one of the primary mechanisms for potential gas intrusion in the Generic Letter subject systems/functions is incomplete post-maintenance fill and vent activities. Consequently, SR 3.6.2.3.3 will be performed following post maintenance fill and vent activities for RHR Suppression Pool Cooling piping to demonstrate the system is sufficiently fill with water to ensure RHR Suppression Pool Cooling operability.

REFERENCES

1. UFSAR, Section 15.2.1.1.
2. UFSAR, Section 6.2.2.2.1
3. ASME Code for Operation and Maintenance of Nuclear Power Plants
4. Letter, R. Anderson (FPL Energy Duane Arnold) to NRC, Nine-Month Response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," NG-08-0777, October 13, 2008.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.2 (continued)

rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR Suppression Pool Spray 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.

DAEC's nine-month response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat removal, and Containment Spray Systems," stated that RHR Suppression Pool Cooling Spray piping was not susceptible to gas intrusion when in standby readiness. Due to the lack of susceptibility to gas intrusion, SR 3.6.2.4.2 is not performed on a regularly scheduled basis. However, DAEC's response also stated that, based on DAEC specific operating experience, one of the primary mechanisms for potential gas intrusion in the Generic Letter subject systems/functions is incomplete post-maintenance fill and vent activities. Consequently, SR 3.6.2.4.2 will be performed following post maintenance fill and vent activities for RHR Suppression Pool Spray piping to demonstrate the system is sufficiently fill with water to ensure RHR Suppression Pool Spray operability.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 15.2.1.

2. UFSAR, Section 6.2.2.3

3. NG-98-0342, J. Franz (IES) to U.S. NRC, "Request for Technical Specification Change (RTS-291): Revision E to the Duane Arnold Energy Center Improved Technical Specifications," February 26, 1998.

4. Letter, R. Anderson (FPL Energy Duane Arnold) to NRC, Nine-Month Response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," NG-08-0777, October 13, 2008.

B 3.7 PLANT SYSTEMS

B 3.7.3 Emergency Service Water (ESW) System

BASES

BACKGROUND

The ESW System is designed to provide cooling water for the removal of heat from equipment, such as the Diesel Generators (DGs), room coolers for Emergency Core Cooling System equipment, Control Building Chillers, and various minor heat loads required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The ESW System also provides cooling to unit components, as desired during normal operation. Upon receipt of a start signal from its associated DG, each ESW pump is automatically started, i.e., the ESW start circuitry is dependent upon the DG control system.

The ESW System consists of two independent and redundant subsystems. Each of the two ESW subsystems is made up of a header, one 1200 gpm pump, a suction source, valves, piping and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity to support the required systems. The two subsystems are separated from each other so failure of one subsystem will not affect the OPERABILITY of the other subsystem. The capability exits to manually cross connect the ESW subsystems using a removable spool piece, however, one ESW pump cannot be used to supply both ESW loops.

Cooling water is pumped from the RHRSW/ESW pump pit in the pump house by the ESW pumps to the essential components through the two main headers. After removing heat from the components, the water is either discharged from the DG directly to a storm sewer, or is combined with water discharged from the RHRSW System and Well Water System and discharged to the Circulating Water System to replace evaporation losses from the cooling towers, or directly to the river via the dilution structure or deicing. A complete description of the ESW System is presented in the UFSAR, Section 9.2.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ability of the ESW System to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the UFSAR, Chapter 15 (Ref. 2). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. It is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for manual fuel transfer systems are OPERABLE. Additional assurance of fuel oil transfer pump OPERABILITY is provided by meeting the testing requirements for pumps that are contained in the ASME Code (Ref. 13). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

The slow transfer of each 4.16 kV essential bus power supply from the preferred offsite circuit (i.e. - the startup transformer) to the alternate preferred offsite circuit (i.e. the standby transformer) demonstrates the OPERABILITY of the alternate preferred circuit distribution network to power the shutdown loads. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of the Surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed on this Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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

REFERENCES (continued)

6. Regulatory Guide 1.93.
7. Generic Letter 84-15.
8. UFSAR, Section 3.1.2.2.9
9. Regulatory Guide 1.108.
10. Regulatory Guide 1.137.
11. [Deleted]
12. UFSAR, Section 15.2.1
13. ASME Code for Operation and Maintenance of Nuclear Power Plants.
14. IEEE Standard 308.
15. [Deleted]
16. UFSAR, Table 8.3-1.
17. Regulatory Guide 1.9.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

The Diesel Generators (DGs) are provided with a storage tank having a fuel oil capacity sufficient to operate one DG for a period of 7 days while the DG is supplying maximum post Loss of Coolant Accident (LOCA) load demand discussed in UFSAR, Section 9.5.4 (Ref. 1) and Regulatory Guide 1.137 (Ref. 2). The maximum load demand is calculated using the assumption that only one DG is available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to the day tanks by either of two transfer pumps. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe or valve, to result in the loss of more than one DG. The outside tanks, pumps, and piping are located underground.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity (or Saybolt Universal Viscosity), specific gravity (or API gravity), and total particulate contamination.

The DG Lubrication System is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. The DG Lube Oil System is designed to provide automatic lube oil makeup to the DG crankcase for a minimum of 7 days of operation per UFSAR Section 8.3.1 (Ref. 7). This supply is sufficient to allow the operator to replenish lube oil from outside sources.

Each DG has two independent starting air supply systems each with adequate capacity for five successive normal start attempts per air receiver without recharging. A minimum of fifteen normal DG starts are provided for each DG per UFSAR

(continued)

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 31,238 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

ACTIONS

A.1 (continued)

be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

In this Conditions, the 7 day lube oil inventory, i.e., sufficient lube oil to support 7 days of continuous DG operation at full load conditions is not available. However, the Condition is restricted to lube oil level reductions that maintain at least a 6 day supply. The lube oil inventory equivalent to a 6 day supply is 221 gallons. This restriction allows sufficient time for obtaining the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion for particulates. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, since particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and since proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a period prior to declaring both DGs inoperable. The 30 day Completion Time starts when initial analysis results are known and allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that

(continued)

ACTIONS

D.1 (continued)

the mixture of new fuel oil and previously stored fuel oil remains acceptable, or to restore the stored fuel oil properties. If testing of the new fuel oil reveals fuel oil properties are not within limits, those properties not within limits must be tested, or restored to within limits. This restoration may involve feed and bleed procedures, filtering, or combination of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is high likelihood that the DG would still be capable of performing its intended function. Note that when the entire inventory of stored fuel oil is replaced with new fuel oil, the new fuel oil properties must be within specification prior to declaring the DGs OPERABLE.

E.1

With all starting air receivers associated with a DG with pressure < 150 psig, sufficient capacity for five successive DG start attempts may not exist. However, as long as any one receiver's pressure is ≥ 75 psig, there is adequate capacity for at least one start attempt (as shown during initial plant startup testing when successful DG starting was demonstrated with air receiver pressure as low as 50 psig), and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or the stored diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(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 36,317 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)