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U.S. Nuclear Regulatory Commission
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Subject: River Bend Station - Unit 1
Docket No. 50-458
License No. NPF-47
Revisions to the Technical Requirements Manual and the Technical Specifications Bases

File Nos.: G9.5, G9.25.1.5, G9.41.1

RBG-46179
RBF1-03-0180

Gentlemen:

Pursuant to 10CFR50.71(e), Entergy Operations, Inc. (EOI) herein submits changes to the River Bend Station (RBS) Technical Requirements Manual (TRM). The revised pages cover changes made during the period from April 1, 2002, through October 1, 2003. This includes TRM revisions 77 through 90. A list of effective pages is included to identify the current pages of the TRM through revision 90.

Pursuant to RBS Technical Specification 5.5.11, revised pages for the Technical Specification Bases pages are included. The revised pages cover changes made from April 1, 2002, through October 1, 2003. This includes Bases revisions 103 through 116. A list of effective pages is included to identify the current pages of the Bases through Revision 116.

As required by 10CFR50.71(e)(2)(i), the below affirmation certifies that the information in this submittal accurately reflects changes made since the previous submittal, necessary to represent information and analyses submitted or prepared pursuant to NRC requirements.

A001

Should you have any questions, please advise Mr. J. W. Leavines at (225) 381-4642.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 1, 2003.

R. J. King

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RJK/DNL

enclosures: 1) Technical Requirements Manual Revision Package
2) Technical Specification Bases Revision Package

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Enclosure 1 to RBG-46179

River Bend Station

Technical Requirements Manual Revision Package

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1.0 USE AND APPLICATION

1.01 Technical Requirements Manual(TRM) Description

The Technical Requirements Manual (TRM) provides those limitations upon plant operations which are part of the licensing basis for the station but do not meet the criteria for continued inclusion in the Technical Specifications.

It also provides information which supplements the Technical Specifications such as specific plant setpoints for Technical Specification equipments. Nothing in the TRM shall supersede any Technical Specification requirement.

1.02 TRM Use

The Technical Specification rules for Use and Application of section 1.0 apply to the TRM. TRM LCOs and SRs are designated as TLCO and TSR.

TRM sections with no associated TLCO apply to the Technical Specification of the same number.

1.03 Document Control

The TRM is considered a licensing basis document and as such, overall control of the document shall be in accordance with site procedures for document control. Distribution of the TRM is controlled by the River Bend Station Licensing Department. Licensing specifies the proper distribution for the TRM which includes those personnel/locations which receive the Technical Specifications as well as any other groups which need access to the information contained in the TRM. Changes to the TRM will be issued on a replacement page basis to controlled document holders following approval of the change in accordance with site procedures on document control.

Revision 5 is the initial issue of the Improved Technical Specification TRM replacing the existing TRM Revision 0 through 3 in its entirety. Future revised pages will be reflected in the List of Effective Pages.

1.04 Changes to the TRM

Changes made at River Bend Station have the potential to affect (or be affected by) the TRM. These include items such as design modifications, procedure changes, other licensing document changes, etc. Changes to the TRM shall be controlled by procedure. Changes to the TRM shall be evaluated per the 10CFR50.59 program. This program requires that the TRM be considered in a manner similar to the USAR when screening changes to determine if a license amendment is required.

Changes to the TRM will be reported to the NRC annually as part of the USAR update. Related safety evaluations will be reported as part of the 10CFR50.59 annual report. Proposed TRM changes that are determined to require a license amendment (as defined by 10CFR50.59(c)(2)) will either not be made or will be submitted to the NRC for prior review and approval.

Where additional reviews and approvals are required by regulations to effect a change in a TRM requirement, such as Fire Protection Program changes, those reviews and approvals shall also be completed as required to implement a change to the TRM.

1.1 Definitions

---------------------------------NOTE---------------------------------
The defined terms of this section and Technical Specification Section 1.1 appear in capitalized type and are applicable throughout these Technical Requirements.

Term	Definition
AVERAGE PLANAR EXPOSURE	The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.
GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM	The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.
LOWER LIMIT OF DETECTBILITY (LLD)	The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD for radioactive effluents is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

LLD for environmental samples is the "a priori" lower limit of detection as defined above, then multiplied by 10^6, to yield picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample, as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

(continued)

1.1 Definitions (continued)

	(LLD continued)
	V is the sample size in units of mass or volume,
	2.22×10^6 is the number of disintegrations per minute per microcurie,
	Y is the fractional radiochemical yield, when applicable,
	λ is the radioactive decay constant for the particular radionuclide, and
	Δt for plant effluents is the elapsed time between the midpoint of sample collection and the time of counting.
	Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting.
	Typical values of E, V, Y, and Δt should be used in the calculation.
	It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.
MEMBER(S) OF THE PUBLIC	MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.
OFFSITE DOSE CALCULATION MANUAL (ODCM)	The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints. It shall also contain a table and figure defining current radiological environmental monitoring sample locations.

(continued)

1.1 Definitions (continued)

Operations With a Potential for Draining the Reactor Vessel (OPDRVs)	An OPDRV consists of those operations or maintenance that: a. have the potential to uncover irradiated fuel in the reactor pressure vessel or for Operations With the Potential to Drain the Reactor Cavity (OPDRCs), containment fuel storage pool, and b. involve vessel penetrations or piping greater than 1-1/4 inches which penetrate the RPV below the LPCI nozzles (not to include the LPCI nozzles). Work on multiple lines with a total equivalent diameter exceeding 1-1/4 inches are included, and c. an acceptable barrier or otherwise addressed measures are not in place to provide reasonable assurance that an error in the maintenance or operations activity will not cause drainage of the reactor pressure vessel.
REPORTABLE EVENT	A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.
SITE BOUNDARY	The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.
SOLIDIFICATION	SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.
SOURCE CHECK	A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.
UNRESTRICTED AREA	An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.
VENTILATION EXHAUST TREATMENT SYSTEM	A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

TR 3.0 APPLICABILITY

3.0 LIMITING CONDITION FOR OPERATION (TLCO) APPLICABILITY

TLCO 3.0.1 TLCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in TLCO 3.0.2.

TLCO 3.0.2 Upon discovery of a failure to meet a TLCO, the Required Actions of the associated Conditions shall be met, except as provided in TLCO 3.0.5 and TLCO 3.0.6.

If the TLCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

TLCO 3.0.3 When a Technical Requirements Manual TLCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the following actions shall be taken:

1. Implement appropriate compensatory actions as needed.
2. Verify that a required safety function is not compromised by the inoperabilities.
3. Within 7 hours, obtain duty manager approval of the compensatory actions and a plan for exiting TLCO 3.0.3.

Where corrective measures are completed that permit operation in accordance with the TLCO or ACTIONS, completion of the actions required by TLCO 3.0.3 is not required.

TLCO 3.0.3 is always applicable to Technical Requirements Manual TLCOs.

Actions to exit TLCO 3.0.3 should be pursued without delay and in a controlled manner.

TLCO 3.0.4 When a TLCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS, or that are part of a shutdown of the unit.

(continued)

TR 3.1.5 Control Rod Scram Accumulators

------------------------------------NOTE------------------------------------
The following surveillance requirement applies to Technical Specification LCO 3.1.5. Failure to meet this surveillance requirement requires entry into Technical Specification LCO 3.1.5.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.5.1	(Not Used)	
TSR 3.1.5.2	Measure and record the time, for up to 10 minutes, that each individual accumulator check valve maintains the associated accumulator pressure above the alarm set point.	18 months

TR 3.1.5.1 Control Rod Scram Accumulator Detectors/alarm Instrumentation

TLCO 3.1.5.1 Each control rod scram accumulator alarm shall be OPERABLE.

APPLICABILITY: When associated control rod scram accumulator is OPERABLE per Technical Specification LCO 3.1.5.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod scram accumulator detector/alarm.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more accumulator pressure detectors or alarms inoperable.	A.1 Verify the affected accumulator pressure ≥ 1540 psig.	Once per 24 hours
B. One or more accumulator leak detectors or alarms inoperable.	B.1 Verify the affected accumulator water drained.	Once per 48 hours <u>AND</u> Within 24 hours prior to reactor startup
C. Required Action and associated Completion Time not met.	C.1 Declare the associated accumulator inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.5.1.1	Perform a CHANNEL FUNCTIONAL TEST on the leak detector and associated alarm for each control rod scram accumulator.	18 months
TSR 3.1.5.1.2	Perform a CHANNEL CALIBRATION of the pressure detector for each control rod scram accumulator and verify a nominal alarm setpoint of 1600 psig on decreasing pressure.	18 months

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	NOMINAL SETPOINT/ RESPONSE TIME
8. Scram Discharge Volume Water Level - High					
a. Transmitter/Trip Unit	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	49 inches
	5(a)	2	I	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	49 inches
b. Float Switch	1,2	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	47.32 inches for LSN013A, B 45.44 inches for LSN013C, D
	5(a)	2	I	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	47.32 inches for LSN013A, B 45.44 inches for LSN013C, D
9. Turbine Stop Valve Closure	≥ 40% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 TSR 3.3.1.1.16 SR 3.3.1.1.18	5% closed (g) ≤ 0.06 sec
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 40% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 TSR 3.3.1.1.16 SR 3.3.1.1.18	530 psig (g) ≤ 0.07 sec(f)
11. Reactor Mode Switch - Shutdown Position	1,2	2	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
	5(a)	2	I	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
12. Manual Scram	1,2	2	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	2	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b), (c), (d), (e), (h) not used this page
(f) Measured from start of turbine control valve fast closure.
(g) The Turbine First Stage Pressure nominal setpoint is 188.2 psig with an Allowable value ≤ 199.4 psig.

Table 3.3.2.1-1 (Page 2 of 2)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP FUNCTION	CONDITIONS REFERENCED FROM TLCO REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	NOMINAL SETPOINT	ALLOWABLE VALUE
6. Source Range Monitors						
a. Detector not full in(e)	2	3	B	TSR 3.3.2.1.12 TSR 3.3.2.1.16	NA	NA
	5*	2**	C	TSR 3.3.2.1.12 TSR 3.3.2.1.16	NA	NA
b. Upscale(f)	2	3	B	TSR 3.3.2.1.12 TSR 3.3.2.1.16	1 x 10^5cps	≤ 1.6 x 10^5cps
	5*	2**	C	TSR 3.3.2.1.12 TSR 3.3.2.1.16	1 x 10^5cps	≤ 1.6 x 10^5cps
c. Inoperative(f)	2	3	B	TSR 3.3.2.1.12	NA	NA
	5*	2**	C	TSR 3.3.2.1.12	NA	NA
d. Downscale(g)	2	3	B	TSR 3.3.2.1.12 TSR 3.3.2.1.16	≥ 0.7 cps(i)	≥ 0.5 cps(i)
	5*	2**	C	TSR 3.3.2.1.12 TSR 3.3.2.1.16	≥ 0.7 cps(i)	≥ 0.5 cps(i)
7. Intermediate Range Monitors						
a. Detector not full in	2, 5*	6	B	TSR 3.3.2.1.12	NA	NA
b. Upscale	2, 5*	6	B	TSR 3.3.2.1.12 TSR 3.3.2.1.16	108/125 division of full scale	≤ 110/125 division of full scale
c. Inoperative	2, 5*	6	B	TSR 3.3.2.1.12	NA	NA
d. Downscale(h)	2, 5*	6	B	TSR 3.3.2.1.12 TSR 3.3.2.1.16	5/125 division of full scale	≥ 3/125 division of full scale
8. Scram Discharge Volume Water level-high	1, 2, 5*	2	D	TSR 3.3.2.1.13 TSR 3.3.2.1.15 TSR 3.3.2.1.17		
a. LISN602A					18.00"	≤ 21.12"
b. LISN602B					18.00"	≤ 21.60"
9. Reactor Coolant System Recirculation Flow Upscale	1	2	D	TSR 3.3.2.1.13 TSR 3.3.2.1.14 TSR 3.3.2.1.16 TSR 3.3.2.1.18	114% of rated flow	≤ 117% of rated flow

* With any control rod withdrawn. Not applicable to control rods removed per Technical Specification LCO 3.10.5 or 3.10.6.

** OPERABLE channels must be associated with SRM required OPERABLE per Technical Specification LCO 3.3.1.2.

(a) THERMAL POWER > HPSP.

(b) THERMAL POWER > 35% RTP and ≤ HPSP.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) This function is not required if detector count rate is ≥ 100 cps or the IRM channels are on range 3 or higher.

(f) This function is not required when the associated IRM channels are on range 8 or higher.

(g) This function is not required when the IRM channels are on range 3 or higher.

(h) This function is not required when the IRM channels are on range 1.

(i) Provided the Signal to noise ratio is ≥ 2.0, otherwise trip setpoint of ≥ 3.0 cps and allowable ≥ 1.8 cps.

(j) Allowable Values and Nominal Values specified in COLR. Allowable and nominal value modifications required by the COLR due to reduction in feedwater temperature may be delayed for up to 12 hours. The trip setting for this Function must be maintained in accordance with TLCO 3.2.4.

(k) To address feedwater temperature reductions, set at first stage turbine pressure equivalent to 60.0 % RTP.

TR 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

-----NOTE-----
The following surveillance requirement applies to Technical Specification LCO 3.3.4.1. Failure to meet this surveillance requirement requires entry into Technical Specification LCO 3.3.4.1.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.4.1.1 - 3.3.4.1.8 (Not Used)		
TSR 3.3.4.1.9	Verify that all bypass valves are closed when ≥ 40% RTP with any recirc pump in fast speed.	12 hours

TABLE 3.3.4.1-1

END OF CYCLE RECIRCULATION PUMP TRIP INSTRUMENTATION

FUNCTION	NOMINAL SETPOINT	ALLOWABLE VALUE	RESPONSE TIME
a. Turbine Stop Valve Closure	5% closed	≤ 7% closed	≤ 140 milliseconds
b. Turbine Control Valve Fast closure *	530 psig	≥ 465 psig	≤ 140 milliseconds

* Automatic bypass Turbine First Stage Pressure nominal setpoint is 188.2 psig with an allowable value of ≤ 199.4 psig.

TR 3.3.5.1.1 Emergency Core Cooling System (ECCS) ADS Inhibit Instrumentation

TLCO 3.3.5.1.1 Automatic Depressurization System (ADS) Trip System A and B, Manual Inhibit Function shall be OPERABLE.

APPLICABILITY: MODE 1,
MODE 2 and 3, with reactor steam dome pressure > 100 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more manual inhibit channel(s) inoperable.	A.1 Enter TLCO 3.0.3.	24 hours

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	NOMINAL SETPOINT/ RESPONSE TIME
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	-143 inches $T_L \leq 1.0$ (g)
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	849 psig $T_L \leq 1.0$ (g)
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	185 psid, Line A 189 psid, Line B, C and D $T_L \leq 0.5$ (g)
d. Condenser Vacuum - Low	1,2(a), 3(a)	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	8.5 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	141°F
f. Deleted					
g. Deleted					

(continued)

(a) With any turbine stop valve not closed.
(b), (c), (d), (e), (f) - Not used this page.
(g) $T_L = T_X + T_C$; where:
T_L = Measured total response time of the isolation system instrumentation
T_X = Hydraulic response time of the channel sensor measured upon initial installation
T_C = Measured response time of the logic circuit excluding the channel sensor

The given numerical value is the acceptance criterion for T_L. Isolation system instrumentation response time for MSIVs only; no diesel generator delays are assumed.

T_L shall be added to the 5-second isolation time shown in Table 3.6.1.3-1 for the MSIVs to obtain ISOLATION SYSTEM RESPONSE TIME for the MSIVs.

In case the sensor is replaced or refurbished, a hydraulic response time test must be performed to establish revised value for T_X.

Table 3.3.6.1-1 (page 2 of 5)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	NOMINAL SETPOINT
1. Main Steam Line Isolation (continued)					
h. Deleted					
i. Deleted					
j. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.6	NA
k. DELETED					
l. Main Steam Line Radiation - High-High	1,2,3	1(f)	TLCO L	TSR 3.3.6.1.1 TSR 3.3.6.1.2 TSR 3.3.6.1.5 TSR 3.3.6.1.6	3.0 x full power background (Allowable Value ≤ 3.6 x full power background) (h)
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2(b)	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	-43 inches
b. Drywell Pressure - High	1,2,3	2(b)	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	1.68 psid
c. Containment Purge Isolation Radiation - High	1,2,3	1	K	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	1.17 R/hr
d. Manual Initiation	1,2,3	2(b)	G	SR 3.3.6.1.6	NA

(continued)

(b) Also required to initiate the associated drywell isolation function.
(f) Only trips and isolates mechanical vacuum pumps, reactor sample valves, and provides monitoring/alarm.
(h) Setpoints to be verified :
1) Within 30 days after a significant change in hydrogen injection, or
2) During Mode 1 or 2 with a mechanical vacuum pump in operation.

TABLE 3.3.6.1-2 (page 1 of 3)
Primary Containment and Drywell Isolation Instrumentation

TRIP FUNCTION			VALVE GROUP OPERATED BY SIGNAL *
1.	MAIN STEAM LINE ISOLATION		
	a.	Reactor Vessel Water Level- Low Low Low Level 1	6
	b.	Main Steam Line Pressure - Low	6
	c.	Main Steam Line Flow - High	6
	d.	Condenser Vacuum - Low	6
	e.	Main Steam Tunnel Temperature - High	6
	f.	DELETED	
	g.	DELETED	
	h.	DELETED	
	i.	DELETED	
	j.	Manual Initiation	6
	k.	DELETED	
	l.	Main Steam Line Radiation - High	9, (a)
2.	PRIMARY CONTAINMENT ISOLATION		
	a.	Reactor Vessel Water Level- Low Low Level 2	1, 7, 8, 9 15, 16
	b.	Drywell Pressure - High	1, 3, 8
	c.	Containment Purge Isolation Radiation - High	8
	d.	Manual Initiation	1, 7, 8, 15, 16

(continued)

(a) Trip and isolate the mechanical vacuum pumps and provide alarm/indication.

TR 3.3.7.1 Control Room Fresh Air (CRFA) System Instrumentation

Table 3.3.7.1-1
Control Room Fresh Air System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	NOMINAL SETPOINT
1.	Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	-43 inches
2.	Drywell Pressure - High	1,2,3	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	1.68 psid
3.	Control Room Ventilation Radiation Monitors (providing initiation)	1,2,3 (a), (b)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	0.84×10^{-5} µCi/cc

(a) During operations with a potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

TLCO 3.3.7.1 The Control room ventilation remote intake radiation monitor alarm function shall be OPERABLE with its setpoint $\leq 0.97 \times 10^{-5}$ µCi/cc

APPLICABILITY: MODES 1, 2, 3
During movement of recently irradiated fuel assemblies in the primary containment or fuel building

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required alarm channel inoperable.	A.1 Perform area surveys of the monitored area with portable monitoring instrumentation.	once per 24 hours

TR 3.3.7.3 Feedwater/Main Turbine Level 8 Trip Instrumentation

TLCO 3.3.7.3 The feedwater/main turbine trip Function shall have 3 channels of reactor vessel water level - high, level 8, instrumentation OPERABLE.

APPLICABILITY: Mode 1

ACTIONS

-----------------------------------NOTE---------------------------------------
Separate Condition entry is allowed for each Channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channel(s) inoperable.	A.1 Restore channel(s) to OPERABLE status.	7 days
B. The feedwater/main turbine trip Function not maintained.	B.1 Restore the feedwater/main turbine trip capability.	72 hours
C. Required Action and associated Completion Time not met.	C.1 Enter TLCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---
When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided that the trip Function capability is maintained.

SURVEILLANCE		FREQUENCY
TSR 3.3.7.3.1	Perform a CHANNEL CHECK.	24 hours
TSR 3.3.7.3.2	Perform a CHANNEL FUNCTIONAL TEST.	92 days
TSR 3.3.7.3.3	Perform a CHANNEL CALIBRATION. The Allowable Value shall be ≤ 52.5 inches. The Nominal Setpoint is 50.7 inches.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.7.4.1	---------------------NOTE-------------------- Only required during cold shutdowns exceeding 24 hours for instruments not accessible during unit operation --- Perform a CHANNEL FUNCTIONAL TEST	24 months
TSR 3.3.7.4.2	The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each required fire detection instruments shall be demonstrated OPERABLE.	24 months

TR 3.3.7.5 Seismic Monitoring Instrumentation

TLCO 3.3.7.5 The seismic monitoring instrumentation shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

-------------------------------NOTE-------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required seismic monitoring instruments inoperable.	A.1 Restore required seismic monitoring instruments to OPERABLE.	30 days
B. Required Action A.1 and associated Completion Time not met.	B.1 Initiate action to prepare an appropriate deficiency document.	Immediately

TR 3.3.7.8.2 Offgas System Radiation Monitoring Instrumentation

TLCO 3.3.7.8.2 The Offgas System Radiation Monitoring Instrumentation shown in Table T3.3.7.8.2-1 shall be OPERABLE with its alarm/trip setpoints within the specified limits.

APPLICABILITY: During operation of the main condenser air ejector

ACTIONS

---NOTE---

1. Separate Condition entry is allowed for each channel.

2. The provisions of Technical Requirement TLCO 3.0.4 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. A radiation monitoring instrumentation channel alarm/trip setpoint exceeding the limit.	A.1 Adjust the setpoint to within the limit.	4 hours
	<u>OR</u>	
	A.2 Declare the channel inoperable.	4 hours
B. One or more required radiation monitoring channels inoperable.	B.1 Enter the Condition Referenced in Table T3.3.7.8.2-1 for the channel.	Immediately
C. As required by Required Action B.1 and referenced in Table T3.3.7.8.2-1.	C.1 Obtain a grab sample of the monitored parameter	Once per 12 hours
	<u>AND</u>	
	C.2 Analyze the sample for gross radioactivity and verify effluent radioactivity is below trip setpoint.	Within 24 hours of sample collection
	<u>AND</u>	
	C.3 Restore required instrumentation to OPERABLE status.	30 days

(continued)

TABLE 3.3.7.8.2-1

OFFGAS AND RADIATION MONITORING INSTRUMENTATION

INSTRUMENTS	MINIMUM INSTRUMENTS OPERABLE	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	NOMINAL SETPOINT	ALLOWABLE UPPER LIMIT
1. Main Condenser Offgas Post-Treatment System Effluent Monitoring System					
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release)	1 (b)	C	TSR 3.3.7.8.2.1 TSR 3.3.7.8.2.2 TSR 3.3.7.8.2.3	4.16×10^5 cpm	4.99×10^5 cpm
2. Condenser Air Ejector Pretreatment Radioactivity Monitor					
a. Noble Gas Activity Monitor	1	D	TSR 3.3.7.8.2.1 TSR 3.3.7.8.2.2 TSR 3.3.7.8.2.3	1.5 X Full Power Process Background Radiation Level (alarm only)	(a)

(a) The nominal setpoint of 1.5 times the full power process background radiation level shall not exceed a value corresponding to the Technical Specification LCO 3.7.4 allowable release rate.

(b) For Item 1.a., the monitoring system is provided with two detector channels. One detector may be in an inoperable, tripped condition, or placed in "inop" and the second detector remains capable of initiating the logic to isolate the system. However, a single inoperable detector may not be bypassed (e.g. jumpered out) without entering condition C.

TR 3.3.8.1 Loss of Power (LOP) Instrumentation

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER DIVISION	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Divisions 1 and 2 - 4.16 kV Emergency Bus Undervoltage			
a. Loss of Voltage - 4.16 kV basis	3	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2910 V and ≤ 3030 V
b. Loss of Voltage - Time Delay	3	SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2.7 seconds and ≤ 3.3 seconds
c. Degraded Voltage - 4.16 kV basis	3	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 3665 V and ≤ 3815 V[a] ≥ 3692 V and ≤ 3733 V[b]
d. Degraded Voltage - Time Delay, No LOCA	3	SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 54 seconds and ≤ 66 seconds
e. Degraded Voltage-Time Delay, LOCA	3	SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2.7 seconds and ≤ 3.3 seconds[a] ≥ 4.56 seconds and ≤ 5.54 seconds[b]
2. Division 3 - 4.16 kV Emergency Bus Undervoltage			
a. Loss of Voltage - 4.16 kV basis	2	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2892 V and ≤ 3198 V
b. Loss of Voltage - Time Delay	2	SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2.7 seconds and ≤ 3.3 seconds
c. Degraded Voltage - 4.16 kV basis	2	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 3747 V and ≤ 3807 V[a] ≥ 3675 V and ≤ 3720 V[b]
d. Degraded Voltage - Time Delay, No LOCA	2	SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 54 seconds and ≤ 66 seconds
e. Degraded Voltage - Time Delay, LOCA	2	SR 3.3.8.1.2 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2.7 seconds and ≤ 3.3 seconds[a] ≥ 4.63 seconds and ≤ 5.57 seconds[b]

(a) Prior to implementation of ER-RB-2001-0360, OPERABILITY shall be maintained using these trip setpoints.

(b) Upon completion of ER-RB-2001-0360 for each division, OPERABILITY shall be restored and maintained using these trip setpoints. These trip setpoints were derived from the Technical Specification Allowable values approved by Amendment 128.

TR 3.3.8.2 RPS Electric Power Monitoring

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

The following surveillance Note applies to Technical Specification SR 3.3.8.2.2.

1) Underfrequency nominal trip setpoint is between 57 HZ and 58.14 HZ.

TR 3.3.12 Meteorological Monitoring Instrumentation

TLCO 3.3.12 The meteorological monitoring instrumentation channels shown in Table 3.3.12-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

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

Separate Condition entry is allowed for each Channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore to OPERABLE	7 days

SURVEILLANCE REQUIREMENTS

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

Refer to Table 3.3.12-1 to determine which TSRs apply to each channel.

SURVEILLANCE		FREQUENCY
TSR 3.3.12.1	Perform CHANNEL CHECK.	24 hours
TSR 3.3.12.2	--------------------NOTE-------------------- Wind Speed and Direction Sensors are excluded. Sensors that have not been placed into service since their last calibration by the manufacturer shall be installed at the specified CHANNEL CALIBRATION frequency to maintain loop integrity. -- Perform CHANNEL CALIBRATION.	184 days

TR 3.3.13 Ultrasonic Feedwater Flow Meters

TLCO 3.3.13 The ultrasonic feedwater flow meters (UFFMs) shall be OPERABLE.

APPLICABILITY: MODE 1 with THERMAL POWER greater than 75% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With one or both UFFMs out of service and THERMAL POWER greater than or equal to 90% RTP.	A.1 Verify all feedwater fouling and temperature correction factors are available.	Immediately AND Every 12 hours
	AND	
	A.2.1 Reduce THERMAL POWER to less than or equal to 3039 MWt. OR	72 hours
	A.2.2 Maintain THERMAL POWER less than 3039 Mwt if correction factors are set equal to 1.0.	Immediately
	AND	
	A.3 Substitute feedwater fouling and temperature correction factors with a value of 1.0. -------------NOTE------------- If THERMAL POWER is reduced to less than 90% RTP while in this Action, immediately enter Action B below. -------------------------------	72 hours
B. With one or both UFFMs out of service and THERMAL POWER less than 90% RTP.	B.1 Maintain THERMAL POWER less than or equal to 3039 Mwt.	Immediately
	AND	
	B.2 Substitute feedwater fouling and temperature correction factors with a value of 1.0.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Actions and Completion Times for Condition A not met.	C.1 Reduce THERMAL POWER to less than 3039 Mwt. AND C.2 Substitute feedwater fouling and temperature correction factors with a value of 1.0.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.13.1	Confirm both UFFMs are OPERABLE.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.1.1.1	Verify volumetric loop flow rate of the loop in operation is ≤ 33000 gpm.	Initially, within 1 hour and once per 12 hours thereafter
TSR 3.4.1.1.2	Verify THERMAL POWER is ≤ 77.6% RTP.	Initially, within 1 hour and once per 12 hours thereafter
TSR 3.4.1.1.3	Verify flow control is in Loop Manual.	Initially, within 1 hour and once per 12 hours thereafter

TR 3.4.11 RCS Pressure and Temperature (P/T) Limits

Note: The pressure-temperature limits given in Technical Specification Figure 3.4.11-1 are limited for use up to 16 EFPY based on the NRC Safety Evaluation Report for Amendments 114 and 120.

Table 3.4.11-1
REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
WITHDRAWAL SCHEDULE

Table Deleted

Table 3.4.11-2
REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
CAPSULE DATA

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR at I.D./¼T
1*	3°	0.67/0.89
2	177°	0.67/0.89
3	183°	0.67/0.89

* Note: Capsule No. 1 was removed from and remained out of vessel during cycle 7. This capsule is designated as the "standby" capsule.

TR 3.5.4 Suppression Pool Pumpback System (SPPS)

TLCO 3.5.4 Two Suppression Pool Pumpback Systems shall be OPERABLE, each consisting of:

a. at least one OPERABLE crescent area sump pump and

b. an OPERABLE flow path to the suppression pool.

APPLICABILITY: MODE 1, 2, and 3
When Suppression Pool level is required to be maintained per SR 3.5.2.1 or SR 3.5.2.2.a for LCO 3.5.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SPPS subsystem inoperable	A.1 Restore to OPERABLE	31 days
B. Both required SPPS subsystems inoperable	B.1 Restore one subsystem to OPERABLE	7 days
C. Required Action and associated Completion Time for Condition A or B not met in MODES 1, 2, or 3.	C.1 Enter TLCO 3.0.3	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition B not met when suppression pool level is being maintained for LCO 3.5.2.	D.1 Establish compliance with LCO 3.6.1.10, Primary Containment - Shutdown.	8 hours
	AND	
	D.2.1.1 Provide an alternate pumpback method	24 hours
	AND	
	D.2.1.2 Demonstrate the OPERABILITY of an alternate pumpback method.	once per 24 hours thereafter
	OR	
	D.2.2.1 Suspend CORE ALTERATIONS	Immediately
	AND	
	D.2.2.2 Suspend operations with a potential for draining the reactor vessel (OPDRVs)	Immediately
	AND	
	D.2.2.3 Lock the reactor mode switch in SHUTDOWN	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.5.4.1 Perform a functional test of each crescent area sump pump to verify it is capable of developing 50 gpm when aligned to the suppression pool.	18 months
TSR 3.5.4.2 Verify the flow path can be aligned to the suppression pool.	18 months

TABLE 3.6.1.3-1 (page 1 of 6)
PRIMARY CONTAINMENT ISOLATION VALVES

SYSTEM	VALVE NUMBER[a]	PENETRATION NUMBER	VALVE GROUP[1]	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. Automatic Isolation Valves					
MSIV	1B21*AOVF022A[b][g]	1KJB*Z1A	6	5	No
MSIV	1B21*AOVF022B[b][g]	1KJB*Z1B	6	5	No
MSIV	1B21*AOVF022C[b][g]	1KJB*Z1C	6	5	No
MSIV	1B21*AOVF022D[b][g]	1KJB*Z1D	6	5	No
MSIV	1B21*AOVF028A[g]	1KJB*Z1A	6	5	No
MSIV	1B21*AOVF028B[g]	1KJB*Z1B	6	5	No
MSIV	1B21*AOVF028C[g]	1KJB*Z1C	6	5	No
MSIV	1B21*AOVF028D[g]	1KJB*Z1D	6	5	No
Turbine Plant Misc. Drains	1B21*MOVF067A	1KJB*Z1A	6	19.8	No
Turbine Plant Misc. Drains	1B21*MOVF067B	1KJB*Z1B	6	19.8	No
Turbine Plant Misc. Drains	1B21*MOVF067C	1KJB*Z1C	6	19.8	No
Turbine Plant Misc. Drains	1B21*MOVF067D	1KJB*Z1D	6	19.8	No
Turbine Plant Misc. Drains	1B21*MOVF016[b]	1KJB*Z2	6	16.5	No
Turbine Plant Misc. Drains	1B21*MOVF019[g]	1KJB*Z2	6	17.6	No
RHR Return to FW	1E12*MOVF053A[m]	1KJB*Z3A	5	39	No
RHR Return to FW	1E12*MOVF053B[m]	1KJB*Z3B 1DRB*Z13	5	39	No
RHR Shutdown Cooling Supply	1E12*MOVF008	1KJB*Z20	5	29.7	No
RHR Shutdown Cooling Supply	1E12*MOVF009[b]	1KJB*Z20	5	25.3	No
LPCI A to Reactor	1E12*MOVF037A[m]	1KJB*Z21A	14	73.7	No
LPCI B to Reactor	1E12*MOVF037B[m]	1KJB*Z21B	14	74.8	No
MS-PLCS Line	1E33*MOVF008[d][k]	1KJB*Z1A,B,C,D	4	14.5	No
RWCU Disch. to Condenser	1G33*MOVF028	1KJB*Z4	15	20.9	Yes
RWCU Return to FW	1G33*MOVF040	1KJB*Z6	15	24.2	No
RWCU Pump Suction	1G33*MOVF001[b]	1KJB*Z7	16	19.8	No
RWCU Pump Disch.	1G33*MOVF053	1KJB*Z129	15	6.5	No
RWCU Disch. to Condenser	1G33*MOVF034	1KJB*Z4	15	20.9	Yes
RWCU Return to FW	1G33*MOVF039	1KJB*Z6	15	24.2	No
RWCU Pump Suction	1G33*MOVF004	1KJB*Z7	7	19.8	No
RWCU Pump Disch.	1G33*MOVF054	1KJB*Z129	15	6.5	No
RWCU Backwash Disch.	1WCS*MOV178	1KJB*Z5	1	12.1	Yes
RWCU Backwash Disch.	1WCS*MOV172	1KJB*Z5	1	12.6	Yes
HPCS Test Return-Supp. Pool	1E22*MOVF023[j]	1KJB*Z11	1	50	No

continued

TR 3.6.1.8 Penetration Valve Leakage Control System (PVLCS)

---------------------------------NOTE---------------------------------
The following Action applies to Condition B of Technical Specification LCO 3.6.1.9 and is to be entered concurrently.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Not Used	A.1 Not used	
B. Two PVLCS subsystems inoperable.	B.1 Perform Action B.1 of TS LCO 3.6.1.9. AND B.2.1 Verify by administrative means availability of the Diesel driven instrument air compressor. OR B.2.2 --------NOTE--------- An alternate air supply to the SVV accumulators is considered to be air pressure at or above 101 psig delivered by a diesel-driven air compressor or other source not dependent on off-site power. --------------------- Provide for a source of alternate air pressure for SR/V operation.	Per Condition B.1 4 hours AND once per 24 hours thereafter. 12 hours AND once per 24 hours thereafter.
C. Not used	C.1 Not used	

TR 3.6.2.1 Suppression Pool Average Temperature

TLCO DELETED

TR 3.6.2.2 Suppression Pool Water Level

TLCO DELETED

TR 3.7.2 Control Room Fresh Air (CRFA) System

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

The following surveillance Note applies to SR 3.7.2.3 of Technical Specification LCO 3.7.2.

1) This SR includes verification that the isolation valves close in $\leq$ 30 seconds.

TR 3.7.7 Snubbers

TLCO 3.7.7 All required snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODE 1, 2 and 3,
MODES 4 and 5 for snubbers located on systems required OPERABLE in MODES 4 or 5.

ACTIONS

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

Separate Condition entry is allowed for each required snubber.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required snubbers inoperable on a system	A.1 Declare the supported system inoperable,	Immediately
	AND	
	A.2 Replace or restore the inoperable snubber(s) to OPERABLE status,	Prior to returning supported system to OPERABLE status
	AND	
	A.3 Perform engineering evaluations per the applicable section of the approved ISI Program.	Prior to returning supported system to OPERABLE status

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.7.1	-----------------NOTE----------------- Only a previously approved revision of the ISI Program may be implemented. Subsequent revisions to the program shall be submitted to the NRC in accordance with the requirements of 10 CFR 50.55a(g). -------------------------------------	
	Each required snubber shall be demonstrated to be OPERABLE by implementing the examination and test requirements of the approved ISI Program.	As specified in the approved ISI program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.9.3.1.	(Deleted)	
TSR 3.7.9.3.2.	Verify Halon storage tank weight and pressure.	24 months
TSR 3.7.9.3.3	-----------------NOTE--------------------- Actual Halon release and Halon bottle explosive initiator valve actuation may be excluded from the test -- Verify the system actuates, manually and automatically, upon receipt of a simulated actuation signal.	24 months
TSR 3.7.9.3.4	Perform a flow test through headers and nozzles to assure no blockage.	24 months

TR 3.7.10 Area Temperature Monitoring

TLCO 3.7.10 The temperature of each area shown in Table 3.7.10-1 shall be maintained within the limits indicated in Table 3.7.10-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each area.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more areas exceeding the temperature limit(s) shown in Table 3.7.10-1	A.1 Restore the area to within its temperature limit.	8 hours
B. One or more areas exceeding the temperature limit(s) shown in Table 3.7.10-1 by > 30°F	B.1 Enter Condition C <u>AND</u> B.2.1 Restore the area to within its temperature limit OR B.2.2 Declare the equipment in the affected area inoperable.	Immediately 4 hours 4 hours
C. Condition B entered <u>OR</u> Required Action and associated Completion Time for Condition A not met	C.1 Prepare and submit a Special Report to the Commission, pursuant to Requirement 5.6.9, providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment	30 days

TR 3.8.11 Electrical Equipment Protective Devices

TLCO 3.8.11 Each of the primary and backup overcurrent protective devices associated with each primary containment electrical penetration circuit as shown in Table 3.8.11-1 shall be OPERABLE. The scope of the protective devices excludes those circuits for which credible fault currents would not exceed the penetrations' design ratings.

APPLICABILITY: MODE 1, 2, and 3.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each primary containment penetration conductor overcurrent protective device.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the 480 volt MCC circuit breaker/fuse combination starters inoperable.	A.1 Determine the operability of the affected system or component.	Immediately
	<u>AND</u>	
	A.2 Remove the inoperable starter(s) from service by locking the breaker(s) open and removing the control power fuse(s).	72 hours
	<u>AND</u>	
	A.3 Verify the inoperable starter(s) circuit breakers(s) to be locked open and the control power fuse(s) removed.	Once per 7 days
B. One or more of the 480 volt circuit breakers inoperable.	--------------NOTE-------------- Complete removal of breaker(s) is an acceptable alternative to racking out. --------------------------------	
	B.1 Determine the operability of the affected system or component.	Immediately
	<u>AND</u>	
	B.2 Remove the inoperable circuit breaker(s) from service by racking out the breaker.	72 hours
	<u>AND</u>	
	B.3 Verify the inoperable breakers(s) racked out.	Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more of the 4.16 kV breaker(s) inoperable.	C.1 Determine the operability of the affected system or component.	Immediately
	AND	
	C.2 De-energize the 4.16 kV circuit(s) by tripping the associated redundant circuit breaker(s).	72 hours
	AND	
	C.3 Verify the redundant circuit breaker tripped.	Once per 7 Days
D. One or more of the 120/240 volt molded case circuit breaker(s) inoperable.	D.1 Determine the operability of the affected system or component.	Immediately
	AND	
	D.2 Remove the inoperable circuit breaker(s) from service by tripping both 120/240 volt breakers open and locking the upstream 480 volt MCC breaker(s) open.	72 hours
	AND	
	D.3 Verify the 480 volt MCC breaker(s) to be locked open.	Once per 7 Days

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TSR 3.8.11.1	A CHANNEL CALIBRATION of the associated protective relays for the 4.16 kV breakers.	18 months on a STAGGERED TEST BASIS
TSR 3.8.11.2	Perform an integrated system functional test of the 4.16 kV breakers which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.	18 months on a STAGGERED TEST BASIS
TSR 3.8.11.3	-----------------NOTES------------------ Testing of these circuit breakers shall consist of injecting currents in excess of the breaker's nominal setpoint and measuring the response time of the long time and short time delay elements and the setpoint of the instantaneous element, as appropriate. The measured data shall be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. --------------------------------------- Functionally test a representative sample of at least 10% of each type of lower voltage (≤ 480 volt) circuit breakers.	18 months on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
TSR 3.8.11.4	----------------NOTES---------------- Testing of these motor starters shall consist of injecting a current with a value equal to the locked rotor current of the associated motor and verifying that the motor starter operates to interrupt the current within the associated thermal overload time delay band width for that current as specified by the manufacturer. Motor starters found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. ------------------------------------- Functionally test a representative sample of at least 10% of each type of motor starter used for penetration redundant overcurrent protection.	18 months on a STAGGERED TEST BASIS
TSR 3.8.11.5	Subject each circuit breaker to an inspection and preventive maintenance program in accordance with procedures prepared in conjunction with its manufacturer's recommendations.	60 months

Table 3.8.13-1
A. C. circuits inside containment

Equipment ID	Location	Device
1F42-D002H	1SCA-PNL8C1	Circuit Breaker 15
Deleted		
Deleted		
1HVR-UC1CH	1SCA-PNL2C1	Circuit Breaker 9
1HVR-FN1AH	1SCA-PNL2A2	Circuit Breaker 3
1HVR-FN1BH	1SCA-PNL2F1	Circuit Breaker 6
1HVR-FN1CH	1SCA-PNL2E1	Circuit Breaker 1
1HVR-FN1DH	1SCA-PNL2B1	Circuit Breaker 6
1DRS-UC1AH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1CH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1EH	1SCA-PNL2E1	Circuit Breaker 2
1WCS-P5AH	1SCA-PNL2E1	Circuit Breaker 4
1DRS-UC1BH	1SCA-PNL2F1	Circuit Breaker 3
1DRS-UC1DH	1SCA-PNL2F1	Circuit Breaker 3
1DRS-UC1FH	1SCA-PNL2F1	Circuit Breaker 3
1WCS-P5BH	1SCA-PNL2F1	Circuit Breaker 2

TR 3.9.5 Control Rod Scram Accumulators - Refueling

-----------------------------------NOTE-----------------------------------
The following surveillance requirement applies to Technical Specification LCO 3.9.5. Failure to meet this surveillance requirement requires entry into Technical Specification LCO 3.9.5.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.9.5.1 and 3.9.5.2 (Not Used)		
TSR 3.9.5.3	Measure and record the time, for up to 10 minutes, that each individual accumulator check valve maintains the associated accumulator pressure above the alarm set point.	18 months

TR 3.9.10 Decay Time

TLCO 3.9.10 The reactor shall be subcritical for ≥ 24 hours.*

APPLICABILITY: MODE 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The reactor subcritical < 24 hours.	A.1 Suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.9.10.1	The reactor shall be determined to have been subcritical ≥ 24 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor pressure vessel.

* The reactor shall be subcritical for at least 24 hours prior to opening vent and drain line pathways under the provisions of Technical Specification 3.6.1.10.

TR 3.11 RADIOACTIVE EFFLUENTS

TR 3.11.1.1 Liquid Effluents - Concentration

TLCO 3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see USAR Section 2.1) shall be limited to ten times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits	A.1 Restore the concentration to within the above limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.11.1.1.1	Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 3.11.1.1-1	according to the sampling and analysis program of Table 3.11.1.1-1.
TSR 3.11.1.1.2	The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Requirement 3.11.1.1	according to the sampling and analysis program of Table 3.11.1.1-1.

TABLE 3.12.1-1 (page 1 of 4)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1.DIRECT RADIATION[b]	24 routine monitoring stations either with two or more dosimmeters or with one instrument for measuring and recording dose rate continuously, placed as follows: one ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY; the balance of the stations (8) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	mR exposure quarterly.
2.AIRBORNE			
Radioiodine and Particulates	Samples from 4 locations: 2 samples from close to the 2 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average groundlevel D/Q. 1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q. 1 sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]	Continuous sampler operation with sample collection every two weeks, or more frequently if required by dust loading.	<u>Radioiodine Cannister</u>: I-131 analysis every two weeks. <u>Particulate Sampler</u>: Gross beta radioactivity analysis following filter change every two weeks;[d]

(continued)

TABLE 3.12.1-1 (page 2 of 4)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface[f]	1 sample upstream and 1 sample downstream.	Grab samples quarterly.	Gamma isotopic analysis[e] quarterly, tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources only if likely to be affected[g]	Semiannually	Gamma isotopic[e] and tritium analysis semiannually.
c. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value.	Annually	Gamma isotopic analysis[e] annually.
4. INGESTION			
a. Milk	If commercially available 1 sample from milking animals within 8 km distant where doses are calculated to be greater than 1 mrem per yr.[h] 1 sample from milking animals at a control location 15-30 km distant when an indicator location exists.	Quarterly when animals are on pasture	Gamma isotopic[e] and I-131 analysis quarterly when animals are on pasture.
b. Fish and Inverte-brates	1 sample of a commercially and/or recreationally important species in vicinity of plant discharge area. 1 sample of similar species in areas not influenced by plant discharge.	Annually	Gamma isotopic analysis[e] on edible portions annually.
c. Food Products	1 Sample of one type of broad leaf vegetation grown near the site boundary location of highest predicted annual average ground level D/Q if milk sampling is not performed.	Quarterly during the growing season.	Gamma isotopic[e] and I-131 analysis quarterly.
	1 sample of similar broad leaf vegetation grown 15-30 km distant, if milk sampling is not performed.[c]	Quarterly during the growing season.	Gamma isotopic[e] and I-131 analysis quarterly.

TR 3.12.3 Interlaboratory Comparison Program

TLCO 3.12.3 Analyses shall be performed on radioactive materials, that correspond to samples required by Table 3.12.1-1, supplied as part of an Interlaboratory Comparison Program.

APPLICABILITY: At all times.

ACTIONS

a. With analyses not being performed as required above, report to the Commission, in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2, the corrective actions taken to prevent a recurrence.

SURVEILLANCE REQUIREMENTS

TSR 3.12.3.1 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

TR 5.0 ADMINISTRATIVE CONTROLS

TR 5.1 Responsibility

TR 5.1.1 (Not used)

TR 5.1.2 A management directive restating the Technical Specification 5.1.2 Control Room command function requirements signed by the Vice President-RBS shall be issued to all station personnel on an annual basis.

TR 5.2 Organization

TR 5.2.1 Plant Specific Titles

The following are the plant specific titles for the personnel fulfilling responsibilities of positions delineated in Technical Specifications:

a. The corporate executive responsible for overall plant nuclear safety is Vice President-River Bend Station (RBS).

b. The Plant manager is the General Manager.

c. The shift superintendent is the Shift Manager.

d. A non-licensed operator is a Nuclear Equipment Operator.

e. The operations manager is the Manager - Operations.

f. The operations middle manager is the Assistant Operations Manager.

g. The radiation protection manager is the Manager - Radiation Protection.

h. A health physics technician is an individual qualified as a Radiation Protection Technician.

i. Health Physics supervision is Radiation Protection personnel, Supervisor and above.

TR 5.6 Reporting Requirements

The reports required by Technical Specification 5.6 and the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Office of the NRC and a copy to the NRC Resident Inspector, unless otherwise noted.

TR 5.6.1 (Not Used)

TR 5.6.2 Annual Radiological Environmental Operating Report

Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted as required by Technical Specification 5.6.2.

The Annual Radiological Environmental Operating Reports shall also include a comparison of data for the period (as appropriate) with preoperational studies, operational controls and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Requirement 3.12.2.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor plant; the results of licensee participation in the Interlaboratory Comparison Program, required by Requirement 3.12.3; discussion of all deviations from the Sampling Schedule of Table 3.12.1-1; and discussion of all analyses in which the LLD required by Table 3.12.1-3 was not achievable.

* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

TR 5.8.2 Technical Review and Control

TR 5.8.2.1 Each procedure and program required by Technical Specifications 5.4, 5.5, and other procedures that affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group that prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group that prepared the procedure. Each such procedure and program, or changes thereto, shall be approved, prior to implementation, by the plant manager, or the Manager - Radiation Protection, or the manager/department head responsible for the program or the activity described in the procedure.

TR 5.8.2.2 Individuals responsible for reviews performed in accordance with Section TR 5.8.2.1 shall be members of River Bend Station supervisory staff, and the reviews shall be performed in accordance with administrative procedures. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary and a verification that the proposed actions do not constitute an unreviewed safety question. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

TR 5.8.2.3 The station security program and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Requirement 5.8.2.1.

TR 5.8.2.4 The station emergency plan and implementing procedures and recommended changes shall be approved in accordance with Requirement 5.8.2.1.

TR 5.8.2.5 The station fire protection plan and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Requirement 5.8.2.1.

TR 5.8.2.6 The station Technical Requirements Manual and implementing procedures and recommended changes shall be approved in accordance with Requirement 5.8.2.1.

Enclosure 2 to RBG-46179

River Bend Station

Technical Specification Bases Revision Package

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

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B 2.0-1	0	B 3.1-24	107	B 3.3-1	0	B 3.3-39e	101
B 2.0-2	6-15	B 3.1-25	107	B 3.3-2	6-7	B 3.3-39f	4-8
B 2.0-3	6-15	B 3.1-26	0	B 3.3-3	0	B 3.3-39g	4-8
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B 3.0-4	0	B 3.1-36	0	B 3.3-11	1	B 3.3-48	6-13
B 3.0-5	0	B 3.1-37	0	B 3.3-12	0	B 3.3-49	1
B 3.0-6	0	B 3.1-38	0	B 3.3-13	6-4	B 3.3-50	0
B 3.0-7	0	B 3.1-39	0	B 3.3-14	0	B 3.3-51	1
B 3.0-8	0	B 3.1-40	0	B 3.3-15	0	B 3.3-52	116
B 3.0-9	0	B 3.1-41	5-6	B 3.3-16	0	B 3.3-53	6-2
B 3.0-10	0	B 3.1-42	1	B 3.3-17	2-7	B 3.3-54	6-2
B 3.0-11	0	B 3.1-43	6-14	B 3.3-18	1	B 3.3-55	1
B 3.0-12	108	B 3.1-44	1	B 3.3-19	1	B 3.3-56	0
B 3.0-13	108	B 3.1-45	115	B 3.3-20	1	B 3.3-57	0
B 3.0-13a	108	B 3.1-46	0	B 3.3-21	1	B 3.3-58	0
B 3.0-14	0	B 3.1-47	0	B 3.3-22	1	B 3.3-59	1
B 3.0-15	0	B 3.1-48	0	B 3.3-23	1	B 3.3-60	0
B 3.1-1	0	B 3.1-49	115	B 3.3-24	1	B 3.3-61	0
B 3.1-2	0	B 3.2-1	0	B 3.3-25	6-4	B 3.3-62	0
B 3.1-3	0	B 3.2-2	6-4	B 3.3-25a	4-8	B 3.3-63	0
B 3.1-4	0	B 3.2-3	6-4	B 3.3-26	4-8	B 3.3-64	0
B 3.1-5	0	B 3.2-4	3-7	B 3.3-27	6-15	B 3.3-65	0
B 3.1-6	0	B 3.2-5	0	B 3.3-28	1	B 3.3-66	0
B 3.1-7	3-10	B 3.2-6	6-15	B 3.3-29	5-1	B 3.3-67	1
B 3.1-8	3-10	B 3.2-7	6-4	B 3.3-30	4-8	B 3.3-68	1
B 3.1-9	3-10	B 3.2-8	6-15	B 3.3-31	4-8	B 3.3-69	1
B 3.1-10	3-10	B 3.2-9	0	B 3.3-31a	4-8	B 3.3-70	0
B 3.1-11	3-10	B 3.2-10	6-4	B 3.3-32	0	B 3.3-71	0
B 3.1-12	0	B 3.2-11	6-4	B 3.3-33	0	B 3.3-72	1
B 3.1-13	0	B 3.2-12	4-8	B 3.3-34	0	B 3.3-73	1
B 3.1-14	0	B 3.2-13	4-8	B 3.3-35	0	B 3.3-74	2-1
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B 3.1-17	6-13	B 3.2-16	4-8	B 3.3-38	0	B 3.3-77	106
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B 3.3-88	0	B 3.3-128	0	B 3.3-168	0	B 3.3-208	0
B 3.3-89	0	B 3.3-129	0	B 3.3-169	0	B 3.3-209	1
B 3.3-90	0	B 3.3-130	0	B 3.3-170	110	B 3.3-210	1
B 3.3-91	0	B 3.3-131	0	B 3.3-171	6-5	B 3.3-211	1
B 3.3-92	0	B 3.3-132	0	B 3.3-172	6-5	B 3.3-212	107
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B 3.3-118	0	B 3.3-158	115	B 3.3-198	3-4	B 3.4-16	6-7
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B 3.6-94	6-5	B 3.6-133	3-4	B 3.7-30	0	B 3.8-36	0
B 3.6-95	6-5	B 3.6-134	2-8	B 3.7-31	115	B 3.8-37	115
B 3.6-96	0	B 3.6-135	2-8	B 3.8-1	0	B 3.8-38	110
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B 3.6-102	0	B 3.6-141	2-8	B 3.8-7	0	B 3.8-44	0
B 3.6-103	0	B 3.6-142	2-8	B 3.8-8	105	B 3.8-45	0
B 3.6-104	6-5	B 3.7-1	110	B 3.8-8a	105	B 3.8-46	0
B 3.6-105	110	B 3.7-2	110	B 3.8-9	105	B 3.8-47	0
B 3.6-106	0	B 3.7-3	110	B 3.8-10	0	B 3.8-48	3-2
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B 3.6-110	6-5	B 3.7-7	3-1	B 3.8-14	102	B 3.8-52	0
B 3.6-111	6-5	B 3.7-8	1	B 3.8-15	102	B 3.8-53	0
B 3.6-112	0	B 3.7-9	0	B 3.8-16	102	B 3.8-54	1
B 3.6-113	6-5	B 3.7-10	110	B 3.8-17	102	B 3.8-55	0
B 3.6-114	6-5	B 3.7-11	110	B 3.8-18	102	B 3.8-56	103
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B 3.6-118	0	B 3.7-15	0	B 3.8-22	113	B 3.8-60	110
B 3.6-119	110	B 3.7-16	110	B 3.8-23	113	B 3.8-61	115
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B 3.6-121	101	B 3.7-18	110	B 3.8-25	102	B 3.8-63	0
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B 3.6-125	2-4	B 3.7-22	0	B 3.8-29	113	B 3.8-67	4-5
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B 3.8-75	115	B 3.9-23	0	B 3.10-28	0		
B 3.8-76	110	B 3.9-24	115	B 3.10-29	0		
B 3.8-77	0	B 3.9-25	0	B 3.10-30	0		
B 3.8-78	0	B 3.9-26	0	B 3.10-31	0		
B 3.8-79	1	B 3.9-27	4-2	B 3.10-32	0		
B 3.8-80	0	B 3.9-28	4-2	B 3.10-33	0		
B 3.8-81	0	B 3.9-28a	4-2	B 3.10-34	0		
B 3.8-82	0	B 3.9-29	0	B 3.10-35	0		
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B 3.8-85	103	B 3.9-32	4-2	B 3.10-38	6-14		
B 3.8-86	0	B 3.9-32a	4-2				
B 3.8-87	0	B 3.10-1	0				
B 3.8-88	1	B 3.10-2	0				
B 3.8-89	110	B 3.10-3	0				
B 3.8-90	115	B 3.10-4	0				
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B 3.9-14	0	B 3.10-20	0				
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B 3.9-16	0	B 3.10-22	0				
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BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," Limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

2.2.3

If any SL is violated, the General Manager and the Vice President shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be provided to the General Manager and the Vice President.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II," (latest approved revision).
3. 10 CFR 50.72.
4. 10 CFR 50.67.
5. 10 CFR 50.73.
6. ANF-524(P)(A),Revision 2, Supplements 1, and 2, November 1990.
7. EMF-2209(P)(A), Revision 1, July 2000.
8. Letter: CEXO-2000-00293, J.B. Lee (EOI) to K. V. Walker (SPC), "Grand Gulf Nuclear Station Unit 1 and River Bend Station Unit 1, Reload Transition Data – GE11 Additive Constants", July 25, 2000.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67, "Accident Source Term" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

2.2.3

If any SL is violated, the General Manager and the Vice President shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be submitted to the General Manager and the Vice President.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWA-5000.
4. 10 CFR 50.67.
5. ASME, Boiler and Pressure Vessel Code, 1971 Edition, Addenda, summer of 1973.
6. ASME, Boiler and Pressure Vessel Code, 1977 Edition, Addenda, summer of 1977.
7. 10 CFR 50.72.
8. 10 CFR 50.73.

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SR include a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Primary Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been

(continued)

BASES

SR 3.0.3 (continued)

performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10CFR 50 Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability or personnel, and the time required to perform the Surveillance. This risk impact should be managed through

BASES

SR 3.0.3 (continued)

the program in place to implement 10CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182 "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guidance addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable then is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated (i.e., charging valve closed), the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 950 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Scram insertion times increase with increasing reactor pressure because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure greater than 950 psig ensures that the scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure scram time testing is performed within a reasonable time following a refueling or after a shutdown ≥ 120 days, all control rods are required to be tested before exceeding 40% RTP. This Frequency is acceptable, considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by work on control rods or the CRD System.

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains "representative" if no more than 7½% of the control rods in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.2 (continued)

the tested sample are determined to be "slow." If more than 7½% of the sample is declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 7½% criterion (e.g., 7½% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data were previously tested in a sample. The 7½% of sample size criteria is intended to align with the 7½% of the total control rods allowed to have scram times that exceed the specified limit. The 200 day Frequency is intended to allow consistency with control rod sequence exchanges and is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable, based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures from zero to the maximum permissible pressure. The scram testing must be performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate that the affected control rod is still within acceptable limits. The limits for reactor pressures < 950 psig are established based on a high probability of meeting the acceptance criteria at reactor pressures ≥ 950 psig. Limits for ≥ 950 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within the 7 second limit of Table 3.1.4-1 Note 2, the control rod can be declared OPERABLE and "slow."

Specific examples of work that could affect the scram times include (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator isolation valve, or check valves in the piping required for scram.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 50.67 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 50.67 (Ref. 2) and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves also

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.3 (continued) reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 18 month 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 the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. USAR, Section 4.6.1.1.2.4.2.5. 2. 10 CFR 50.67. 3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

BASES

LCO

4. Suppression Pool Water Level (continued)

provided to detect a breach in the reactor coolant pressure boundary (RCPB). This variable is also used to verify and provide long term surveillance of ECCS function. The wide range suppression pool water level measurement provides the operator with sufficient information to assess the status of the RCPB and to assess the status of the water supply to the ECCS. The wide range water level indicators monitor the suppression pool level from 5 ft above the bottom of the pool to 23 ft 9 inches above the bottom of the pool. Two wide range suppression pool water level signals are transmitted from separate differential pressure transmitters and are continuously recorded on two recorders in the control room. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

5. Suppression Pool Sector Water Temperature

Suppression pool sector water temperature is a Category I variable provided to detect a condition that could potentially lead to containment breach, and to verify the effectiveness of ECCS actions taken to prevent containment breach. The suppression pool water temperature instrumentation allows operators to detect trends in suppression pool water temperature in sufficient time to take action to prevent steam quenching vibrations in the suppression pool.

Fourteen temperature sensors are provided for normal monitoring of suppression pool sector water temperature. Four of the fourteen are located below the post LOCA ECCS drawdown water level of the suppression pool and are used for post accident monitoring. The output of both normal pool temperature monitoring sensors and the PAM sensors are recorded in the main control room. One sector is monitored by PAM sensors CMS-RTD40A and CMS-RTD40D. The other sector is monitored by PAM sensors CMS-RTD40B and CMS-RTD40C. Each PAM sensor and its recorder represent a channel. Two channels are required in each sector.

These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channels.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ATWS-RPT is assumed in the safety analysis. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation is included as required by the NRC Policy Statement.

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.4. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the associated recirculation pump drive motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The individual Functions are required to be OPERABLE in MODE 1 to protect against common mode failures of the

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BASES

ACTIONS

G.1 and G.2 (continued)

Required Action G.1 states that Required Action G.1 is only applicable for Functions 4.c, 4.e, 4.f, 4.g, 5.c, 5.e, and 5.f. Required Action G.1 is not applicable to Functions 4.h and 5.g (which also require entry into this Condition if a channel in these Functions is inoperable), since they are the Manual Initiation Functions and are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 96 hours or 8 days (as allowed by Required Action G.2) is allowed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action G.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable channels within similar ADS trip system Functions, as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status if both HPCS and RCIC are OPERABLE (Required Action G.2). If either HPCS or RCIC is inoperable, the time is reduced to 96 hours. If the status of HPCS or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCS or RCIC inoperability. However, total time for an inoperable channel cannot exceed 8 days. If the status of HPCS or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.4 and SR 3.3.5.1.5

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

The Frequency of SR 3.3.5.1.4 and SR 3.3.5.1.5 is based upon the assumption of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage (except for Division III which can be tested in any operational condition) and the potential for unplanned transients if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Main Steam Line Isolation

1.a. Reactor Vessel Water Level-Low Low Low, Level 1

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, Level 1 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, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSL on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level-Low Low Low, Level 1 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, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 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 6 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/hour 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/hour)

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure-Low (continued)

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 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 23.8% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure-Low 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 high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure-Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 6 valves.

1.c. Main Steam Line Flow-High

Main Steam Line Flow-High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow-High Function is directly assumed in the analysis of the main steam line break (MSLB) accident (Ref. 1). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e. Main Steam Tunnel Ambient Temperature-High

Ambient Temperature-High 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 USAR, since bounding analyses are performed for large breaks such as MSLBs.

Ambient temperature signals are initiated from thermocouples located in the area being monitored. Four channels of each ambient temperature Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each Function has one temperature element.

The ambient temperature monitoring Allowable Value is chosen to detect a leak equivalent to 25 gpm.

This Function isolates the Group 6 valves.

1.j. Manual Initiation

The Manual Initiation push button channels introduce signals into the MSL isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of Manual Initiation Function are available and are required to be OPERABLE.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Primary Containment and Drywell Isolation

2.a. Reactor Vessel Water Level-Low Low, Level 2

Low RPV water level indicates the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Reactor Vessel Water Level-Low Low, Level 2 Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA. In addition, Function 2.a provides an isolation signal to certain drywell isolation valves. The isolation of the drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the primary containment.

Reactor Vessel Water Level-Low Low, Level 2 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level-Low Low, Level 2 Function are available and are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level-Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level-Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 1, 7, 8, 9, 15, and 16 valves. The isolation of valve Group 9 also includes the actuation of the Standby Gas Treatment System, the Control Room Fresh Air System, and the containment hydrogen analyzers.

2.b. Drywell Pressure-High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the automatic isolation valves on high

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	2.b. Drywell Pressure-High (continued)

drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell Pressure-High Function associated with isolation of the primary containment is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA. In addition, Function 2.b provides an isolation signal to certain drywell isolation valves. The isolation of the drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the primary containment.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure-High per 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 may be indicative of a LOCA inside primary containment.

This Function isolates the Group 1, 3, and 8 valves. The isolation of valve Group 8 also includes the actuation of the Standby Gas Treatment System, the Control Room Fresh Air System, and the containment hydrogen analyzers.

2.c. Containment Purge Isolation Radiation-High

The Containment Purge Isolation Radiation - High isolation instrumentation is provided to contain the radioactivity released to the Primary Containment after either an MSIV closure event or the Design Basis LOCA event. The allowable setpoint for the Containment Purge Isolation Radiation - High instrumentation is set at a value determined by the radioactivity released by an MSIV closure event. Since the radioactivity levels for the MSIV closure event are less than for the LOCA event, the allowable setpoint ensures isolation capability for both events. When high purge exhaust radiation is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.d. RCIC Turbine Exhaust Diaphragm Pressure-High (continued)

The Allowable Values are low enough to prevent damage to the system turbine.

This Function isolates the Group 2 valves.

3.e, 3.h. Ambient Temperature-High

Ambient 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 USAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks. However, Function 3.e is credited in the high energy line break analysis described in USAR Section 3.6 and USAR Appendix 3B.

Ambient Temperature-High signals are initiated from thermocouples that are appropriately located to protect the system that is being monitored. Two instruments monitor each area. Six channels for RHR and RCIC Ambient Temperature-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. There are two for the RCIC room and four for the RHR area.

The Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

This Function isolates the Group 2 valves.

3.f. Main Steam Line Tunnel Ambient Temperature-High

Ambient Temperature-High 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 limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the USAR, since bounding analyses are performed for large breaks such as MSLBs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.i. RCIC/RHR Steam Line Flow-High (continued)

could uncover. Therefore, the isolation is initiated at high flow to prevent or minimize core damage. Specific credit for this Function is not assumed in any USAR accident or transient analysis since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC/RHR steam line break from becoming bounding. This function is credited in the high energy line break analysis described in USAR Section 3.6 and USAR Appendix 3B.

The RCIC/RHR steam line flow signals are initiated from two transmitters that are connected to the steam line. Two channels with one channel in each trip system are available and required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Allowable Value is selected to ensure that the trip occurs to prevent fuel damage and maintains the MSLB as the boundary event.

This Function actuates the Group 2 valves.

3.J. Drywell Pressure-High

High drywell pressure can indicate a break in the RCPB. The RCIC isolation of the turbine exhaust is provided to prevent communication with the drywell when high drywell pressure exists. A potential leakage path exists via the turbine exhaust. The isolation is delayed until the system becomes unavailable for injection (i.e., low steam line pressure). The isolation of the RCIC turbine exhaust by Drywell Pressure-High is indirectly assumed in the USAR accident analysis because the turbine exhaust leakage path is not assumed to contribute to offsite doses.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Two channels of RCIC 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 3 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.c, 4.d, 4.e, 4.f, 4.g. Ambient Temperature-High (continued)

instruments is not taken in any transient or accident analysis in the USAR, since bounding analyses are performed for large breaks such as MSLBs. However, Functions 4.c, 4.d, and 4.f are credited in the high energy line break analysis described in USAR Sections 3.6 and 6.2.1.1.3.2 as well as USAR Appendix 3B.

Ambient temperature signals are initiated from temperature elements that are located in the room that is being monitored. There are fourteen thermocouples that provide input to the Area Temperature-High Functions (two per area). Fourteen channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Ambient Temperature-High Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

These Functions isolate the Group 7, 15, and 16 valves.

4.h. Main Steam Line Tunnel Ambient Temperature-High

Ambient Temperature-High 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, since bounding analyses are performed for large breaks such as MSLBs.

Ambient temperature signals are initiated from thermocouples located in the area being monitored. Two channels of Main Steam Tunnel Temperature-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each Function has one temperature element.

The Allowable Values are chosen to detect a leak equivalent to 25 gpm.

This Function isolates the Group 7, 15, and 16 valves.

4.i. Reactor Vessel Water Level-Low Low, Level 2

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.b. Reactor Vessel Water Level-Low, Level 3 (continued)

maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

The Reactor Vessel Water Level-Low, Level 3 Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level-Low, Level 3 Allowable Value (LCO 3.3.1.1) since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level-Low, Level 3 Function is required to be OPERABLE in MODES 1, 2, 3 with reactor pressure less than the RHR cut-in permissive pressure, 4, and 5 to prevent this potential flow path from lowering reactor vessel level to the top of the fuel. This instrumentation is required to be OPERABLE in MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure to support actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded.

This Function isolates the Group 5 and 14 valves.

5.c. Reactor Vessel Water Level-Low Low Low, Level 1

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the shutdown cooling portion of the RHR System occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level-Low Low Low, Level 1 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, Level 1 Function associated with isolation is implicitly assumed in the USAR analysis since these leakage paths are assumed to be isolated for a DBA.

Reactor vessel water level signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level-Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.c. Reactor Vessel Water Level-Low Low Low, Level 1 (continued)

The Reactor Vessel Water Level-Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the shutdown cooling portion of the RHR System isolates on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the Group 10 valves.

5.d. Reactor Steam Dome Pressure-High

The Shutdown Cooling System Reactor Steam Dome Pressure-High Function is provided to isolate the shutdown cooling portion of the RHR System. This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario and credit for the interlock is not assumed in the accident or transient analysis in the USAR.

The Reactor Steam Dome—High pressure signals are initiated from four transmitters. Four channels of Reactor Steam Dome 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 chosen to be low enough to protect the system equipment from overpressurization.

This Function isolates the Group 5 valves.

5.e. Drywell Pressure-High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the PCIVs on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell Pressure-High Function associated with isolation of the RHR Shutdown Cooling System is not modeled in any USAR accident or transient analysis because other leakage paths (e.g., MSIVs) are more limiting.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four 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.

(continued)

BASES

ACTIONS (continued)

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours, depending on the Function, has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to OPERABLE status. Functions that share common instrumentation with the RPS have a 12 hour allowed out of service time consistent with the time provided for the associated RPS instrumentation channels. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic isolation capability being lost for the associated penetration flow path(s). The MSL isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that both trip systems will generate a trip signal from the given Function on a valid signal. The other isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two automatic isolation valves in the associated penetration flow path can receive an isolation signal from the given Function. For Functions 1.a, 1.b, 1.d, and 1.e, this would require both trip systems to have one

(continued)

B 3.3 INSTRUMENTION

B 3.3.6.2 Secondary Containment and Fuel Building Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation dampers (SCIDs) and starts appropriate ventilation subsystems. Similarly, the fuel building isolation instrumentation automatically initiates closure of appropriate fuel building isolation dampers (FBIDs) and initiates fuel building ventilation flow through the filtration system. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1), such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.67 that are part of the NRC staff approved licensing basis. Secondary containment isolation and establishment of vacuum within the assumed time limits ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment or during certain operations when primary containment is not required to be OPERABLE are maintained within applicable limits. Fuel building isolation ensures that fission products released due to fuel uncovery or a dropped fuel assembly are also maintained within regulatory limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (a) reactor vessel water level, (b) drywell pressure, and (c) fuel building ventilation exhaust radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation parameters. In addition, manual initiation of the logic is provided.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure – High (continued)

Drywell Pressure-High signals are initiated from four pressure transmitters that sense drywell pressure. Four channels of Drywell Pressure-High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can preclude CRFA System initiation.

The Drywell Pressure-High Allowable Value was chosen to be the same as the Secondary Containment Isolation Drywell Pressure-High Allowable Value (LCO 3.3.6.2).

The Drywell Pressure-High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. In MODES 4 and 5, the Drywell Pressure-High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure-High setpoint.

3. Control Room Local Intake Ventilation Radiation Monitors

The Control Room Local Intake Ventilation Radiation Monitors measure radiation levels exterior to the inlet ducting of the MCR. A high radiation level may pose a threat to MCR personnel; thus, a detector indicating this condition automatically signals initiation of the CRFA System.

The Control Room Local Intake Ventilation Radiation Monitors Function consists of two independent monitors. Two channels of Control Room Local Intake Ventilation Radiation Monitors are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CRFA System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Room Local Intake Ventilation Radiation Monitors Function is required to be OPERABLE in MODES 1, 2, and 3, and during operations with a potential for draining the reactor vessel (OPDRVs) and movement of recently irradiated fuel in the primary containment or fuel building to ensure that control room personnel are protected during a LOCA, fuel handling event, or a vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c, 1.d, 1.e, 2.c, 2.d, 2.e. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV emergency bus indicates that while offsite power may not be completely lost to the respective emergency bus, power may be insufficient for starting large motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the bus is transferred from offsite power to onsite DG power when the voltage on the bus drops below the Degraded Voltage Function Allowable Values (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The Bus Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. To ensure an inadvertent power supply transfer does not occur, no more than three 1250 HP motors may be powered by Preferred Station Transformer RTX-XSR1D with grid voltage below the main control room Low-Low Grid Voltage alarm setpoint. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

Three channels of Division I and II - 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) Function per associated emergency bus and two channels of Division III - 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) Functions per associated emergency bus are only required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the USAR. A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3). The transient analyses of Chapter 15 of the USAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement.
LCO	Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, THERMAL POWER must be ≤ 77.6% RTP, total core flow limits are implemented through maintaining the operating loop flow within allowed flow rate, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM

(continued)

BASES

ACTIONS (continued)

B.1

Should a LOCA or transient occur with THERMAL POWER > 77.6% RTP, during single loop operation the core response may not be bounded by the safety analyses. Therefore, only a limited time is allowed to reduce THERMAL POWER to ≤ 77.6% RTP.

The 1 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing changes in THERMAL POWER to be quickly detected.

(continued)

BASES

LCO (continued)	+ 36 psig of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism. Operation with fewer valves OPERABLE than specified, or with setpoints outside the established limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.
APPLICABILITY	In MODES 1, 2, and 3, the specified number of S/RVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the heat. In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.
ACTIONS	A.1 and A.2 With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If one or more required S/RVs are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.4.4.1 This Surveillance demonstrates that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 2. The "as-left" SRV safety function lift setpoints are required to be within ± 1% of the specified nominal lift setpoint. Additionally, the sample size will be increased by two valves for each valve found outside of the "as found" safety lift setpoint. These requirements formed a portion of the basis for increasing the "as-found" lift setpoint tolerance to+ 3, -5% of the safety function lift setpoint. The demonstration of the S/RV safety function (continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1 (continued)

lift settings must be performed during shutdown, since this is a bench test, and 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 Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.

SR 3.4.4.2

The required relief function S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify the mechanical portions of the automatic relief function operate as designed when initiated either by an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.4 overlaps this SR to provide complete testing of the safety function.

The 18 month 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 SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.

SR 3.4.4.3

A manual actuation of each required S/RV (those valves removed and replaced to satisfy SR 3.4.4.1) is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods. If performed by method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 1), prior to valve installation. Therefore, this SR is modified by a note that states the surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued)

a reasonable time to complete the SR. If performed by method 2), valve OPERABILITY has been demonstrated for all installed S/RVs based upon the successful operation of a test sample of S/RVs.

1. Manual actuation of the S/RV, with verification of the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or pressure). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate reactor steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer.

2. The sample population of S/RVs tested each refueling outage to satisfy SR 3.4.4.1 will be stroked in the relief mode during "as-found" testing to verify proper operation of the S/RV. The successful performance of the test sample of S/RVs provides reasonable assurance that the remaining installed S/RVs will perform in a similar fashion. After the S/RVs are replaced, the relief-mode actuator of the newly installed S/RVs will be uncoupled from the S/RV, and cycled to ensure that no damage has occurred to the S/RV during transportation and installation. Following cycling, the relief-mode actuator is recoupled and the proper connection to the S/RV lever is independently verified.

This verifies that each replaced S/RV will properly perform its intended function.

If the valve fails to actuate due only to the failure of the solenoid, but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The STAGGERED TEST BASIS frequency ensures that each solenoid for each S/RV relief-mode actuator is alternately tested. The frequency of the required relief-mode actuator testing was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (ref. 1) as implemented by the Inservice Testing Program of

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.4.3 (continued) Specification 5.5.6. The testing frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. ASME, Boiler and Pressure Vessel Code, Section III and XI. 2. USAR, Section 5.2.2.2.3. 3. USAR, Section 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 50.67 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 50.67 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the USAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 50.67.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site. RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.
LCO	The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 50.67 limits.
APPLICABILITY	In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment. In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.
ACTIONS	A.1 and A.2 When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of a limiting event while exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 50.67 during a postulated MSLB accident.

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.4.8.1 This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level. This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.
REFERENCES	1. 10 CFR 50.67. 2. USAR, Section 15.6.4.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

The pump flow rates are verified with a pump differential pressure that is sufficient to overcome 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 LOCAs. These values may be established during pre-operational testing. The Frequency for this Surveillance is in accordance with the Inservice Testing Program requirements.

SR 3.5.1.5

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance test verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCS, LPCS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup, and actuation of all automatic valves in the flow path to their required positions. This test may be performed by means of any series of sequential, overlapping, or total system steps so that the entire system is tested. This Surveillance also ensures that the HPCS System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage (except for Division III which can be tested in any operational condition) 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 the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5 (continued)

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

SR 3.5.1.6

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.7 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month 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 valve actuation. This prevents an RPV pressure blowdown.

SR 3.5.1.7

A manual actuation of each required ADS valve (those valves removed and replaced to satisfy SR 3.4.4.1) is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods. If performed by method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 16) prior to valve installation. Therefore, this SR is modified by a note that states the surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by method 2), valve

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7 (continued)

OPERABILITY has been demonstrated for all installed ADS valves based upon the successful operation of a test sample of S/RVs.

1. Manual actuation of the ADS valve, with verification of the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or pressure). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate reactor steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the ADS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer.

2. The sample population of S/RVs tested each refueling outage to satisfy SR 3.4.4.1 will be stroked in the relief mode during "as-found" testing to verify proper operation of the S/RV. The successful performance of the test sample of S/RVs provides reasonable assurance that all ADS valves will perform in a similar fashion. After the S/RVs are replaced, the relief-mode actuator of the newly installed S/RVs will be uncoupled from the S/RV, and cycled to ensure that no damage has occurred to the S/RV during transportation and installation. Following cycling, the relief-mode actuator is recoupled and the proper connection to the S/RV lever is independently verified. This verifies that each replaced S/RV will properly perform its intended function.

SR 3.5.1.6 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The STAGGERED TEST BASIS frequency ensures that both solenoids for each ADS valve relief-mode actuator are alternately tested. The frequency of the required relief-mode actuator testing was developed based on the tests required by ASME OM, Part 1, (ref. 16) as

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7 (continued)

implemented by the Inservice Testing Program of Specification 5.5.6. The testing frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.8

This SR ensures that the ECCS RESPONSE TIMES are within limits for each of the ECCS injection and spray subsystems. This SR is modified by a Note which identifies that the associated ECCS actuation instrumentation is not required to be response time tested. Response time testing of the remaining subsystem components is required. This is supported by Reference 14. Response time testing acceptance criteria are included in Reference 15.

ECCS RESPONSE TIME tests are conducted every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

REFERENCES

1. USAR, Section 6.3.2.2.3.
2. USAR, Section 6.3.2.2.4.
3. USAR, Section 6.3.2.2.1.
4. USAR, Section 6.3.2.2.2.
5. USAR, Section 15.2.8.
6. USAR, Section 15.6.4.
7. USAR, Section 15.6.5.
8. 10 CFR 50, Appendix K.
9. USAR, Section 6.3.3.
10. 10 CFR 50.46.
11. USAR, Section 6.3.3.3.
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.
13. USAR, Section 5.2.2.4.1.
14. NEDO-32291-A, "System Analyses for Elimination of Selected Response Time Testing Requirements," January 1994.
15. RBS Technical Requirements Manual.
16. ASME/ANSI OM-1987, "Operation and Maintenance of Nuclear Power Plants, Part 1."

BASES

BACKGROUND (continued)	DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.
APPLICABLE SAFETY ANALYSES	The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.325% by weight of the containment and drywell air per 24 hours at the calculated maximum peak containment pressure (P_a) of 7.6 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks. Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment. During plant operations in other than MODES 1, 2, and 3, the primary containment contains the fission products from a fuel handling accident (FHA), involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) inside the primary containment (Ref. 4), to limit doses at the site boundary to within limits. The primary containment air lock OPERABILITY assures a leak tight fission product barrier during activities with the unit shutdown. Primary containment air locks satisfy Criterion 3 of the NRC Policy Statement.
LCO	As part of the primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA, an FHA involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) or other unanticipated reactivity or water level excursion. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such events.

(continued)

BASES

LCO (continued)	The primary containment air locks are required to be OPERABLE. For each air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from primary containment.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining OPERABLE primary containment air locks in MODE 4 or 5 to ensure a control volume is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs) or during fuel movement of recently irradiated fuel assemblies in the primary containment. Due to radioactive decay, primary containment air locks are only required during fuel handling in the primary containment involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).
ACTIONS	The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door, then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the primary containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the primary containment boundary is

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2.3

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure (Ref. 3), closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the nature of this interlock, and given that the interlock mechanism is only challenged when the primary containment airlock door is opened, this test is only required to be performed upon entering or exiting a primary containment air lock, but is not required more frequently than once per 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other administrative controls.

SR 3.6.1.2.4

A seal pneumatic system test to ensure that pressure does not decay at a rate equivalent to > 1.50 psig for a period of 24 hours from an initial pressure of 90 psig is an effective leakage rate test to verify system performance.

The 18 month Frequency is based on the fact that operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 3.8.
2. 10 CFR 50, Appendix J, Option B.
3. USAR, Table 6.2-1.
4. USAR, 15.7.4.
5. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

BASES

ACTIONS

F.1 and F.2 (continued)

vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. If suspending the OPDRVs would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valves to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valve.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR verifies that the 36 inch primary containment purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of the limits.

The SR is also modified by a Note (Note 1) stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. At times other than MODE 1, 2, or 3 when the purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open (automatic isolation capability would be required by SR 3.6.1.3.4 and SR 3.6.1.3.7).

The SR is modified by a Note (Note 2) stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for pressure control, ALARA, or air quality considerations for personnel entry or for Surveillances, or special testing on the purge system that require the valves to be open (e.g., testing of the containment purge radiation monitors). These primary containment purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other PCIV requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.5

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 4), is required to ensure OPERABILITY. The acceptance criterion for each purge exhaust valve is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$, 7.6 psig. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that which occurs to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.6

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The full closure isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

evaluations of Reference 4 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated power operated or automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J, Option B maximum pathway leakage limits are to be quantified in accordance with Appendix J, Option B). The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 5).

A note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions the Reactor Coolant System is not pressurized and primary containment leakage limits are not required.

SR 3.6.1.3.10

The analyses in References 2 and 3 are based on leakage out of the primary containment that is less than the specified leakage rate. The leakage rate of 50 scfh when pressurized to $\geq P_a$, 7.6 psig, per main steam line provides assurance that the assumptions in the radiological evaluations of Reference 4 are met. Leakage through the valves sealed in each division of MS-PLCS must be $\leq$ 150 scfh per division when tested at $\geq P_a$, 7.6 psig. The leakage rate must be verified to be in accordance with the leakage test requirements of Reference 4, as modified by approved exemptions.

A note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required. The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 5).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.11

Surveillance of hydrostatically tested lines at ≥ 1.1 P_a, 8.36 psig provides assurance that the calculation assumptions of References 2 and 3 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is 1.0 gpm times the total number of hydrostatically tested PCIVs when tested at 1.1 P_a. The combined leakage rates must be demonstrated at the frequency of the leakage test requirements of the Primary Containment Leakage Rate Testing Program (Ref. 5).

A note is added to this SR which states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Low-Low Set (LLS) Values

BASES

BACKGROUND

The safety/relief valves (S/RVs) can actuate either in the relief mode, the safety mode, the Automatic Depressurization System mode, or the LLS mode. In the LLS mode (one of the power actuated modes of operation), a pneumatic operator and mechanical linkage overcome the spring force and open the valve. The valve can be maintained open with valve inlet steam pressure as low as 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure exceeds the safety mode pressure setpoints.

Five of the S/RVs are equipped to provide the LLS function. The LLS logic causes two LLS valves to be opened at a lower pressure than the relief or safety mode pressure setpoints and causes all the LLS valves to stay open longer, such that reopening of more than one S/RV is prevented on subsequent actuations. Therefore, the LLS function prevents excessive short duration S/RV cycles with valve actuation at the relief setpoint.

Each S/RV discharges steam through a discharge line and quencher to a location near the bottom of the suppression pool, which causes a load on the suppression pool wall. Actuation at lower reactor pressure results in a lower load.

APPLICABLE SAFETY ANALYSES

The LLS relief mode functions to ensure that the containment design basis of one S/RV operating on "subsequent actuations" is met (Ref. 1). In other words, multiple simultaneous openings of S/RVs (following the initial opening) and the corresponding higher loads, are avoided. The safety analysis demonstrates that the LLS functions to avoid the induced thrust loads on the S/RV discharge line resulting from "subsequent actuations" of the S/RV during Design Basis Accidents (DBAs). Furthermore, the LLS function justifies the primary containment analysis assumption that multiple simultaneous S/RV openings occur only on the initial actuation for DBAs. Even though five

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

A manual actuation of each required LLS valve (those valves removed and replaced to satisfy SR 3.4.4.1) is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods. If performed by method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 3), prior to valve installation. Therefore, this SR is modified by a note that states the surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by method 2), valve OPERABILITY has been demonstrated for all installed LLS valves based upon the successful operation of a test sample of S/RVs.

1. Manual actuation of the LLS valve, with verification of the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or pressure). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate reactor steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the LLS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer.

2. The sample population of S/RVs tested each refueling outage to satisfy SR 3.4.4.1 will be stroked in the relief mode during "as-found" testing to verify proper operation of the S/RV. The successful performance of the test sample of S/RVs provides reasonable assurance that all LLS valves will perform in a similar fashion. After the S/RVs are replaced, the relief-mode actuator of the newly installed S/RVs will be uncoupled from the S/RV, and cycled to ensure that no damage has occurred to the S/RV during transportation and installation. Following cycling, the relief-mode actuator is recoupled and the proper connection to the S/RV lever is independently verified. This verifies that each replaced S/RV will properly perform its intended function.

The STAGGERED TEST BASIS frequency ensures that both solenoids for each LLS valve relief-mode actuator are alternately tested. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

frequency of the required relief-mode actuator testing was developed based on the tests required by ASME OM, Part 1, (ref. 3) as implemented by the Inservice Testing Program of Specification 5.5.6. The testing frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.6.2

The LLS designed S/RVs 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 automatic LLS function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.4 overlaps this SR to provide complete testing of the safety function.

The 18 month 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 the 18 month 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. GESSAR-II, Appendix 3B, Attachment A, Section 3BA.8.
2. USAR, Section 5.2.2.
3. ASME/ANSI OM-1987, Operation and Maintenance of Nuclear Power Plants, Part 1.

BASES

APPLICABLE SAFETY ANALYSES (continued)	primary containment (Ref.2), to limit doses at the site boundary to within limits. The primary containment performs no active function in response to this event; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage rates assumed in safety analyses. Primary containment satisfies Criterion 3 of the NRC Policy Statement.
LCO	Primary containment OPERABILITY is maintained by providing a contained volume to limit fission product escape following a FHA or other unanticipated reactivity or water level excursion. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Since no credit is assumed for automatic isolation valve closure, and any leakage which would occur prior to valve closure is similarly not accounted for, all penetrations which could communicate gaseous fission products to the environment must remain closed. However, a limited number of primary containment penetration vent and drain valves may remain opened, and the primary containment considered OPERABLE provided the calculated leakage flow rate through the open vent and drain valves is less ≤ 70.2 cfm. Leakage rates specified for the primary containment and air locks, addressed in LCO 3.6.1.1 and LCO 3.6.1.2 are not directly applicable during the shutdown conditions addressed in this LCO.

(continued)

BASES (continued)

APPLICABILITY

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining an OPERABLE primary containment in MODE 4 or 5 to ensure a control volume, is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the primary containment. Due to radioactive decay, the primary containment is only required to be OPERABLE during fuel handling involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1.

ACTIONS

A.1 and A.2

In the event that primary containment is inoperable, action is required to immediately suspend activities that represent a potential for releasing radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.10.1

This SR verifies that each primary containment penetration that could communicate gaseous fission products to the environment during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive gases outside of the primary containment boundary is within design limits. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Single isolation barriers that meet this criterion are a closed and de-activated power operated or automatic valve, a closed manual valve, a blind flange, or equivalent. This does not preclude the use of two active (ie, power operated and/or automatic) valves in the closed position for a given penetration. This SR does not require any testing or valve manipulation.

(continued)

BASES

LCO (continued)	auxiliary building, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.
APPLICABILITY	In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY. In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during movement of irradiated fuel assemblies in the fuel building. The fuel building OPERABILITY during recently irradiated fuel handling is addressed in LCO 3.6.4.7, "Fuel Building Ventilation Systems-Fuel Handling."
ACTIONS	<u>A.1</u> If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal. <u>B.1 and B.2</u> If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.4 and SR 3.6.4.1.6

The SGT System exhausts the shield building annulus and auxiliary building atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the shield building annulus and auxiliary building that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the shield building annulus and auxiliary building to ≥ 0.5 and ≥ 0.25 inches of vacuum water gauge in ≤ 18.5 and ≤ 34.5 seconds, respectively. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.6 demonstrates that each SGT subsystem can maintain ≥ 0.5 and ≥ 0.25 inches of vacuum water gauge for 1 hour. The 1 hour test period allows shield building annulus and auxiliary building to be in thermal equilibrium at steady state conditions. Therefore, these two tests are used to ensure the integrity of this portion of the secondary containment boundary. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 15.6.5.

2. USAR, Section 15.7.4.

BASES

APPLICABILITY (continued)

such as during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical core within the previous 24 hours). Moving recently irradiated fuel assemblies in the Primary Containment is addressed adequately in LCO 3.6.1.10, "Primary Containment-Shutdown."

Moving recently irradiated fuel assemblies in the fuel building will require only the FBIDs associated with the fuel building to be OPERABLE.

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when the need for secondary containment isolation is indicated.

The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCID or FBID. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIDs or FBIDs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCID or FBID.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCID or one FBID inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criteria are a closed and de-activated automatic damper, a closed manual damper or a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to the applicable isolation boundary. This Required Action must be completed within the 8 hour Completion

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

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

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

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. When applicable, movement of recently irradiated fuel assemblies in the fuel building must be immediately suspended. Suspension of this activity shall not preclude completion of movement of a component to a safe position.

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

Verifying the isolation time of each required power operated automatic SCID and FBID is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIDs and FBIDs will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is 92 days.

(continued)

BASES (continued)

APPLICABILITY

Regardless of the plant operating MODE, anytime recently irradiated fuel is being handled there is the potential for a FHA and the fuel building OPERABILITY is required to mitigate the consequences.

ACTIONS

A.1

With the fuel building inoperable the plant must be brought to a condition in which the LCO does not apply since it is incapable of performing its required accident mitigation function. To achieve this, handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical core within the previous 24 hours) must be suspended immediately. Suspension shall not preclude completion of fuel movement to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.5.1

This SR ensures that the fuel building boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to fuel building vacuum variations during the applicable MODES and the low probability of a FHA occurring between surveillances.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal fuel building vacuum condition.

SR 3.6.4.5.2 and SR 3.6.4.5.3

Verifying that fuel building equipment hatches are installed and access doors are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the fuel building will not occur. Maintaining fuel building OPERABILITY requires verifying each door in the access opening is closed, except when the access opening is being used for entry and exit.

(continued)

BASES

ACTIONS

A.1 (continued)

drywell is inoperable is minimal. Also, the Completion Time is the same as that applied to inoperability of the primary containment in LCO 3.6.1.1, "Primary Containment–Operating."

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.1

The seal air flask pressure is verified to be at ≥ 75 psig every 7 days to ensure that the seal system remains viable. It must be checked because it could bleed down during or following access through the personnel door. The 7 day Frequency has been shown to be acceptable through operating experience and is considered adequate in view of the other indications available to operations personnel that the seal air flask pressure is low.

SR 3.6.5.1.2

A seal pneumatic system test to ensure that pressure does not decay at a rate equivalent to > 20.0 psig for a period of 24 hours from an initial pressure of 75 psig is an effective leakage rate test to verify system performance. The 18 month 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 the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.2.5 (continued)

system pressure does not decay at an unacceptable rate. The air lock seal will support drywell OPERABILITY down to a pneumatic pressure of 75 psig. Since the air lock seal air flask pressure is verified in SR 3.6.5.2.2 to be ≥ 75 psig, a decay rate ≤ 20.0 psig over 24 hours is acceptable. The 24 hour interval is based on engineering judgment, considering that there is no postulated DBA where the drywell is still pressurized 24 hours after the event. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage when the air lock OPERABILITY is not required. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix J.
2. USAR, Chapters 6 and 15.

B 3.7 PLANT SYSTEMS

B 3.7.1 Standby Service Water (SSW) System and Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The SSW System is designed to provide cooling water for the removal of heat from unit auxiliaries, such as Residual Heat Removal (RHR) System heat exchangers, standby diesel generators (DGs), HPCS DG, and room coolers for Emergency Core Cooling System equipment required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The SSW System also provides cooling to unit components, as required, during normal shutdown and reactor isolation modes. During a DBA, the equipment required for normal operation only is isolated from the SSW System, and cooling is directed only to safety related equipment.

The SSW System consists of two independent cooling water headers (subsystems A and B), and their associated pumps, piping, valves, and instrumentation. The two SSW pumps on each supply header are sized to provide sufficient cooling capacity to support the required safety related systems during safe shutdown of the unit following a loss of coolant accident (LOCA). Subsystems A and B service equipment in SSW Divisions 1 and 2, respectively. Additionally, the two redundant systems merge to supply the HPCS diesel generator jacket water cooler and the HPCS pump room unit cooler.

The UHS consists of one 200% cooling tower and one 100% capacity water storage basin. The basin is sized such that sufficient water inventory is available to provide heat removal capability to safely shut down the plant and to maintain it in a cold shutdown condition for a 30 day period with no external makeup water source available (Regulatory Guide 1.27, Ref. 1). This assumes the failure of Division II at the beginning of the accident. If failure does not occur, actions are required to ensure long term availability of the UHS. Makeup water sources are available when both Divisions of SSW System are operating and for post DBA system leakage (see USAR section 9.2.5 for additional details). The UHS uses five vaneaxial fans for each of four tower cells in an induced draft system arrangement. Each of the four tower cells is powered by either Standby Diesel Generator A or B (Division 1 or 2). Two operating cells are sufficient for safe shutdown. Normal makeup for the UHS basin is manually controlled and provided through the Makeup Water Treatment System by plant makeup wells.

Cooling water is pumped from the cooling tower basin by the four SSW pumps to the essential components through the two main supply headers (subsystems A and B). After removing

(continued)

BASES

BACKGROUND (continued)	heat from the components, the water is discharged to the cooling tower where the heat is rejected through direct contact with ambient air. Subsystems A and B supply cooling water to equipment required for a safe reactor shutdown. Additional information on the design and operation of the SSW System and UHS along with the specific equipment for which the SSW System supplies cooling water is provided in the USAR, Section 9.2 and the USAR, Table 9.2-15 (Refs. 2 and 3, respectively). The SSW System is designed to withstand a single active or passive failure, coincident with a loss of offsite power, without losing the capability to supply adequate cooling water to equipment required for safe reactor shutdown. Following a DBA or transient, the SSW System will operate automatically without operator action. Manual initiation of supported systems (e.g., suppression pool cooling) is, however, performed for long term cooling operations.
APPLICABLE SAFETY ANALYSES	The UHS consists of one 200% cooling tower and one 100% capacity water storage basin. The basin is sized such that sufficient water inventory is available to provide heat removal capability to safely shut down the plant and to maintain it in a cold shutdown condition for a 30 day period with no additional makeup water source available (Ref. 1) (see USAR section 9.2.5 for additional information concerning inventory requirement). The ability of the SSW System to support long term cooling of the reactor or containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the USAR, Sections 9.2, 6.2.1, and Chapter 15, (Refs. 2, 4, and 5, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA. The SSW System provides cooling water for the RHR suppression pool cooling mode to limit suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its intended function of limiting the release of radioactive materials to the environment following a LOCA. The SSW System also provides cooling to other components assumed to function during a LOCA. Also, the ability to provide onsite emergency AC power is dependent on the ability of the SSW System to cool the DGs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The safety analyses for long term containment cooling were performed, as discussed in the USAR, Sections 6.2.1 and 6.2.2 (Refs. 4 and 6, respectively), for a LOCA, concurrent with a loss of offsite power, and minimum available DG power. The worst case single failure affecting the performance of the SSW System is the failure of one of the two standby DGs, which would in turn affect one SSW subsystem. If failure does not occur, actions are required to ensure long term availability of the UHS (see USAR section 9.2.5 for additional details). The SSW flow assumed in the analyses is 5800 gpm per pump to the RHR heat exchanger (USAR, Table 6.2-2, Ref. 7). Reference 2 discusses SSW System performance during these conditions. The SSW System, together with the UHS, satisfy Criterion 3 of the NRC Policy Statement.
LCO	The OPERABILITY of subsystem A (Division 1) and subsystem B (Division 2) of the SSW System is required to ensure the effective operation of the RHR System in removing heat from the reactor, and the effective operation of other safety related equipment during a DBA or transient. Requiring both subsystems to be OPERABLE ensures that either subsystem A or B will be available to provide adequate capability to meet cooling requirements of the equipment required for safe shutdown in the event of a single failure. A subsystem is considered OPERABLE when: a. The associated pumps are OPERABLE; and b. The associated piping, valves, instrumentation, and controls required to perform the safety related function are OPERABLE. OPERABILITY of the UHS is based on a maximum water temperature of 88°F with a minimum basin water level at or above elevation 111 ft 10 inches mean sea level (equivalent to an indicated level of ≥ 78%) and four OPERABLE cooling tower fan cells. The isolation of the SSW System to components or systems may render those components or systems inoperable, but may not affect the OPERABILITY of the SSW System.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.2 Control Room Fresh Air (CRFA) System

BASES

BACKGROUND

The CRFA System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the CRFA System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems for treatment of recirculated air or outside supply air. Each subsystem consists of a demister, an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork and dampers. Demisters remove water droplets from the airstream. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

In addition to the safety related standby emergency filtration function, parts of the CRFA System are operated to maintain the control room environment during normal operation. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CRFA System automatically switches to the isolation mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and control room air flow is recirculated and processed through either of the two filter subsystems.

The CRFA System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA, per the requirements of GDC 19 and 10CFR50.67. CRFA System operation in maintaining the control room habitability is discussed in the USAR, Sections 6.4.1 and 9.4.1 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CRFA System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the USAR, Chapters 6

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	and 15 (Refs. 3 and 4, respectively). The isolation mode of the CRFA System is assumed to operate following a loss of coolant accident, main steam line break, fuel handling accident, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of outside or recirculated air from the control room. The CRFA System satisfies Criterion 3 of the NRC Policy Statement.
LCO	Two redundant subsystems of the CRFA System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in a failure to meet the dose requirements of GDC 19 and 10CFR50.67 in the event of a DBA. The CRFA System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated: a. Fan is OPERABLE; b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.
APPLICABILITY	In MODES 1, 2, and 3, the CRFA System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release. In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRFA System

(continued)

BASES

APPLICABILITY (continued)

OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with a potential for draining the reactor vessel (OPDRVs); and

b. During the movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

A.1

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

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CRFA subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the

(continued)

BASES

ACTIONS

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

fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of recently irradiated fuel assemblies in the primary containment or fuel building or during OPDRVs, if the inoperable CRFA subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRFA subsystem may be placed in the emergency mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

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

If applicable, movement of recently irradiated fuel assemblies in the primary containment or fuel building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

D.1

If both CRFA subsystems are inoperable in MODE 1, 2, or 3, the CRFA System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

During movement of recently irradiated fuel assemblies in the primary containment or fuel building or during OPDRVs, with two CRFA subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that a subsystem in a standby mode starts on demand from the control room and continues to operate with flow through the HEPA filters and charcoal adsorbers. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized to demonstrate the function of the system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

(continued)

BASES (continued)

REFERENCES

1. USAR, Section 6.4.1.
2. USAR, Section 9.4.1.
3. USAR, Chapter 6.
4. USAR, Chapter 15.
5. Regulatory Guide 1.52, Revision 2, March 1978.
6. 10CFR50.67.

BASES (continued)

LCO

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

The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, chillers, compressors, ductwork, dampers, and associated instrumentation and controls.

APPLICABILITY

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

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

a. During operations with a potential for draining the reactor vessel (OPDRVs) and;

b. During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning

(continued)

BASES

ACTIONS

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

If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

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

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

If applicable, movement of recently irradiated fuel assemblies in the primary containment and fuel building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

E.1 and E.2

During movement of recently irradiated fuel assemblies in the primary containment or fuel building or during OPDRVs if the Required Action and associated Completion Time of Condition B is not met, action must be taken to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, handling of recently irradiated fuel in the primary containment or fuel building must be suspended immediately. Suspension of these activities shall

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.6 Fuel Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pool and upper containment fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. A general description of the spent fuel storage pool and upper containment fuel storage pool design is found in the USAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Section 15.7.4 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG-0800, Section 15.7.4, Ref. 3) of the 10 CFR 50.67 (Ref. 4) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.183 (Ref. 5). The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto stored fuel bundles. The consequences of a fuel handling accident inside the fuel building and inside primary containment are documented in Reference 2. The water levels in the spent fuel storage pool and upper containment fuel storage pool provide for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident. The fuel pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

REFERENCES	1.	USAR, Section 9.1.2.
	2.	USAR, Section 15.7.4.
	3.	NUREG-0800, Section 15.7.4, Revision 1, July 1981.
	4.	10 CFR 50.67.
	5.	Regulatory Guide 1.183.

BASES

ACTIONS (continued)

A.2

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours.

This Completion Time assumes sufficient offsite power remains to power the minimum loads needed to respond to analyzed events. In the event one or more division is without offsite power, this assumption is not met. Therefore, the optional Completion Time is specified. Should two (or more) divisions be affected, the 24 hour Completion Time is conservative with respect to the Regulatory Guide assumptions supporting a 24 hour Completion Time for both offsite circuits inoperable. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E distribution system.

The Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The third Completion Time for Required Action A.2 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 14 days. This situation could lead to a total of 17 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 14 days (for a total of 31 days) allowed prior to complete restoration of the LCO. The 17 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "<u>AND</u>" connector between the 72 hour and 14 day Completion Times means that both

(continued)

BASES

ACTIONS

B.3.1 and B.3.2 (continued)

is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those DG(s).

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the Condition Report Program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable time to confirm that the OPERABLE DG(s) are not affected by the same problem as the inoperable DG.

B.4

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. Although Condition B applies to a single inoperable DG, several Completion Times are Specified for this Condition. The first completion time applies to an inoperable Division III DG. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period. This Completion Time begins only "upon discovery of an inoperable Division III DG" and, as such, provides an exception to the normal "time zero" for beginning the allowed outage time "clock" (i.e., for beginning the clock for an inoperable Division III DG when Condition B may have already been entered for another equipment inoperability and is still in effect).

The second Completion Time (14 days) applies to an inoperable Division I or Division II DG and is risk-informed allowed out-of-service time (AOT) based on plant specific risk analysis. The extended AOT would typically be use for voluntary planned maintenance or inspections but can also be used for corrective maintenance. However, use of the extended AOT for voluntary planned maintenance should be limited to once within an operating cycle (18 months) for each DG (Division I and Division II). Additional contingencies are to be in place for any extended AOT duration (greater than 72 hours and up to 14 days) as follows:

1. An DG extended AOT will not be entered for voluntary planned maintenance purposes if severe weather conditions are expected.

2. The condition of the offsite power supply and switchyard, including transmission lines and ring bus breakers, will be evaluated.
3. No elective maintenance will be scheduled within the switchyard that would challenge offsite power availability during the proposed extended DG AOT.
4. Operating crews will be briefed on the DG work plan, with consideration given to key procedural actions that would be required in the event of a LOOP or SBO.
5. High pressure injection systems will not be taken OOS for maintenance while DG Division I or II is out of service for extended maintenance.

The third Completion Time for Required Action B.4 established a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This situation could lead to a total of 17 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 20 days) allowed prior to complete restoration of the LCO. The 17 day Completion Time provides a limit on the time allowed in a specified

(continued)

BASES

ACTIONS

B.4 (continued)

condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the Completion Times means that the three Completion Times apply simultaneously, and the most restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This exception results in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered.

C.1 and C.2

Required Action C.1 addresses actions to be taken in the event of concurrent failure of redundant required features. Required Action C.1 reduces the vulnerability to a loss of function. The rationale for the 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety divisions are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are designed with redundant safety related divisions (i.e., single division systems are not included in the list, although, for this Required Action, Division III is considered redundant to Division I and II ECCS. Additionally, the Division III powered SSW pump 2C is considered redundant to SSW pumps 2B and 2D powered from Division II). Redundant required features failures consist of any of these features that are inoperable, because any inoperability is on a division redundant to a division with inoperable offsite circuits.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load while maintaining a specified margin to the overspeed trip. The referenced load for DG 1A is the 917.5 kW low pressure core spray pump; for DG 1B, the 462.2 kW residual heat removal (RHR) pump; and for DG 1C the 1995 kW HPCS pump. The Standby Service Water (SSW) pump values are not used as the largest load since the SSW supplies cooling to the associated DG. If this load were to trip, it would result in the loss of the DG. As required by IEEE-308 (Ref. 13), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower. For the River Bend Station the lower value results from the first criteria. The 18 month frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9).

This SR has been modified by two Notes. (Note 1 is not applicable to DG1C) The reason for Note 1 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. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, Note 2 requires that, if synchronized to offsite power, testing be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the DG could experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load, i.e., maximum expected accident load, without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG does not trip upon loss of the load. These acceptance criteria provide DG damage protection. While the DG is not expected to experience this transient during an event and continue to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9) and is intended to be consistent with expected fuel cycle lengths.

This SR has been modified by a Note. (Note is not applicable to DG1C) The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbation 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. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for DG 1A and DG 1B. For DG 1C, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by an immersion heater and circulates through the system by natural circulation. (Note 2 is not applicable to DG1C) The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (10 seconds for DG 1A and DG 1B and 13 seconds for DG 1C) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

The Frequency of 18 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for DG 1A and DG 1B. For DG 1C, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by an immersion heater and circulates through the system by natural circulation. (Note 2 is not applicable to DG1C) The reason for Note 2 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. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.13

This Surveillance demonstrates that DG non-critical protective functions (e.g., high jacket water temperature)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

are bypassed on an ECCS initiation test signal and critical protective functions trip the DG to avert substantial damage to the DG unit. The non-critical trips are bypassed during DBAs and provide alarms on abnormal engine conditions. These alarms provide the operator with necessary information to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 18 month Frequency is based on engineering judgment, taking into consideration 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 at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. (The Note is not applicable to DG1C) The reason for the Note is that performing the Surveillance removes a required DG from service. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours-22 hours of which is at a load

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and load transfer from the respective DG to each required offsite power source can be made and that the respective DG can be returned to ready-to-load status when offsite power is restored. It also ensures that the undervoltage logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready-to-load status when the DG is at rated speed and voltage, the output breaker is open and can receive an auto-close signal on bus undervoltage, and the load sequence timers are reset.

Portions of the synchronization circuit are associated with the DG and portions with the respective offsite circuit. If a failure in the synchronization requirement of the Surveillance occurs, depending on the specific affected portion of the synchronization circuit, either the DG or the associated offsite circuit is declare inoperable.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration plant conditions required to perform the Surveillance.

This SR is modified by a Note. (The Note is not applicable to DG1C) The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions is not compromised as the result of testing. Interlocks to the LOCA sensing circuits cause the DG to automatically reset to ready-to-load operation if an ECCS initiation signal is received during operation in the test mode. Ready-to-load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.13. The intent in the requirement associated with SR 3.8.1.18.b is to show that the emergency loading is not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8); takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

This SR has been modified by a Note. (The Note is not applicable to DG1C) The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.18

Under accident conditions, loads are sequentially connected to the bus by the load sequencing logic. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the bus power supply due to high motor starting currents. The 10% load sequence time tolerance ensures that sufficient time exists for the bus power supply to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. (Note that this surveillance requirement pertains only to the load sequence timer itself, and not to the interposing logic which comprises the remainder of the circuit.) Reference 2 provides a summary of the automatic loading of ESF buses.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2); takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. (The Note is not applicable to DG1C) The reason for the Note is that performing the Surveillance during these MODES would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.18 (continued)

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.12, during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by two Notes. (Note 2 is not applicable to DG1C) The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for DG 1A and DG 1B. For DG 1C, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by an immersion heater and circulates through the system by natural circulation. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources–Shutdown

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

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

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

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

During MODES 1, 2, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that

(continued)

BASES

LCO (continued)

nonessential loads, is a required function for DG OPERABILITY. In addition, proper load sequence operation is an integral part of offsite circuit and DG OPERABILITY since its inoperability impacts the ability to start and maintain energized any loads required OPERABLE by LCO 3.8.10.

It is acceptable for divisions to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required AC electrical power distribution subsystems.

As described in Applicable Safety Analyses, in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty.

APPLICABILITY

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

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

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

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

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

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

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

(continued)

BASES

ACTIONS (continued)

A.1

An offsite circuit is considered inoperable if it is not available to one required ESF division. If two or more ESF 4.16 kV buses are required per LCO 3.8.10, division(s) with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By the allowance of the option to declare required features inoperable which are not powered from offsite power, appropriate restrictions can be implemented in accordance with the required feature(s) LCOs' ACTIONS. Required features remaining powered from offsite power (even though that circuit may be inoperable due to failing to power other features) are not declared inoperable by this Required Action.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required divisions, the option still exists to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the primary containment or fuel building, and activities that could potentially result in inadvertent draining of the reactor vessel.

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

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.4.6 (continued)

the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied. Momentary transients that are not attributable to charger performance do not invalidate this test.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.7

A battery service test is a special test of the battery's capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests not to exceed 18 months.

This SR is modified by two Notes. Note 1 allows the once per 60 months performance of SR 3.8.4.8 in lieu of SR 3.8.4.7. This substitution is acceptable because the battery performance test (SR 3.8.4.8) represents a more severe test of battery capacity than the battery service test (SR 3.8.4.7). Because both the battery service test and the battery performance test involve battery capacity determination, complete battery replacement invalidates the previous performance of these surveillance requirements. In addition to requiring the re-performance of both of these surveillance tests prior to declaring the battery OPERABLE, complete battery replacement also resets the 60 month time period used for substitution of the service test by the performance test. For this reason, substitution is acceptable for performance testing conducted within the first two years of service of a new battery as required by Reference 8. The reason for Note 2 is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources- Shutdown

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

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

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

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

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

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One DC electrical power subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the division, associated with Division

(continued)

BASES

LCO (continued)	I or II onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems–Shutdown" is required to be OPERABLE. Similarly, when the High Pressure Core Spray (HPCS) system is required to be OPERABLE, the Division III DC electrical power subsystem associated with the Division III onsite Class 1E DC electrical power distribution subsystem required to be OPERABLE by LCO 3.8.10 is required to be OPERABLE. In addition to the preceding subsystems required to be OPERABLE, a Class 1E battery or battery charger and the associated control equipment and interconnecting cabling capable of supplying power to the remaining Division I or II onsite Class 1E DC electrical power distribution subsystem(s), when portions of both Division I and II DC electrical power distribution subsystems are required to be OPERABLE by LCO 3.8.10. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).

APPLICABILITY

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

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

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

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

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

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

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

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

If more than one DC distribution subsystem is required according to LCO 3.8.10, the DC subsystems remaining OPERABLE with one or more DC power sources inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated DC power source(s) inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies, and any activities that could result in inadvertent draining of the reactor vessel).

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

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

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters– Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. There are two inverters for Division I and two inverters for Division II, making a total of four inverters. The function of the inverter is to provide AC electrical power to the vital buses. The inverters are powered from both AC and DC sources.

Specific details on inverters can be found in the USAR, Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the USAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to portions of the ESF instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining electrical power sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC or all onsite AC electrical power; and

b. A worst case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The inverters ensure the availability of AC electrical power for the instrumentation for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the ESF instrumentation and controls is maintained. The four battery powered inverters ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized.

Division I and II each have two totally redundant inverters installed (total of four). Through the use of a manual transfer switch, for either Division, either of these two divisional inverters may be aligned to supply power to the 120 Volt Vital Bus. Thus, only one of these inverters per Division is required to be OPERABLE.

OPERABLE inverters require that the associated vital bus is powered by the inverter via inverted DC voltage from the required Class 1E battery or from an internal AC source via a rectifier with the battery available as backup, with the output within the design voltage and frequency tolerances.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 4 and 5 are covered in the Bases for LCO 3.8.8, "Inverters – Shutdown."

ACTIONS

With a required inverter inoperable, its associated AC vital bus is inoperable if not energized from one of its Class 1E voltage sources. LCO 3.8.9 addresses this action; however, pursuant to LCO 3.0.6, these actions would not be entered even if the AC vital bus were de-energized. Therefore, the ACTIONS are modified by a Note stating that ACTIONS for LCO 3.8.9 must be entered immediately. This ensures the vital bus is re-energized within 8 hours.

(continued)

BASES

ACTIONS (continued)

A.1

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service or align an OPERABLE inverter to the Vital Bus. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the plant is exposed because of the inverter inoperability. This risk has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems that such a shutdown might entail. When the AC vital bus is powered from one of its Class 1E sources, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters – Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient accident analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to portions of the ESF instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

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

The OPERABILITY of the minimum inverters to each AC vital bus during MODES 4 and 5, and during movement of recently irradiated fuel assemblies in the primary containment or fuel building ensures that:

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

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

c. Adequate power is available to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The inverters were previously identified as part of the Distribution System and, as such, satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO	The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABLE inverters require the associated AC vital bus be powered by the inverter through inverted DC voltage from the required Class 1E battery, or from an internal AC source via a rectifier with the battery available as backup. This ensures the availability of sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).
APPLICABILITY	The inverters required to be OPERABLE in MODES 4 and 5 and also any time during movement of recently irradiated fuel assemblies in the primary containment or fuel building provide assurance that: a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel; b. Systems needed to mitigate a fuel handling accident are available; c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition. Inverter requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.
ACTIONS	The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

(continued)

BASES

ACTIONS (continued)

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

If two divisions are required by LCO 3.8.10, "Distribution Systems–Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required feature(s) to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By the allowance of the option to declare required feature(s) inoperable with the associated inverter(s) inoperable, appropriate restrictions are implemented in accordance with the affected required feature(s) of the LCOs' ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the primary containment and fuel building, and any activities that could result in inadvertent draining of the reactor vessel).

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

Notwithstanding performance of the above conservative Required Actions, the unit is still without sufficient AC vital power sources to operate in a safe manner. Therefore, action must be initiated to restore the minimum required AC vital power sources and continue until the LCO requirements are restored.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers

(continued)

ACTIONS

C.1 (continued)

b. The potential for decreased safety when requiring entry into numerous applicable Conditions and Required Actions for components without DC power while not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 10 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC division could again become inoperable, and DC distribution could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This allowance results in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential of failing to meet the LCO indefinitely.

D.1 and D.2

If the inoperable electrical power distribution system cannot be restored to OPERABLE status within the associated Completion Times, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems–Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems–Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

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

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

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

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO	Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of Technical Specifications' required systems, equipment, and components - both specifically addressed by their own LCOs, and implicitly required by the definition of OPERABILITY. Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).
APPLICABILITY	The AC, DC, and AC vital bus electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel building provide assurance that: a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel; b. Systems needed to mitigate a fuel handling accident are available; c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition. The AC, DC, and AC vital bus electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.9.

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

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

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

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal–shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR-SDC inoperable, which results in taking the appropriate RHR-SDC ACTIONS.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Additionally, prior to fuel reload, all control rods must be fully inserted to minimize the probability of an inadvertent criticality. Control rod position satisfies Criterion 3 of the NRC Policy Statement.
LCO	All control rods must be fully inserted during applicable refueling conditions to minimize the probability of an inadvertent criticality during refueling.
APPLICABILITY	During MODE 5, loading fuel into core cells with control rods withdrawn may result in inadvertent criticality. Therefore, the control rods must be inserted before loading fuel into a core cell. All control rods must be inserted before loading fuel to ensure that a fuel loading error does not result in loading fuel into a core cell with the control rod withdrawn. In MODES 1, 2, 3, and 4, the reactor pressure vessel head is on, and no fuel loading activities are possible. Therefore, this Specification is not applicable in these MODES.
ACTIONS	<u>A.1</u> With all control rods not fully inserted during the applicable conditions, an inadvertent criticality could occur that is not analyzed in the USAR. All fuel loading operations must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.3.1</u> During refueling, to ensure that the reactor remains subcritical, all control rods must be fully inserted prior to and during fuel loading. Periodic checks of the control rod position ensure this condition is maintained.

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level –Irradiated Fuel

BASES

BACKGROUND	The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 23 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the upper containment pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 50.67 limits, as provided by the guidance of Reference 3.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows an effective decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the total fuel rod I-131 inventory (Ref. 1). Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4). While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.6.1 (continued) The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.
REFERENCES	1. Regulatory Guide 1.183. 2. USAR, Section 15.7.4. 3. NUREG-0800, Section 15.7.4. 4. 10 CFR 50.67.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Reactor Pressure Vessel (RPV) Water Level–New Fuel or Control Rods

BASES

BACKGROUND	The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 50.67 limits, as provided by the guidance of Reference 3.
APPLICABLE SAFETY ANALYSES	During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows an effective decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the total fuel rod I-131 inventory (Ref. 1). Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4). The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.7.1 (continued) operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.
REFERENCES	1. Regulatory Guide 1.183. 2. USAR, Section 15.7.4. 3. NUREG-0800, Section 15.7.4. 4. 10 CFR 50.67.