10 CFR 50.90

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Serial: RNP-RA/00-0017

FEB 25 2000

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

SUPPLEMENT 1 TO TECHNICAL SPECIFICATIONS
CHANGE REQUEST FOR ULTIMATE HEAT SINK (UHS)

Sir or Madam:

This letter provides Supplement 1 to Carolina Power & Light (CP&L) Company's H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 Technical Specification change request for the Ultimate Heat Sink, provided by letter dated May 27, 1999. The requested change increases the maximum allowable service water temperature permitted by Surveillance Requirement from 95°F to 97°F. This letter also provides additional information discussed with the NRC in a meeting held between CP&L and the NRC on February 4, 2000, and subsequent teleconference.

Attachment I provides an affidavit as required by 10 CFR 50.30(b).

Attachment II provides clarifications to the initial technical specification change request identified as necessary in a meeting conducted between CP&L and the NRC on February 4, 2000. The proposed changes provide additional information and do not change the no significant hazards consideration or the environmental impact consideration provided with the initial technical specification change request.

Attachment III provides revised pages for the markup of current TS and Bases. Attachment IV provides a complete set of retyped pages for the TS and Bases including pages revised as a result of this supplement.

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In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of this letter with attachments.

CP&L requests that this proposed change be reviewed and approved by May 1, 2000, to permit implementation of the change this summer. The administrative changes to the Technical Specifications proposed by this supplement have no impact on the previous conclusion that the proposed Technical Specification change does not involve a significant hazards consideration.

If you have any questions concerning this matter, please contact Mr. Harold Chernoff at (843) 857-1437.

Very truly yours,

R. L. Warden
Manager - Regulatory Affairs

EBS/ebs

Attachments

I. Affidavit
II. Provides Clarifications To The Initial Technical Specification Change Request
III. Revised Markup Of Current TS And Bases
IV. Retyped Technical Specifications And Bases

c: Mr. Max K. Batavia, Chief, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, USNRC, Region II
Mr. R. Subbaratman, USNRC
USNRC Resident Inspector, HBRSEP
Attorney General (SC) (w/out Enclosures)

Affidavit

State of South Carolina
County of Darlington

J. W. Moyer, having been first duly sworn, did depose and say that the information contained in letter RNP-RA/00-0017 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

JMoyer

Sworn to and subscribed before me

this 25th day of February 2000

(Seal) Albert L Garron
Notary Public for South Carolina

My commission expires: March 22nd 2005

CLARIFICATIONS TO THE
TECHNICAL SPECIFICATION CHANGE REQUEST FOR
ULTIMATE HEAT SINK

On February 4, 2000, Carolina Power and Light (CP&L) Company met with the NRC regarding the Ultimate Heat Sink (UHS) Technical Specification (TS) submittal. During that meeting, CP&L agreed to provide editorial corrections to the May 27, 1999, UHS TS change request, administrative changes to the proposed TS change, and clarifications regarding the submittal.

1. Editorial Corrections

The following are editorial corrections to information provided in Attachment II of the UHS TS Submittal:

a. Page 9 of Attachment II to the May 27, 1999, Request for Technical Specification Change (initial UHS submittal) indicates that WCAP 8264, "Westinghouse Mass and Energy Release Data for Containment Design" was used to determine mass and energy release for the new containment analyses for a Loss of Coolant Accident (LOCA). WCAP 10325, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," was used for that purpose.

b. On page 37 of Attachment II to the initial UHS submittal, the Table 3 column heading for "Energy" (columns 3 and 5) should show that these values are in thousands.

c. On page 42 of Attachment II to the initial UHS submittal, the Table 4 parameter for Reactor Containment Air Recirculation Fan Coolers Heat Removal Rate should show that the heat removal value for 200°F is 7459.49 BTU/sec rather than 7456.49 BTU/sec.

d. Table 4, entitled "Large Break LOCA Containment Response Analysis (COCO) Parameters," provides the input parameters for the containment response analyses (COCO) performed for limiting and non limiting LOCAs and Main Steam Line Breaks (MSLBs). On page 42 of Attachment II to the initial UHS submittal, the Table 4 parameter for Containment Spray Delay Time Without Offsite Power shows that this is "seconds after setpoint actuation." This should be seconds after event initiation. Similarly, on page 42, the Table 4 parameter for Reactor Containment Air Recirculation Fan Cooler Delay Time Without Offsite Power shows that this is "seconds after setpoint actuation." This should be seconds after event initiation.

   Also, on page 42, the Table 4 parameter for Reactor Containment Air Recirculation Fan Cooler Delay Time With Offsite Power shows a delay time of 30.4 seconds and 35.4 seconds for HVH-1&3 and HVH-2&4 respectively. However, these delay times were not used in the limiting MSLB case. For the limiting MSLB event, the fan coolers are assumed to start upon receipt of a safety injection (SI) signal. This is because there is no fault assumed on the emergency diesel generator (EDG) or vital bus, so the delays associated with EDG start and EDG breaker timer need not be considered.

e. Table 4, entitled "Large Break LOCA Containment Response Analysis (COCO) Parameters," provides the input parameters for the containment response analysis (COCO) performed for LOCA and MSLB. The value for Containment Spray Flow Rate listed in Table 4 (Page 42) is applicable only for the LOCA analysis. The flow rate used for the MSLB containment response analysis was 1933 gpm.

2. Administrative Changes to Proposed Technical Specification Change

The May 27, 1999, UHS Technical Specification (TS) Change Request was based on the Technical Specification amendment in effect at the time of the submittal. The following administrative changes to the proposed TS change are made to incorporate the latest TS amendments and TS Bases revisions and to remove a requirement that is no longer applicable.

a. The markup in the initial submittal used Revision 0 to Bases page B 3.6-26. Subsequent to May 27, 1999, submittal, Revision 10 (implemented on August 27, 1999, and transmitted to the NRC by letter dated September 29, 1999) to the Bases was issued to correctly depict that the initial pressure condition used in the containment analysis was 15.0 psia (0.3 psig). A revised TS markup for Page B 3.6-26 based on Revision 10 (latest) to the Bases is provided in Attachment III.

b. The TS amendment for TS 3.7.8, "Ultimate Heat Sink (UHS)," applicable at the time of the submittal was used to create the proposed TS change. This TS version had a note that made Conditions A and B no longer applicable after September 30, 1998, and was subsequently superseded by a similar change that expired after September 30, 1999. These Conditions provided Required Actions and Completion Times for restoring SW temperature which are no longer applicable. This supplement revises the proposed change for TS 3.7.8 to eliminate Notes 1 and 2 and Conditions A and B to the ACTIONS Table. This is an administrative change since the change removes requirements that are not applicable. Revised TS markup pages, based on the latest amendment to the page (i.e., Amendment 184), and the retyped TS pages are provided in Attachments III and IV respectively.

3. Clarifications

During the February 4, 2000, meeting, CP&L agreed to provide the following clarifying information in a supplement to the May 27, 1999, UHS TS submittal.

a. The MSLB analysis result shows an early peak containment temperature as a result of superheat. This temperature is less than 280°F, the qualification temperature.

Consequently, no thermal analyses were required to qualify components to temperatures higher than the manufacturer's peak qualification temperature. The MSLB component temperature curve is included as Figure 1.

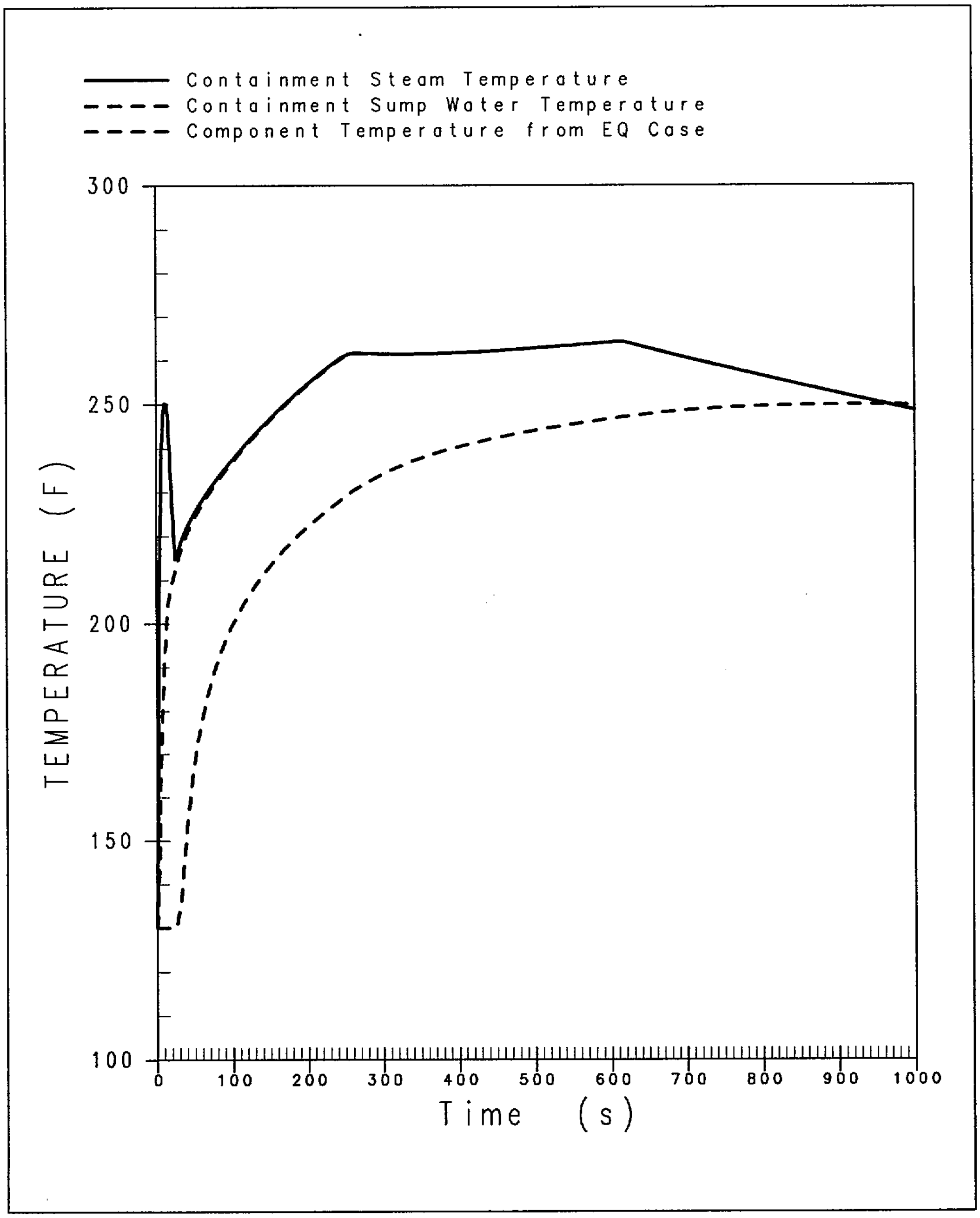
It should be noted that the UFSAR currently has a component qualification temperature of 263°F. In the current MSLB analysis there is a short temperature peak of 350°F. However, because of the short duration of the temperature peak, the surface temperature of components has been determined to not be impacted beyond the current qualification limit.

b. As stated in letter dated May 27, 1999, CP&L's commitment to Inspection and Enforcement Bulletin (IEB) 80-06, "Engineered Safety Features (ESF) Reset Controls," must be modified as a result of the new analyses supporting the Service Water (SW) temperature increase. The submittal further states that the Containment Spray (CS) system actuation circuitry will be modified so that it can be blocked while the CS pump suction is being switched from the Refueling Water Storage Tank following a large break LOCA.

   A permanent modification will be completed to provide this feature during an outage of sufficient duration and conditions (i.e., approximately 5 days in MODE 5). To allow TS change implementation prior to implementation of the permanent modification an interim action will be implemented. Administrative procedures will be revised to require operators to block the CS actuation signal using installed plant equipment outside of the control room. This control scheme meets IEB 80-06 requirements since following block of the CS actuation signal, the associated safety related equipment remains in its emergency mode. Operators will then be able to change the state of actuated equipment while the signal is blocked.

Figure 1

MSLB HZP with Check Valve Failure - Temperature



United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/00-0017
7 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
ULTIMATE HEAT SINK (UHS)

REVISED MARKUP OF CURRENT TECHNICAL SPECIFICATIONS AND BASES PAGES

3.7 PLANT SYSTEMS

3.7.8 Ultimate Heat Sink (UHS)

LCO 3.7.8 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

NOTES

1. ~~Conditions A and B and associated Required Actions and Completion Times shall only be applicable prior to, and on September 30, 1999.~~

2. ~~Condition C and associated Required Actions and Completion Times shall only be applicable after September 30, 1999.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. Service water temperature > 95EF.~~	~~A.1 Restore service water temperature to ≤ 95°F.~~	~~72 hours~~
	~~AND~~	
	~~A.2 Verify service water temperature is ≤ 99°F.~~	~~1 hour~~ ~~AND~~ ~~Once per hour thereafter~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. Required Action and Completion Time of Condition A not met.~~ ~~OR~~ ~~UHS inoperable for reasons other than Condition A.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~
~~C~~. UHS inoperable. A	~~C~~.1 Be in MODE 3. AND ~~C~~.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify water level of UHS is ≥ 218 ft mean sea level.	24 hours
SR 3.7.8.2	Verify service water temperature is ≤ ~~95~~°F.	24 hours

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes designed to predict the resultant pressure and temperature transient. The containment pressure analysis indicates the containment peak pressure for the limiting SLB slightly exceeds the peak pressure for the limiting LOCA (Ref. 1).

The initial pressure condition used in the containment analysis was ~~15~~ 15.7 psia (~~0.3~~ 1.0 psig). This resulted in a maximum peak pressure from a LOCA of ~~40~~ 40.5 psig. The containment analysis (Ref. 1) confirms that this calculated peak containment pressure from the limiting LOCA is the same as the P_a determined in accordance with 10 CFR 50, Appendix J, Option B. The maximum containment pressure resulting from the worst case SLB, ~~40.5~~ 41.9 psig, does not exceed the containment design pressure, 42 psig.

The containment was also designed for an external pressure load equivalent to -3.0 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. Since the UHS is the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs at the time that recirculation begins after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems and RHR are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA and maintaining adequate net positive suction head (NPSH) for the SWS pumps. The UHS at the minimum allowable level of 218 ft MSL provides a 22 day supply of cooling water to the SWS pumps under worst case local meteorological conditions. After 22 days, the minimum NPSH for the SWS pumps is reached when the lake level drops to 210.64 ft MSL. The lake surface area at 210.64 ft MSL is capable of providing decay heat cooling for 97 the plant without exceeding the ~~95~~°F maximum SWS temperature requirement. Therefore, the necessary lake level for adequate NPSH for the SWS pumps is more limiting than the lake surface area necessary for decay heat removal. The 22 day supply of water is based on the lake volume and surface area values provided in References 2 and 3, an evaporation rate of 35 ft^3/sec (Ref. 4) that assumes both Unit 1 (fossil Plant) and Unit 2 operating at 100% power for 6 hours, an 17 evaporation rate of ~~17.27~~ ft^3/sec that assumes Unit 1 in operation and Unit 2 shut down for the remaining 22 day period under maximum evaporation conditions, a head flow of 16 ft^3/sec which is based upon the minimum head flow measured at the Black Creek inlet over the past 30 years (Ref. 5), and a fully open Howell Bunger valve which provides an average flow of 260 ft^3/sec. No credit is taken for natural springs, precipitation or other drainage input into the lake for the 22 day period. The opening and testing of the tainter gates is administratively limited to approximately ~~2~~ 2.5 inches except for flood control measures necessary to protect the integrity of the dam which approximates the capacity of one Howell Bunger valve. A failure of a tainter gate to reclose when the gate is raised ~~2~~ inches or less is bounded by a fully open Howell Bunger valve in the analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The UHS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 22 days following the design basis LOCA without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 97 ~~95~~°F and the level should not fall below 218 ft MSL during normal unit operation.

APPLICABILITY

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

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

ACTIONS

~~Notes 1 and 2 have been added in the ACTIONS to provide a clear expiration date for Conditions A and B and associated Required Actions and Completion Times, and a date that Condition C and its associated Required Actions and Completion Times will become applicable. Prior to midnight October 1, 1999, if the LCO is not met, refer to Conditions A or B and associated Required Actions and Completion Times. On midnight October 1, 1999, and thereafter, refer only to Condition C if the LCO is not met.~~

~~A.1~~

~~When service water temperature is greater than 95°F, it must be restored to ≤ 95°F within 72 hours. This Required Action is necessary to return operation to within the design basis of the Service Water System. The 72 hour Completion Time is acceptable considering the low probability of a Design Basis~~

(continued)

BASES

ACTIONS

~~A.1 and A.2 (continued)~~

~~Accident occurring during this period and allows a reasonable time for diurnal effects to act upon the UHS.~~

~~The service water temperature must be monitored more frequently to ensure service water temperatures stay at or below 99°F so that no loss of function occurs for equipment cooled by the UHS. The Completion Time of 1 hour is reasonable considering the limited time that Required Action A.1 allows the service water temperature limit to be exceeded in conjunction with the generally slow rate of temperature increase experienced from thermal changes in Lake Robinson.~~

~~B.1 and B.2~~

~~If Required Actions A.1 and A.2 and Completion Times are not met or the UHS is inoperable for reasons other than Condition A, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 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.~~

A

~~C~~.1, and ~~C~~.2

If the UHS is inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate long term (22 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ 218 ft MSL.

SR 3.7.8.2

This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the service water temperature is ≤ ~~95~~ 97°F.

REFERENCES

1. UFSAR, Section 9.2.4.
2. UFSAR Section 2.4.6.1.
3. UFSAR Section 2.1.1.2.
4. NUREG-75/024, "Final Environmental Statement Related to the Operation of H. B. Robinson Nuclear Steam-Electric Plant Unit 2," U. S. Nuclear Regulatory Commission, Washington DC 20555, April 1975, page 3-7.
5. USGS Historical Daily Values for Station Number 02130900, Black Creek Near McBee, South Carolina, Years 1960-1993.

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/00-0017
23 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
ULTIMATE HEAT SINK (UHS)

RETYPED TECHNICAL SPECIFICATIONS AND BASES

Table 3.3.2-1 (page 2 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
2. Containment Spray						
a. Manual Initiation	1,2,3,4	2 trains	I	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure						
High High	1,2,3,4	6 (2 sets of 3)	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 10.45 psig	10 psig
3. Containment Isolation						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	2 trains	I	SR 3.3.2.6	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
(3) Containment Pressure						
High High	1,2,3,4	6 (2 sets of 3)	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 10.45 psig	10 psig

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.

Table 3.3.2-1 (page 3 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
4. Steam Line Isolation						
a. Manual Initiation	$1,2^{(e)},3^{(e)}$	1 per steam line	F	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	$1,2^{(e)},3^{(e)}$	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure - High High	$1,2^{(e)},3^{(e)}$	6 (2 sets of 3)	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 10.45 psig	10 psig
d. High Steam Flow in Two Steam Lines	$1,2^{(e)},3^{(e)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with T_{avg} - Low	$1,2^{(e)},3^{(e)(b)}$	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 541.50 °F	543°F
e. High Steam Flow in Two Steam Lines	$1,2^{(e)},3^{(e)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with Steam Line Pressure - Low	$1,2^{(e)},3^{(e)}$	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 605.05 psig	614 psig

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.

(b) Above the T_{avg} - Low interlock.

(c) Less than or equal to a function defined as ΔP corresponding to 41.58% full steam flow below 20% load, and ΔP increasing linearly from 41.58% full steam flow at 20% load to 110.5% full steam flow at 100% load, and ΔP corresponding to 110.5% full steam flow above 100% load.

(d) Less than or equal to a function defined as ΔP corresponding to 37.25% full steam flow between 0% and 20% load and then a ΔP increasing linearly from 37.25% steam flow at 20% load to 109% full steam flow at 100% load.

(e) Except when all MSIVs are closed.

3.6 CONTAINMENT SYSTEMS

3.6.8 Isolation Valve Seal Water (IVSW) System

LCO 3.6.8 The IVSW System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. IVSW system inoperable.	A.1 Restore IVSW system to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify IVSW tank pressure is ≥ 44.6 psig.	12 hours
SR 3.6.8.2	Verify the IVSW tank volume is ≥ 85 gallons.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.8.3	Verify the opening time of each air operated header injection valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.8.4	Verify each automatic valve in the IVSW System actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.8.5	Verify the IVSW dedicated nitrogen bottles will pressurize the IVSW tank to ≥ 44.6 psig.	18 months
SR 3.6.8.6	Verify IVSW seal header flow rate is: a. ≤ 52.00 cc/minute for Header A, b. ≤ 16.50 cc/minute for Header B, c. ≤ 32.50 cc/minute for Header C, and d. ≤ 23.00 cc/minute for Header D.	18 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more bypass valves inoperable.	C.1	Close or isolate bypass valve.	8 hours
	<u>AND</u>		
	C.2	Verify bypass valve is closed or isolated.	Once per 7 days
D. Two valves in the same flow path inoperable.	D.1	Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFRV and bypass valve is ≤ 30 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify the closure time of each MFIV is ≤ 50 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.8 Ultimate Heat Sink (UHS)

LCO 3.7.8 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify water level of UHS is ≥ 218 ft mean sea level.	24 hours
SR 3.7.8.2	Verify service water temperature is ≤ 97°F.	24 hours

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program

This program provides controls for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 40.5 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of the containment air weight per day.

Leakage rate acceptance criteria are:

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

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

BASES

BACKGROUND (continued)

b. The air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Lock";

c. The equipment hatch is closed and sealed; and

d. The Isolation Valve Seal Water (IVSW) sytem is OPERABLE, except as provided in LCO 3.6.8.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the LOCA analyses, it is assumed that the containment is OPERABLE such that, for the LOCA, the release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% per day in the safety analysis at P_a = 40.5 psig (Ref. 2).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1) as L_a = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 40.5 psig following a DBA.

The containment air lock satisfies Criterion 3 of the NRC Policy Statement.

LCO

The containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The air lock is required to be OPERABLE. For the 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 of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in the 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 containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND	The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System. Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.
APPLICABLE SAFETY ANALYSES	Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes designed to predict the resultant pressure and temperature transient. The containment pressure analysis indicates the containment peak pressure for the limiting SLB slightly exceeds the peak pressure for the limiting LOCA (Ref. 1). The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). This resulted in a maximum peak pressure from a LOCA of 40.5 psig. The containment analysis (Ref. 1) confirms that this calculated peak containment pressure from the limiting LOCA is the same as the P_a determined in accordance with 10 CFR 50, Appendix J, Option B. The maximum containment pressure resulting from the worst case SLB, 41.9 psig, does not exceed the containment design pressure, 42 psig. The containment was also designed for an external pressure load equivalent to -3.0 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 130°F. This resulted in a maximum containment air temperature of approximately 274°F. The maximum containment air temperature from a LOCA is approximately 262°F. The environmental qualification temperature limit is 280°F. The containment structural design temperature is 263 °F.

The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature briefly during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 1).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.
LCO	During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the values previously analyzed. As a result, the ability of containment to perform its design function is ensured.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit 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 6 hours and to MODE 5 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.

(continued)

BASES

BACKGROUND

Containment Cooling System (continued)

and incore detector raceway, and outside the secondary shield in the lower areas of containment.

During normal operation, all four fan units may be operating. The fans are normally operated with SW supplied to the cooling coils. The Containment Cooling System, operating in conjunction with the Containment Ventilation system, is designed to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running. The temperature of the SW is an important factor in the heat removal capability of the fan units.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and Containment Cooling System being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 41.9 psig (experienced during a SLB). The analysis shows that the peak containment temperature is approximately 274°F (experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a power

(continued)

APPLICABLE SAFETY ANALYSES (continued)

level of 102% for the most limiting pressure response and 0% for the most limiting temperature response, a single failure of a steam line check valve, and initial (pre-accident) containment conditions of 130°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -3.0 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles.

Containment cooling train performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, is also shown in Reference 4. The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the containment high pressure setpoint to achieving full Containment Cooling System air and cooling water flow.

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 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 3.6.8.1

This SR verifies the IVSW tank has the necessary pressure to provide motive force to the seal water. A pressure ≥ 44.6 psig ensures the containment penetration flowpaths that are sealed by the IVSW System are maintained at a pressure which is at least 1.1 times the calculated peak containment internal pressure (P_a) related to the design bases accident. Verification of the IVSW tank pressure on a Frequency of once per 12 hours is acceptable. This Frequency is sufficient to ensure availability of IVSW. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.6.8.2

This SR verifies the IVSW tank has an initial volume of water necessary to provide seal water to the containment isolation valves served by the IVSW System. An initial volume ≥ 85 gallons ensures the IVSW System contains the proper inventory to maintain the required seal. Verification of IVSW tank level on a Frequency of once per 31 days is acceptable since tank level is continuously monitored by installed instrumentation and will alarm in the control room prior to level decreasing to 85 gallons.

SR 3.6.8.3

This SR verifies the stroke time of each automatic air operated header injection solenoid valve is within limits. The frequency is specified by the Inservice Testing

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND	The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators. One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators. The MSIVs close on a main steam isolation signal generated by either high steam flow coincident with low T_{avg} or with low steam pressure; or high-high containment pressure. The MSIVs fail as is on loss of control or actuation power. A bypass valve is provided around each MSIV to equalize pressure across the valve and to warm up the steam line during unit startup. The bypass valves are motor operated, manually actuated valves, which are normally closed. A description of the MSIVs is found in the UFSAR, Section 10.3 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the UFSAR, Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the UFSAR, Section 15.1.5 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). Furthermore, the design can limit the blowdown through the break that would occur while the MSIVs are closing. This is due to a check valve installed downstream of each MSIV. Upon a failure of an MSIV, the check valve will prevent

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

containment analyses with the exception of closure of the MSIVs for a MSLB at 100% RTP, in which case MSIV closure in 2 seconds is assumed for MSIVs which close in the forward flow direction. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program. The specified Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the specified Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 6.2.
3. UFSAR, Section 15.1.5.
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFRV and bypass valve is ≤ 30 seconds on an actual or simulated actuation signal. The MFRV, and bypass valve closure times are assumed in the accident and containment analyses (Ref. 2). This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code, Section XI (Ref. 3).

The Frequency for this SR is in accordance with the Inservice Testing Program. The specified Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the specified Frequency.

SR 3.7.3.2

This SR verifies that the closure time of each MFIV is ≤ 50 seconds on an actual or simulated actuation signal. The MFIV closure times are assumed in the accident and containment analyses (Ref. 2). This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code, Section XI (Ref. 3).

The Frequency for this SR is in accordance with the Inservice Testing Program. The specified Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the specified Frequency.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The design basis of the CCW System is for one CCW pump and one CCW heat exchanger to accommodate the post loss of coolant accident (LOCA) heat removal loads from the RHR heat exchangers and safety injection pump seals. Should either a required CCW pump or a CCW heat exchanger fail, one of the two standby pumps and the standby heat exchanger provide 100 percent backup.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from RHR entry conditions (T_{cold} < 350°F), to MODE 5 (T_{cold} < 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of service water temperature and the number of CCW and RHR trains operating. One CCW train is sufficient to remove decay heat during subsequent operations with T_{cold} < 200°F. This assumes a maximum service water temperature of 97°F occurring simultaneously with the maximum heat loads on the system.

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The CCW trains are independent of each other to the degree that each CCW pump has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW train powered from an emergency power source is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of CCW powered from an emergency power source must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CCW train is considered OPERABLE when:

a. The required pump and heat exchanger are OPERABLE; and

(continued)

BASES

BACKGROUND (continued)

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the UFSAR, Section 9.2.1 (Ref. 1). The principal safety related function of the SWS is the removal of decay heat from the reactor via the Component Cooling Water (CCW) System.

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is to provide cooling water to those components necessary to remove core decay heat following a design basis LOCA as discussed in the UFSAR, Section 6.2 (Ref. 2). The system is sized to ensure adequate heat removal, based on highest expected temperatures of cooling water, maximum loadings, and leakage allowances. The SWS is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The SWS, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in the UFSAR, Section 5.4.4, (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and RHR System trains that are operating and SW supply temperature. One SWS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum SWS temperature of 97°F occurring simultaneously with maximum heat loads on the system.

The SWS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two SWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An SWS train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. Two SWS pumps are OPERABLE;

b. One SWS booster pump is OPERABLE;

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. Since the UHS is the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs at the time that recirculation begins after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems and RHR are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA and maintaining adequate net positive suction head (NPSH) for the SWS pumps. The UHS at the minimum allowable level of 218 ft MSL provides a 22 day supply of cooling water to the SWS pumps under worst case local meteorological conditions. After 22 days, the minimum NPSH for the SWS pumps is reached when the lake level drops to 210.64 ft MSL. The lake surface area at 210.64 ft MSL is capable of providing decay heat cooling for the plant without exceeding the 97°F maximum SWS temperature requirement. Therefore, the necessary lake level for adequate NPSH for the SWS pumps is more limiting than the lake surface area necessary for decay heat removal. The 22 day supply of water is based on the lake volume and surface area values provided in References 2 and 3, an evaporation rate of 35 ft^3/sec (Ref. 4) that assumes both Unit 1 (fossil Plant) and Unit 2 operating at 100% power for 6 hours, an evaporation rate of 17 ft^3/sec that assumes Unit 1 in operation and Unit 2 shut down for the remaining 22 day period under maximum evaporation conditions, a head flow of 16 ft^3/sec which is based upon the minimum head flow measured at the Black Creek inlet over the past 30 years (Ref. 5), and a fully open Howell Bunger valve which provides an average flow of 260 ft^3/sec. No credit is taken for natural springs, precipitation or other drainage input into the lake for the 22 day period. The opening and testing of the tainter gates is administratively limited to approximately 2.5 inches except for flood control measures necessary to protect the integrity of the dam which approximates the capacity of one Howell Bunger valve. A failure of a tainter gate to reclose when the gate is raised 2.5 inches or less is bounded by a fully open Howell Bunger valve in the analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The UHS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 22 days following the design basis LOCA without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 97°F and the level should not fall below 218 ft MSL during normal unit operation.

APPLICABILITY

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

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

ACTIONS

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

If the UHS is inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate long term (22 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ 218 ft MSL.

SR 3.7.8.2

This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the service water temperature is ≤ 97°F.

REFERENCES

1. UFSAR, Section 9.2.4.
2. UFSAR Section 2.4.6.1.
3. UFSAR Section 2.1.1.2.
4. NUREG-75/024, "Final Environmental Statement Related to the Operation of H. B. Robinson Nuclear Steam-Electric Plant Unit 2," U. S. Nuclear Regulatory Commission, Washington DC 20555, April 1975, page 3-7.
5. USGS Historical Daily Values for Station Number 02130900, Black Creek Near McBee, South Carolina, Years 1960-1993.