Template NRR-058

AMENDMENT NO. 187 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. Robinson, UNIT 2

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Mr. D. E. Young, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant, Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

April 18, 2000

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2 - ISSUANCE OF AMENDMENT - TECHNICAL SPECIFICATION CHANGE ON ULTIMATE HEAT SINK (UHS) FROM 95°F to 97°F (TAC NO. MA5612)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 187 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated May 27, 1999, as supplemented by your letters of February 25, 2000, March 30, 2000, and eMail of March 13, 2000.

This amendment revises the maximum allowable service water temperature permitted by Surveillance Requirement 3.7.8.2 for the UHS from the currently permitted limit of 95°F to 97°F. Also, the amendment deletes the Conditions A and B of Limiting Condition for Operation 3.7.8 for required action and completion times, which expired after September 30, 1999; restores the original TS provisions of 6/36 hours to be in mode 3/5 respectively, in the event the UHS temperature were to exceed 97°F; and re-letters condition C to A.

It is noted that in your letter dated March 30, 2000, you have committed to implement the permanent modification to the Containment Spray (CS) Actuation Circuitry prior to unit restart following Refueling Outage Number 20. In the interim, CP&L will implement administrative procedures to require operators to block the CS actuation signal using installed plant equipment outside of, but in close proximity to, the control room.

A copy of the Safety Evaluation and the Technical Evaluation Report (Contract No. NRC-03-95-026) are enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/RA/
Ram Subbaratnam, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. to License No. DPR-23
2. Safety Evaluation
3. Technical Evaluation Report (Contract No. NRC-03-95-026)

cc w/encls: See next page *See previous concurrence

FILENAME - G:\PDII-2\Robinson\RMA5612.amd.wpd

PM:PDII-S2	LA:PDII-S2	SPLB:DSSA	SRXB:DSSA	EEIB:NRR	EMEB:NRR	OGC*NLO	SC:PD II-S2	D:PDII-S2
RSubbaratnam	EDunnington ETD	GHubbarad*	WJensen*	JKnox/HGarg*	DTerao*	AHogdon*	RCorreja	HBerkow
4/ 14/2000	4/14/2000	4 / 7/2000	4 / 10 /2000	4 / 7 /2000	4 /10 /2000	4 / 13/2000	4/19/2000	4/14/2000
Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 18, 2000

Mr. D. E. Young, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant, Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2 - ISSUANCE OF AMENDMENT - TECHNICAL SPECIFICATION CHANGE ON ULTIMATE HEAT SINK (UHS) (TAC NO. MA5612)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 187 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated May 27, 1999, as supplemented by your letters of February 25, 2000, March 30, 2000, and eMail of March 13, 2000.

This amendment revises the maximum allowable service water temperature permitted by Surveillance Requirement 3.7.8.2 for the UHS from the currently permitted limit of 95°F to 97°F. Also, the amendment deletes the Conditions A and B of Limiting Condition for Operation 3.7.8 for required action and completion times, which expired after September 30, 1999; restores the original TS provisions of 6/36 hours to be in mode 3/5 respectively, in the event the UHS temperature were to exceed 97°F; and re-letters condition C to A.

It is noted that in your letter dated March 30, 2000, you have committed to implement the permanent modification to the Containment Spray (CS) Actuation Circuitry prior to unit restart following Refueling Outage Number 20. In the interim, CP&L will implement administrative procedures to require operators to block the CS actuation signal using installed plant equipment outside of, but in close proximity to, the control room.

A copy of the Safety Evaluation and the Technical Evaluation Report (Contract No. NRC-03-95-026) are enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Ram Subbaratnam, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 187 to License No. DPR-23
2. Safety Evaluation
3. Technical Evaluation Report (Contract No. NRC-03-95-026)

cc w/encls: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Carolina Power & Light Company (CP&L, the licensee), dated May 27, 1999, as supplemented by letters dated February 25, 2000, March 30, 2000, and eMail dated March 13, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-23 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 187, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow, Director
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 187

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.3-26	3.3-26
3.3-27	3.3-27
3.6-20	3.6-20
3.6-21	3.6-21
3.7-9	3.7-9
3.7-21	3.7-21
3.7-21a	---
5.0-24	5.0-24
B 3.6-2	B 3.6-2
B 3.6-7	B 3.6-7
B 3.6-26	B 3.6-26
B 3.6-30	B 3.6-30
B 3.6-31	B 3.6-31
B 3.6-35	B 3.6-35
B 3.6-36	B 3.6-36
B 3.6-52	B 3.6-52
B 3.7-8	B 3.7-8
B 3.7-13	B 3.7-13
B 3.7-19	B 3.7-19
B 3.7-37	B 3.7-37
B 3.7-42	B 3.7-42
B 3.7-49	B 3.7-49
B 3.7-50	B 3.7-50
B 3.7-50a	---
B 3.7-51	B 3.7-51

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. AND 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
	AND		
	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
	AND		
	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)

BASES

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

A.1, and A.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 ≤ 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
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ULTIMATE HEAT SINK TECHNICAL SPECIFICATION

H. B. ROBINSON, UNIT 2

DOCKET NO. 50-261

1.0 INTRODUCTION

In a letter dated May 27, 1999, Carolina Power and Light Company (CP&L) requested a change to Technical Specification (TS) 3.7.8, "Ultimate Heat Sink (UHS)," to change the temperature limit from 95°F to 97°F for H. B. Robinson, Unit 2 (Robinson). In support of this request, and consistent with the analyses that were completed in support of this change, TS changes were also requested to decrease the required actuation setpoint and ALLOWABLE VALUE for the Containment Pressure High High signal, decrease the required closure time for the Main Feedwater Isolation Valves, decrease the required closure time for the Main Steam Isolation Valves (but with no accompanying change to the TS surveillance requirement), increase the required Isolation Valve Seal Water (IVSW) tank pressure and IVSW nitrogen bottle pressure, and increase the containment internal pressure P_a. The requested increase in UHS temperature and supporting analyses would also require CP&L to change its commitment to Inspection and Enforcement Bulletin (IEB) 80-06, "Engineered Safety Features (ESF) Reset Controls," to modify the Containment Spray (CS) system actuation circuitry so that it can be blocked to allow switching the CS pump suction from the Refueling Water Storage Tank following a large break Loss-of-Coolant Accident (LOCA). The revised containment analyses indicate that containment pressure will be higher than the CS actuation setpoint at the time switchover is completed, and the CS pumps may need to be manually restarted following the switchover. The actuation circuitry currently does not allow the CS pumps to be restarted while the original actuation signal is present. Finally, the analyses of record for the containment response to a main steam line break inside containment and to a large break LOCA are changed in support of the requested increase in UHS temperature.

In a supplement to the amendment application dated February 25, 2000, the licensee responded to a number of questions that were raised by the staff, provided a revised TS markup for Page B 3.6-26 to correctly depict the initial pressure condition that was used in the containment analysis, and requested an additional change to TS 3.7.8. A note that is included in TS 3.7.8 indicates that Conditions A and B no longer apply after September 30, 1999, and another note indicates that Condition C only applies after September 30, 1999. The supplement revises TS 3.7.8 to remove the two notes and Conditions A and B from TS 3.7.8, and to re-letter Condition C to be Condition A, thus restoring provisions for required action and completion times of 6/36 hours to be in mode 3/5, respectively, in the event the UHS temperature were to exceed 97°F. The supplement also discusses interim measures that will be taken upon issuance of the requested license amendment until such time that the required modifications for blocking the CS actuation circuitry can be completed.

2.0 BACKGROUND INFORMATION

The UHS for H. B. Robinson, Unit 2, is Lake Robinson, which was originally developed for use by H. B. Robinson, Unit 1 (a fossil plant). The lake is the UHS for both of the Robinson units during normal power operation, and also serves as the UHS for H. B. Robinson, Unit 2, during postulated accident conditions. The UHS is described in Section 9.2.4 of the Robinson Updated Final Safety Analysis Report (UFSAR).

During the summer months, the temperature of Lake Robinson increases due to solar heating, and there have been occasions when the lake temperature has approached the maximum value allowed by the TS of 95°F. During the summer of 1999, the lake temperature actually exceeded the TS limit for a brief period of time. The NRC approved a temporary change to the Robinson TS which allowed the 95°F temperature limit to be exceeded for periods of up to 8 hours during the summer of 1998 (NOED No.98-6-010 and follow up exigent Amendment No. 179) while CP&L completed supporting analyses to justify a permanent increase in the UHS temperature limit (allowed by Notes 1 and 2 and Conditions A and B of TS 3.7.8). The analyses have been completed, and the May 27, 1999, submittal is the licensee's request to increase the UHS temperature limit on a permanent basis. Because the staff's review of this request could not be completed until after the summer of 1999, a temporary TS change (similar to the one that was approved for the 1998 summer months) (NOED No.99-6-06 and follow up exigent Amendment No. 184) was approved by the NRC as an interim measure.

3.0 EVALUATION

The staff's evaluation consisted of an audit of the analyses that were completed by CP&L to justify the increase in UHS temperature, including the effect of increased UHS temperature on plant equipment; a review of requested changes to other related TS requirements that are affected; and a review of the changes to the plant design-basis that are reflected in the submittal but are not part of a TS change. The staff's review of these items is summarized in the sections that follow. The staff's review did not include an evaluation of specific analytical assumptions except as discussed below. Assumptions that were not included in the staff's review may be subject to further review during future NRC audit and inspection activities.

3.1 Audit of Supporting Analyses

The analyses that were performed by CP&L are rather complicated and numerous and, in the interest of expediting this effort, Scientech, an NRC contractor, performed an audit (Contract No. NRC-03-95-026, Task 244). During a site visit that was conducted from September 20 through September 22, 1999, Scientech completed a review of the licensee's calculations. The results of this review are documented in a Technical Evaluation Report (TER) that was issued by Scientech in October 1999 (attached). The following findings are documented in the TER:

a. Safety-related components cooled by the service water and component cooling water systems can still perform their safety-related functions during normal operation and under accident conditions if the service water temperature does not exceed 97°F.

b. Sufficient water will be available in Lake Robinson to supply the service water pumps for at least 22 days if the UHS temperature does not exceed 97°F.

c. Increasing the UHS temperature to 97°F will not cause the containment pressure and temperature design limits to be exceeded.

d. Approved methodologies were used for the containment and LOCA analyses.* These methodologies were applied in an appropriate and conservative manner.

Based on the discussion and findings in the Scientech TER, the NRC staff considers the licensee's analytical approach to be acceptable. Also, while the emergency diesel generators become limiting as the UHS temperature approaches 99°F, the licensee's request to increase the UHS temperature to 97°F does not pose a problem for continued operation of plant equipment at that lower temperature.

3.2 Related Changes to the TS Requirements

The licensee has requested changes to several other TS requirements to reflect the assumptions and results of the supporting analyses that have been completed. In addition to the change that is requested to TS 3.7.8, "Ultimate Heat Sink (UHS)," to increase the temperature limit from 95°F to 97°F, the following related TS changes are also requested in the licensee's submittal.

3.2.1 TS 3.3.2, "Engineered Safety Features Actuation System Instrumentation"

The licensee has requested a decrease in the ALLOWABLE VALUE and the NOMINAL TRIP SETPOINT for FUNCTIONS 2.c, 3.b(3), and 4.c, "Containment Pressure High High" from 20.45 psig and 20 psig (respectively), to 10.45 and 10 psig. The revised containment analyses credit actuation of the Containment High High Pressure signal at an analytical limit of 12 psig, which corresponds to a NOMINAL TRIP SETPOINT of 10 psig. The Containment Pressure High High Nominal Trip Setpoint and Allowable Value take into consideration the calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors associated with these instruments. These reduced values remain significantly greater than the pressures that normally exist in containment and, therefore, spurious actuations should not pose a problem.

The NRC staff has reviewed and approved the instrument setpoint methodology used at Robinson during the conversion to Improved Technical Specifications (ITS). The licensee's setpoint methodology is referenced in the TS Bases on pages B 3.3-57 and 58, and the methodology accounts for all of the uncertainties. The NRC Safety Evaluation dated October 24, 1997, for the ITS conversion, documents the staff's evaluation of the licensee's instrument setpoint methodology.

The licensee identified one feature of this change which affects post-accident operation of the containment spray pumps. This aspect of the requested change is considered separately in Section 3.3.2 of this evaluation. Aside from this, the staff considers the change to TS 3.3.2 to be acceptable in all other respects.

* While this is true in most respects, it is noted in Section 3.3.1 of this evaluation that the COCO code has not been approved by the NRC.

3.2.2 TS 3.6.8, "Isolation Valve Seal Water (IVSW) System"

The requested change to TS 3.6.8 increases the minimum allowed IVSW tank pressure and IVSW dedicated nitrogen bottle pressure in Surveillance Requirement (SR) 3.6.8.1 and SR 3.6.8.5 from 44 psig to 44.6 psig. The revised containment analysis calculated a new peak pressure (P_a) of 40.5 psig for the large break LOCA inside containment, and the requested increase in the minimum allowed IVSW tank pressure and IVSW dedicated nitrogen bottle pressure is necessary to satisfy the TS criteria of 1.1 times P_a. During a meeting that was held on February 4, 2000, the licensee indicated that uncertainties would be accounted for by the surveillance procedure. Based on these considerations, the staff finds the requested change to TS 3.6.8 to be acceptable.

3.2.3 TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulation Valves (MFRVs), and Bypass Valves"

The requested change to TS 3.7.3 decreases the maximum allowed MFIV closure time in SR 3.7.3.2 from 80 seconds to 50 seconds. The MFIV valve actuators have been modified to support the faster closing time, and the revised value is consistent with the MFIV closure time that was credited for main line break (MSLB) inside containment. During a meeting that was held on February 4, 2000, the licensee indicated that the design valve stroke time was 37.02 seconds, and the 50-second closure time accounts for valve stall time and provides additional margin above the design closure time. With changes to the gearing and the motor, the MFIV stroke time will be reduced from 68 seconds to 39.2 seconds. These valves are part of the licensee's Generic Letter 89-10 program for motor-operated valves. The licensee indicated that uncertainties would be accounted for in the surveillance procedure. Based on these considerations, the staff finds the requested change to TS 3.7.3 to be acceptable.

3.2.4 TS Section 5.5.16, "Containment Leakage Rate Testing Program"

The requested change to TS Section 5.5.16 increases the peak containment internal containment pressure for the design-basis LOCA (P_a) from 40 psig to 40.5 psig. This new value, which was calculated for the large break LOCA inside containment, remains less than the containment design pressure of 42 psig. Therefore, the requested change to TS Section 5.5.16 is acceptable.

3.2.5 TS 3.7.2, "Main Steam Isolation Valves"

The new MSLB containment analysis credited a 2-second closure time for the Main Steam Isolation Valves (MSIVs), which is shorter than the 5-second closure time that is required by SR 3.7.2.1. The licensee has not proposed a change to SR 3.7.2.1 to reflect the faster MSIV closing time, and intends to rely on the existing 5-second closure time to verify MSIV operability, and to provide assurance that the MSIVs will close within the required 2-second interval for accident conditions.

The MSIVs are 26-inch swing-disc-type isolation valves that are closed by the combined effects of an air operator, steam flow, and a spring (for 9 inches of the actuator's 19-$^{7}/_{16}$ inch stroke). When the MSIV receives a closing signal, the air operator and spring drive the valve disc down

into the steam flow. The swing disc is then accelerated to the fully closed position by the force of the steam acting to seat the valve disc in the direction of the steam flow. The licensee completed an engineering evaluation that examined the time it takes for the valve disc to enter the steam flow, the forces being exerted on the valve disc during closure, and plant data, and concluded that the MSIVs will close within the required 2-second time period. Since it is undesirable to perform a closure test of the MSIVs when the plant is operating under full steam flow conditions, the licensee intends to continue testing the MSIVs during Mode 3 conditions in accordance with SR 3.7.2.1. The licensee has concluded that satisfying the existing 5-second closure time specified in SR 3.7.2.1 combined with trending of the test data will provide adequate assurance that the MSIVs have not degraded and will close within the 2 seconds assumed in the MSLB containment analysis.

3.2.6 Administrative Changes to TS 3.7.8, "Ultimate Heat Sink (UHS)"

In a supplement to the amendment application dated February 25, 2000, the licensee requested an additional change to TS 3.7.8. A note that is included in TS 3.7.8 indicates that Conditions A and B no longer apply after September 30, 1999, and another note indicates that Condition C only applies after September 30, 1999. The two notes and Conditions A and B were added as temporary measures to permit the maximum allowed UHS temperature limit to be exceeded for short durations during the 1998 and 1999 summer months while a permanent TS change was being prepared. The supplement revises TS 3.7.8 to remove the two notes and Conditions A and B from TS 3.7.8, and to re-letter Condition C to be Condition A. These changes are administrative in nature and are, therefore, acceptable.

3.3 Changes to the Plant Design Basis

In addition to the changes discussed in 3.1 and 3.2 above, the supporting analyses for the UHS temperature increase also result in other changes in the Robinson design basis. NRC's evaluation of these changes is discussed below.

3.3.1 Containment Temperature and Pressure Analysis

The licensee reevaluated the containment response to postulated LOCA events at two locations: (1) the double-ended guillotine rupture of a hot leg, and (2) the double-ended guillotine rupture of a cold leg at the suction of a reactor coolant pump. The rupture of a hot leg would cause the highest energy release rate to the containment from the reactor system and the rupture of a cold leg at the suction of a coolant pump would enhance heat removal from the steam generators. The pump suction break location was found to result in the highest calculated containment pressure. In addition to LOCA analyses, the licensee analyzed the containment response resulting from the double-ended rupture of a main steam line inside the containment building. The effects of various single failure assumptions were evaluated for both LOCA and MSLB events.

The calculation of mass and energy release for the postulated LOCAs was calculated by the licensee using the methodology in WCAP-10325-P-A (Ref. 1) which has been approved by the NRC staff for that purpose. The calculation of mass and energy release following a double-ended rupture of a main steam line was calculated using the LOFTRAN code (Ref. 2) which has

also been approved by the NRC staff. In the case of the MSLB, the release to the containment was made conservative by the assumption that only steam would exit the break. For a large MSLB, a considerable amount of water would be expected to be discharged to the containment. The discharge of water would act to reduce the total energy release and would also act to remove any superheating in the containment atmosphere.

The containment response was calculated by the licensee using the COCO code, which has not been approved by the NRC staff. The COCO code assumes that, whenever hot water is discharged from the break, the hot water falls directly to the floor of the containment without mixing with the containment atmosphere. This assumption is sometimes referred to as "pressure flash." This pressure flash assumption results in peak containment pressures for LOCA events that are about 2.0 psi lower than those for the "temperature flash" assumption, which is where the hot water mixes with the containment atmosphere.

The COCO code also assumes that, whenever the containment atmosphere is superheated, as it might be following a main steam line break, heat is removed from the containment atmosphere without producing any condensation. This assumption quickly removes any superheat from the containment atmosphere and is non-conservative for determining containment temperature for equipment qualification purposes. The NRC staff recommends that 92% of the heat removed from the containment atmosphere be assumed to result in condensation and 8% of the heat removal be assumed to occur without condensation (Ref. 3). The condensation fraction recommended by the staff helps account for inaccuracies in calculating local containment heat transfer with a one-node lumped model such as used by COCO. The recommended condensation fraction was determined by correlating full-scale containment test data (Ref. 4) using the CONTEMPT computer code (Ref. 5) so that both the predicted pressures and temperatures matched the test data.

To ensure that containment temperature and pressure would remain below design limits, the staff performed confirmatory analyses using the CONTEMPT code for both the postulated LOCA at a coolant pump suction and for the MSLB. For the pump suction LOCA, the staff calculated a containment pressure that was somewhat higher than that calculated by the licensee but still less than the design pressure.

For the MSLB, the staff also calculated a somewhat higher pressure than the licensee but below the design limit. Higher containment atmosphere temperatures were also calculated. Superheating of the containment atmosphere was predicted as steam exited the break and compressed the original air mass. The containment atmosphere was calculated to reach a maximum temperature of 340°F at about 32 seconds. At that time, the containment sprays actuated on high containment pressure and rapidly reduced containment temperature. A sample electrical structure for equipment qualification was included in the analysis. Because of the short duration of the superheated containment temperature, the sample structure remained within the equipment qualification limit. Additional discussions of equipment qualification are contained in Section 3.4 of this evaluation.

Following an MSLB, as most of the water is exhausted from the broken steam generator, the steam generator tubes would become partially uncovered. Under this condition, the steam passing within the channels between the tubes might become superheated and increase the

containment temperature. The staff has previously reviewed this issue and has concluded that any increase in the containment temperature would not be significant (Ref. 6). In the case of H. B. Robinson, the containment sprays would be actuated early in the event and would thereafter remove any superheat from the containment atmosphere before the steam generator tubes became uncovered.

The NRC staff has evaluated the effect of increasing the allowed UHS temperature on the containment during design basis events, as discussed above, and considers the results to be acceptable. While the staff calculated maximum containment temperatures and pressures in excess of those calculated by the licensee, the staff's results remained within the containment design limits.

3.3.2 CS Actuation Signal and Block

CS initiates on a Containment High High Pressure signal, which currently has a NOMINAL TRIP SETPOINT of 20 psig. However, in support of the requested increase in UHS temperature, the licensee has requested a decrease in this value from 20 psig to 10 psig (see Section 3.2.1, above). Containment pressure reaches 10 psig within the first few seconds after the large break LOCA occurs, which results in CS actuation. The suction of the CS pump must be shifted from the refueling water storage tank (RWST) when RWST level has decreased to 9% and aligned for cold leg recirculation, which occurs about an hour after CS actuation. In order to accomplish this shift, the CS pumps must be stopped to facilitate valve realignment. The CS pump circuit breakers have an anti-pump feature that prevents the CS pump motors from being restarted after being stopped while the original actuation signal is still present. Because containment pressure is still well above 10 psig at the time of the switchover to cold leg recirculation, the CS pumps cannot be restarted following the switchover unless the actuation circuitry is blocked.

The licensee's response to IE Bulletin 80-06 included a commitment to ensure that an automatic safety injection signal and manual signal to CS would not be inhibited when another signal is present, blocked, reset, or overridden. This commitment was implemented through a modification that prevented resetting CS whenever an engineered safety feature (ESF) actuation signal is present. However, the new analysis requires the CS pump to be stopped and restarted even though the ESF signal is still present. Therefore, the licensee is proposing a change to its previous commitment in that blocking the CS actuation signal would be allowed, and emergency operating procedures would be used to control the subsequent operation of the CS pumps. This change in the licensee's commitment meets the guidance of IE Bulletin 80-06 because the actuated equipment will remain in the emergency mode when the actuation circuit is blocked and the operator will be able to change the state of the actuated equipment after the actuation signal is blocked. In accordance with the licensee's analysis, the operator has about 20 to 30 minutes to shift the suction and restart the CS pump. The licensee has also committed to periodically test the blocking signal to ensure that the signal has been properly restored when it is unblocked.

In supplements dated February 25, 2000, and March 30, 2000, the licensee indicated that a permanent modification would be completed to allow the CS actuation signal to be blocked prior to the next refueling outage currently scheduled for spring 2001. In the interim, upon NRC approval and issuance of this UHS license amendment request, CP&L will implement

administrative procedures to require operators to block the CS actuation signal using installed plant equipment outside of, but in close proximity to, the control room. While the staff considers the interim measures to be acceptable, reliance on operator actions that must be completed outside of the control room should be minimized. Therefore, the staff's approval is contingent upon the licensee fully implementing the permanent modification prior to unit restart after the next refueling outage RFO-20. The licensee has, in a letter dated March 30, 2000, committed to this implementation schedule, which will be duly tracked under the provisions of the Agency Commitment Tracking Procedure (Ref. 7).

Based on the information that was submitted and the considerations discussed above, the staff finds that the licensee continues to meet the intent of IE Bulletin 80-06 and that the proposed interim measures, modification, and operator actions are acceptable.

3.3.3 Feedwater Leakage Assumptions for MSLB

Modeling feedwater leakage in the MSLB analysis adds to the mass and energy that is released to the containment through the faulted steam generator. In the past, the licensee has observed some feedwater leakage through the MFRVs and the MFIVs. The highest leak rate that has been measured is 82 gpm for the MFRV and 22 gpm for the MFIV. Since these valves are in series, the smaller leak rate would apply with both valves closed. In the MSLB analysis, CP&L assumed a feedwater leak rate of 75 gpm for the MFRV/MFIV combination, and 125 gpm for a failed open MFIV. This bounds the 22 gpm feedwater leak rate that would be expected for the MFRV/MFIV combination, and bounds the 82 gpm feedwater leak rate that would be expected for a failed open MFIV. Based on discussions with the licensee during a telephone conversation on March 6, 2000, it is the staff's understanding that the feedwater leak rate through the MFIVs and MFRVs is measured each refueling outage in conjunction with valve in-service testing to assure that these assumptions remain valid. The licensee indicated that measurement uncertainties are accounted for in the surveillance procedures. Based on the information provided by the licensee, the staff considers the feedwater leak rate assumptions to be acceptable.

3.4 Environmental Qualification Considerations

One of the basic performance requirements for the UHS is that the design basis temperatures of safety-related equipment not be exceeded. This performance requirement is verified through periodic surveillance that ensures that service water (SW) inlet temperature is less than or equal to 95°F. By amendment dated May 27, 1999, the licensee requested that this 95°F surveillance limit be increased to 97°F with no time limit allowed for SW inlet temperature to exceed the 97°F surveillance limit.

With regard to environmental qualification (EQ), the licensee evaluated equipment required to operate during accident conditions using containment composite (MSLB/LOCA) pressure and temperature profiles developed for the new 97°F SW inlet temperature limit. The licensee found (as indicated on pages 17 and 24 of attachment II to their May 27, 1999, amendment request) that the new accident profiles are generally bounded by the profile used for EQ testing of equipment; however, EQ limits for both temperature and pressure are exceeded for a time period of about 24 hours following the initiation of a steam line break or LOCA after the occurrence of peak temperature and pressure conditions.

In justification for exceeding EQ temperature limits, the licensee compared equipment degradation caused by the EQ test profile to degradation that would be caused by the new accident profile using the Arrhenius calculation. The licensee found greater (or equivalent) calculated degradation from the EQ test profile than from the new accident profile and thus concluded equipment continues to be qualified to the higher post-accident temperature conditions. Based on there being greater (or equivalent) degradation, the NRC staff concluded that there is reasonable assurance that components will continue to have the required capability to perform their safety function when needed and agree that equipment continues to be acceptably qualified for the higher temperature conditions.

With regard to pressure exceeding EQ limits, the licensee, by eMail dated March 13, 2000, identified five components where EQ testing does not entirely envelop the new accident profile. For these five components, the licensee provided the following bases describing why each component continues to be considered qualified for the higher pressure conditions:

EQDP 3.0, Rockbestos Firewall III Cable

During the period from 1.15 to 11.5 days, the predicted accident pressure exceeds the test pressure by a maximum of 2 psi. This difference is accommodated by the difference between the accident and test pressures at the time containment spray is actuated early in the accident. At this time in the accident sequence, the test pressure exceeds the predicted accident pressure by at least 11 psi. Thus, the qualification test subjected the equipment to a more severe pressure at a time when the significant environmental stressors were present. On this basis, it is concluded that the test results continue to demonstrate that the subject equipment remains environmentally qualified.

EQDP 3.1, Rockbestos Coaxial Cable

During the period from 1.15 to 11.5 days, the predicted accident pressure exceeds the test pressure by a maximum of 1 psi. This difference is accommodated by the difference between the accident and test pressures at the time containment spray is actuated early in the accident. At this time in the accident sequence, the test pressure exceeds the predicted accident pressure by at least 10 psi. Thus, the qualification test subjected the equipment to a more severe pressure at a time when the significant environmental stressors were present. On this basis, it is concluded that the test results continue to demonstrate that the subject equipment remains environmentally qualified.

EQDP 4.1, Rosemount 1154 Pressure Transmitters

During the period from 1.15 to 11.5 days, the predicted accident pressure exceeds the test pressure by a maximum of 2 psi. This difference is accommodated by the difference between the accident and test pressures at the time containment spray is actuated early in the accident. At this time in the accident sequence, the test pressure exceeds the predicted accident pressure by at least 22 psi. Thus, the qualification test subjected the equipment to a more severe pressure at a time when the significant environmental stressors were present. On this basis, it is concluded that the test results continue to demonstrate that the subject equipment remains environmentally qualified.

EQDP 18.1, Westinghouse Reference Junction Boxes

The Reference Junction Boxes were subjected to a maximum pressure of 78 psig for approximately 6 hours. Additionally, the qualification test pressure exceeds the new requirements for greater than 300 hours. The maximum degradation caused by pressure would have occurred at the higher pressures. Therefore, the 5 psig pressure for 29 days (696 hours) is not sufficient to cause extensive degradation to this box. Based on these considerations, the Reference Junction Boxes are qualified for the accident and post-accident phases, which include the new pressure profile.

EQDP 36.0, Boston Insulated Wire

The qualification test profile exceeds the new H. B. Robinson accident pressure requirements for the first 4 days of the test. After 4 days of testing, the test pressure was reduced to 0 psig while the new pressure profile remained at 5 psig for the remaining 26 days of the accident. The two test pressure transients with peaks of 110 psig greatly exceed the plant peak pressure requirement of 42 psig. The second pressure transient produces more stress than a sustained pressure. On this basis, the test pressure profile can be considered more severe than the new accident pressure profile and this wire is considered environmentally qualified for the new accident pressure profile.

Based on the licensee's assessment, staff concludes that there is reasonable assurance that each of the above-identified components will continue to have the required capability to perform their safety function when needed and agree that equipment continues to be acceptably qualified for the higher pressure conditions.

In addition to the above-identified cases, review of the licensee's MSLB analysis and further discussion during a February 4, 2000 meeting identified another case where temperature may exceed EQ limits. It was determined that super-heated containment air temperature following a steam line break (prior to initiation of containment spray) may exceed EQ temperature limits. The results of the licensee's MSLB analysis, provided as part of supplemental information transmitted by letter dated February 25, 2000, shows an early peak containment temperature resulting from super heat and shows this temperature to be less than the EQ limits. We disagreed with these results. Our calculations found this early peak containment temperature approaching 350°F and exceeding the EQ limits.

In justification for this additional case where early peak containment temperature may exceed EQ limits, the licensee, as part of telephone discussions, indicated that temperature conditions used as part of an equipment heat transfer analysis (described in section 6.2.1.4.3 of Amendment 11 to the H. B. Robinson UFSAR) bound the temperature conditions that may be generated following a steam line break accident due to the higher SW inlet temperature. Equipment temperatures, thus, do not exceed EQ limits and equipment is considered to be acceptably qualified.

The staff agrees that the temperature conditions assumed in the original plant heat transfer analysis bound temperature conditions that may be generated following a steam line break

accident due to the higher SW inlet temperature. Thus, based on the results of the licensee's original heat transfer analysis, the staff agrees that equipment continues to be acceptably qualified.

Based on the licensee's assessment that design basis temperature/pressure limits of safety-related equipment have not been exceeded, and on the analysis results for the identified cases when design basis limits have been exceeded, we conclude that equipment continues to be acceptably qualified for the 97°F allowable SW inlet temperature.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 9001). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

As a result of its evaluation as described above, the staff concludes that the proposed change to SR 3.7.8.2 provides a significant amount of additional flexibility without any significant reduction in plant safety because of the supporting analysis regarding the cooled equipment. The proposed change is, therefore, acceptable.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Tatum

Date: April 18, 2000

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2. Letter from C. O. Thomas (USNRC) to E. P. Rahe Jr. (Westinghouse), "Acceptance for Referencing of Licensing Topical Reports: WCAP-7907(P)/(NP) 'LOFTRAN Code Description'," July 29, 1983.

3. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," NUREG-0588 Rev. 1, July 1981.

4. "Simulated Design Basis Accident Tests of the Carolinas Virginia Tube Reactor Containment Final Report," IN-1403, December 1970.

5. "CONTEMPT-LT/028-A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," NUREG/CR-0255, March 1979.

6. Letter from Charles E. Rossi, (USNRC) to E. P. Rahe Jr. (Westinghouse), "Acceptance for Referencing of Licensing Topical Reports WCAP-8822-P-S1/WCAP-8860-S1 and WCAP-8822-P-S2/WCAP-8822-S2, "Mass and Energy Release Following a Steam Line Rupture," May 27,1886.

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SCIE-NRC-388-99

TECHNICAL EVALUATION REPORT

HB Robinson Steam Electric Plant Unit 2
Request for Technical Specifications Change

Ultimate Heat Sink (UHS)

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EXECUTIVE SUMMARY

This report presents a technical evaluation of the Carolina Power and Light Co. request for change to Technical Specification (TS) 3.7.8, "Ultimate Heat Sink (UHS)", for the H. B. Robinson Steam Electric Plant, Unit 2, dated May 27, 1999. This change would increase the maximum allowable service water (SW) temperature from 95 ^{0}F to 97 ^{0}F, subject to an allowed outage time (AOT) requirement such that plant shutdown would commence in the event that the SW temperature exceeded 97 ^{0}F for 8 hours or a maximum of 99 ^{0}F. The CP&L request also included several additional TS changes necessitated by the increase in maximum UHS temperature.

The UHS is the source of the service water provided to dissipate reactor residual heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. At H. B. Robinson, Unit 2, the UHS is Lake Robinson, a man-made water impoundment.

This evaluation concludes that while the proposed TS change decreases the heat capacity of the UHS and the cooling capacity of safety-related equipment attached to the service water and closed cooling water systems, analysis has shown that containment peak pressure (GDC 44) requirements will still be met. However, if the lake temperature is permitted to closely approach the 99 ^{0}F limit during the course of a particular day, meteorological effects may cause the SW temperature to exceed the 99 ^{0}F temperature limit on a subsequent day. Because of the long response time and the relative insensitivity of the lake temperature to a change in reactor power level, there are no definitive actions that the operator could take to control the lake temperatures within the prescribed limit. Specific safety-related components, most notably the emergency diesel generators (EDGs), are not qualified for operation above 99 ^{0}F. Loss of the EDGs increases the risk of core melt whether or not the plant is promptly shut down.

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Glossary

AOT	Allowed Outage Time
CCS	Component Cooling System
CS	Containment Spray
CP&L	Carolina Power and Light Co.
DBA	Design Basis Accident
EDG	Emergency Diesel Generators
ESF	Engineered Safety Features
EQ	Equipment Qualification
GDC	General Design Criteria
HBR2	HB Robinson Steam Electric Plant, Unit 2
IEB	Inspection and Enforcement Bulletin
IPE	Individual Plant Examination
IVSW	Isolation Valve Seal Water
LB	Large Break
LCO	Limiting Conditions for Operation
LOCA	Loss-of-Coolant Accident
MFIV	Main Feedwater Isolation Valves
MSIV	Main Steam Isolation Valves
MSLB	Main Steam Line Break
NOED	Notice of Enforcement Discretion
P_a	Test pressure for containment leak rate testing
PCT	Peak clad temperature

SRP	Standard Review Plan
SW	Service Water
SWS	Service Water System
TS	Technical Specifications
UHS	Ultimate Heat Sink
UFSAR	Updated Final Safety Analysis Report
WEC	Westinghouse Electric Company

SCIE-NRC-388-99

TECHNICAL EVALUATION REPORT

HB Robinson Steam Electric Plant Unit 2
Request for Technical Specifications Change

Ultimate Heat Sink (UHS)

1.0 INTRODUCTION

By letter dated May 27, 1999, Carolina Power & Light Company (CP&L) requested a change to Technical Specification (TS) 3.7.8, "Ultimate Heat Sink (UHS)", for the H. B. Robinson Steam Electric Plant, Unit 2. This change would increase the maximum allowable service water (SW) temperature from 95 ^{0}F to 97 ^{0}F subject to an allowed outage time (AOT) requirement such that plant shutdown would commence in the event that the SW temperature exceeded 97 ^{0}F for 8 hours or a maximum of 99 ^{0}F. The CP&L request also included several additional TS changes necessitated by the increase in maximum UHS temperature.

The UHS is the source of the service water provided to dissipate reactor residual heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. At H. B. Robinson, Unit 2, the UHS is Lake Robinson, a man-made water impoundment. Lake Robinson is heated by the sun (insolation), and operation of H. B. Robinson, Units 1 (fossil) and 2 (nuclear). The lake is cooled by natural phenomena, such as evaporative cooling, natural convection, and diurnal radiant heat losses. During the summer, the average solar heat input is comparable to the total heat input from both Robinson Units 1 and 2.

The background and technical evaluation of these changes is presented in the following sections.

2.0 BACKGROUND AND REVIEW PROCESS

Because of an increase in the frequency and severity of hot, dry weather conditions experienced at Lake Robinson, located near Hartsville, South Carolina, Carolina Power and Light submitted several successive requests to change TS 3.7.8, "Ultimate Heat Sink (UHS)" for the HB Robinson Steam Electric Plant, Unit 2 (HBR2). Lake Robinson is the UHS for HBR2 and is the only site for both intake and discharge of service water from the plant.

The first CP&L TS change request, dated June 26, 1998, modified TS 3.7.8 to permit the SW temperatures to exceed 95 ^{0}F for up to 8 hours. Just after that TS change had been submitted (June 27, 1998), CP&L requested a Notice of

Enforcement Discretion (NOED) until that TS change request could be approved. However, the NOED also modified the original TS change request to add the requirement that plant shutdown would commence should the service water temperature exceed 99 ^{0}F instantaneously. The NRC accepted CP&L's request for NOED on July 1, 1998.

On July 22, 1998, at the NRC's request, CP&L revised the TS change to apply only to the temporary period until September 30, 1998. The NRC approved the revised TS change as part of License Amendment No.179, dated July 29, 1998.

In March 1999, CP&L essentially resubmitted the TS 3.7.8 change, asking that the change be made permanent. As before, plant shutdown would be commenced should the service water temperature exceed 95 ^{0}F for 8 hours or 99 ^{0}F instantaneously. The NRC approved these temperature limits for a one-time TS change in April 1999.

In May, CP&L submitted the current TS change request [Ref. 1], which increased the requested maximum SW temperature to 97 ^{0}F. The submittal also included additional component and containment analyses to support their request for a permanent change in the UHS/SW temperature limits. As before, CP&L requested an 8 hour allowed outage time (AOT) with an upper temperature limit of 99 ^{0}F.

On September 20, 1999, a meeting was held at the HBR2 site. Attendees at this meeting included staff members from the Regulatory Affairs Group at Carolina Power and Light Co., Mr. Jim Tatum, USNRC, Office of Nuclear Reactor Regulation, and Mr. Ben Gitnick, SCIENTECH Inc., a consultant to the USNRC for the review of the HBR2 UHS TS change request. During that meeting, Mr. Tatum stated that the purpose of the NRC visit was to review the documentation supporting the TS change request, and if necessary, discuss items needing clarification directly with members of the CP&L staff.

CP&L stated their goal was to provide the NRC review team free access to staff and documentation resources, as much as possible under the circumstances, given that the plant was entering a scheduled outage and some staff may be unavailable as a result. During the remainder of that day and for the next two days the supporting documentation was reviewed. Discussions were held with several CP&L engineers, including,

Harold Chernoff, Director of Regulatory Affairs, HBR2
Albert Garrou
Alan Helms
David Martrano
Roger Stewart
Peter Yandow

The three-day review meeting was very productive. The CP&L analyses were well documented, and much of the documentation had been set aside for the review prior to the arrival of the review team. The CP&L staff quickly responded to all NRC requests for additional documentation or clarification.

3.0 APPLICABLE REQUIREMENTS AND ACCEPTANCE CRITERIA

The design criteria for Ultimate Heat Sinks is specified in General Design Criteria (GDC) 44, delineated in 10 CFR 50, Appendix A. Additional guidelines for the design bases are contained in Regulatory Guide 1.27, Rev. 2 [Ref. 3]. Review considerations and acceptance criteria for UHS designs are given in Standard Review Plan (SRP) 9.2.5 [Ref. 4].

For a passive UHS system such as Lake Robinson, the GDC 44 requirements can be summarized as follows: Does the lake have sufficient water, at a cool enough temperature, to safely remove the heat from the reactor and required safety systems for both steady-state and post-accident operation?

Regulatory Guide 1.27 requires that a UHS should contain a minimum 30 day supply of cooling water. However, HB Robinson was licensed on July 31, 1970, prior to the promulgation of this regulatory guide, which Lake Robinson was not designed to meet. The Bases Section for HB Robinson TS 3.7.8 states that Lake Robinson is capable of providing cooling water to HB Robinson, Unit 2, for at least 22 days following a design basis accident. As a 22 day period has been the basis requirement for HB Robinson, Unit 2 for a considerable period of time, it will be used to determine if Lake Robinson has a sufficient water quantity to support the proposed UHS TS change.

The proposed TS change increases the maximum allowable SW temperature, an input parameter in the LOCA and MSLB containment analyses. The licensing basis for containments is delineated in GDCs 16, 38 and 50, Sections 6.2.1.1 and 6.2.1.2 of the Standard Review Plan (SRP) [Ref. 4] as well as Regulatory Guide 1.70 [Ref. 5]. In the instance of this UHS TS change request, all of these containment analysis acceptance criteria can be summarized into the requirement that the peak calculated pressure remain less than the containment design pressure.

Other requirements for this TS change request are in the area of LB LOCA Peak Clad Temperature, (PCT), and Environmental Qualification (EQ). The LOCA PCTs are required to remain below 2200 ^{0}F to satisfy 10 CFR 50, Appendix K. Post-LOCA or MSLB pressure and temperature profiles should not exceed the profiles that have previously been environmentally qualified for the safety-related equipment required to operate during and following an accident at HB Robinson.

4.0 TECHNICAL EVALUATION OF SYSTEM PERFORMANCE

4.1 ULTIMATE HEAT SINK

The UHS is the source of the service water provided to dissipate reactor residual heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. At H. B. Robinson, Unit 2, the UHS is Lake Robinson, a man-made water impoundment. Lake Robinson is heated by the sun (insolation), and operation of H. B. Robinson, Units 1 (fossil) and 2 (nuclear). The lake is cooled by natural phenomena, such as evaporative cooling, natural convection, and diurnal radiant heat losses. CP&L has stated (in previous TS submittals) that the average summer heat input due to insolation, is comparable to the total heat input from both Robinson Units 1 and 2.

Carolina Power and Light performed an analysis to demonstrate that there is sufficient water in Lake Robinson to feed the service water pumps for 22 days. The natural evaporation rate used was reasonable and the forced evaporation (due to the excess heat introduced into the lake) rate was conservatively estimated.

The service water intake from Lake Robinson is near the lower end of the lake. Service water discharged from the plant is returned to Lake Robinson via a 4.2-mile long discharge canal that empties into the lake near its upper end. During full power operation, the normal transient time of water through the discharge canal is approximately 3.5 hours. An even longer period of time would be required for the water exiting from the discharge canal to mix in Lake Robinson and return to the service water intake. Hence, the effect of a plant shutdown in the event that the SW temperature limit is exceeded will not immediately effect the temperature of the service water entering the plant. The Carolina Power and Light Co. TS change submittal did not include data or analyses from which the thermal response time of Lake Robinson could be determined.

Steady State Calculation

As part of the technical review of the CP&L TS change request, calculations were performed to determine the sensitivity of the steady-state temperature of Lake Robinson to changes in the power output of HB Robinson, Unit 2. These calculations used two sets of weather conditions for the Columbia, SC area, "normal" and "extreme", to evaluate the steady-state UHS temperature resulting from different power levels. These weather conditions are based on data taken prior to 1971 in Columbia, SC. The "normal" conditions represent an average summer month, "extreme" is the worst average monthly conditions expected in a 30 year period.

The maximum steady state service water temperatures for full (102%) power operation were calculated as 95.6 ^{0}F (normal) and 99.5 ^{0}F (extreme). If the

reactor was operated at 50% power, then the steady-state intake temperatures would be 92.9 ^{0}F (normal) and 97.1 ^{0}F (extreme). These calculations show that reducing the reactor power level 52% would only drop the steady-state lake temperature 2.3 to 2.4 ^{0}F.

The model used for these calculations assumed a "deep" cooling impoundment in which a plug-flow surface layer is underlain by a fully mixed return flow layer [Ref. 7]. The discharge canal flow mixes with the return flow to form the surface layer. This lake characterization was confirmed by measurements from a thermal survey of the lake on 7/14/72, as presented in Fig. 3-12 of the Environmental Report [Ref. 6]. The operating parameters used for these calculations were:

surface area	2280	acres	[Ref. 8, p. 2]
discharge flow rate	506100	gpm	[Ref. 2, p. 2.4.4-1]
temperature rise through condensers at 2300 MWt	20.67	^{0}F	[Ref. 6, p. 3.7]

The heat and mass transfer parameters used were:

	monthly average meteorology (normal)	(extreme) [Ref. 9]
lake equil. temp.	90 ^{0}F	94.5 ^{0}F
surf. heat xfer coefficient	129 Btu/ft^2-day-F	138 Btu/ft^2-day-F

A discharge dilution factor is used to quantify the fraction of the surface flow that is discharged from the condensers and the fraction that is return flow from the underlying layer. This factor is defined as the ratio of total surface layer flow (i.e., the discharge canal flow plus the return flow) divided by discharge canal flow. The smaller this factor (i.e., the less turbulent the discharge) the more efficient the surface cooling. A factor of 1.5 (1 part condenser water, 0.5 parts underlying return flow water) is considered to be a good design [Ref. 10]. However, in the case of Lake Robinson, the discharge is over the weir "at high velocity, thereby promoting rapid turbulent mixing with water in the lake" [Ref. 2, p. 2.4.4-3]. Therefore, a discharge dilution factor of 2 was chosen to represent this rapid turbulent mixing. Field measurements could show this factor to be even larger.

These steady-state calculations are only an indicator of realistic lake temperatures, which are subject to a transient diurnal heat load. It is possible that a more refined transient analysis would demonstrate adequate lake

temperatures and safety system performance, for the actions requested by the applicant. However, in order to have confidence in such a mathematical model of lake behavior, it should be shown to match existing operational parameters (i.e., simulation with actual meteorological and operating conditions should result in lake temperatures which are verified by those measured in the lake).

4.2 CONTAINMENT RESPONSE

4.2.1 Peak Pressure and Temperature

The SW temperature is an input to the containment analysis contained in the Updated Final Safety Analysis Report (UFSAR) Section 6.2 [Ref. 2]. In support of this request for TS change, the peak pressures and temperatures for the limiting large break (LB) loss-of-coolant accident (LOCA) and main steam line break (MSLB) inside containment were analyzed assuming a 97 ^{0}F SW temperature. Westinghouse's approved mass and energy release models for LB LOCA and MSLB were used to generate the input for the containment analysis. This is a change from the methodology previously used for HB Robinson, Unit 2. The new containment analysis also used an approved Westinghouse methodology which was similar to the approach used previously. The following table shows that even though the old and new methodologies had a number of differences, the LB LOCA peak pressure results obtained were comparable, with the new values 0.5 psi less than the old:

Peak LB LOCA Containment Pressure (Psig)		
	Previous	New
Initial	0	1
Peak	40	40.5
Delta	40	39.5
Design	42	42

More significant differences were seen in the MSLB results. These differences resulted from changes which have occurred in MSLB methodology since the previous analysis. Even with these differences, the new MSLB peak pressure of 41.85 psig remained below the containment design pressure, and the temperatures and pressure profiles effectively remain within previously qualified Environmental Qualification (EQ) profiles.

4.2.2 Peak Clad Temperature

The Siemens large break (LB) LOCA methodology was used to determine if a 97 ^{0}F SW temperature had an impact on the LOCA peak clad temperature (PCT) analysis. This methodology is based on 10 CFR 50 Appendix K, and has been

reviewed and approved in accordance with 10 CFR 50.46. The LB LOCA analysis determined that the cumulative effects of the higher SW temperature and other updates to the HB Robinson model had only a very minor impact (–2 ^{0}F) on the LOCA PCT.

4.3 COOLING OF SAFETY-RELATED COMPONENTS

4.3.1 Service Water System

The service water (SW) system is designed to provide cooling water to components necessary for plant safety during normal operation or under accident conditions. The SW system also supplies cooling water to various other heat loads in the primary and secondary portions of the plant. The system is capable of supplying water to the suction of the auxiliary feedwater pumps in the event of loss of other sources.

The following safety-related components are cooled by the SW system, and could potentially be degraded if the SW were to exceed temperature limits:

COMPONENT	SUBCOMPONENTS	MAX. SW TEMP ANALYZED
Emergency Diesel Generators	Air cooler, lube oil cooler, water jacket cooler	97 ^{0}F w/2% Max Tube Plugging. 99 ^{0}F Operation in Alarm Mode.
Containment Air Recirculation Coolers		TS 3.6.5 120 ^{0}F limit is estimated at ~ 97 ^{0}F SW.
Containment Fan Motor Coolers		100 ^{0}F
Safety Injection Pumps	Thrust Bearings	100 ^{0}F
Auxiliary Feedwater Pumps (Steam and Motor Driven)	Bearing oil coolers	100 ^{0}F
Reactor Auxiliary Building Ventilation	SI, CS, RHR, AFW Pump Room Air Coolers	100 ^{0}F
Control Room Water Cooled Condensing Units		N/A (Refrigerator Not Sensitive to Sink Temp.)
CCW Heat Exchangers	Heat loads includes the RHR System and the Spent Fuel Pool Cooling System (SFPCS)	100 ^{0}F

For all components, the licensee's evaluation determined that all of these components could operate normally with service water temperatures up to 97 ^{0}F, and most are able to operate with service water temperatures up to 100 ^{0}F. Two components are limited by SW temperature: the emergency diesel generators (EDGs) and the containment air recirculation coolers (CARCs).

Unlike the other components that are aligned in parallel, (i.e., they draw from a common intake header and return to a common outlet header) the EDGs have three subcomponents in series: an air cooler, a lube oil cooler and a water jacket cooler. All subcomponents must meet design temperature limits. In order to raise the SW temperature from 95 ^{0}F to 97 ^{0}F, CP&L found it necessary to reduce the EDG maximum tube plugging allowance from 5% to 2%. (No EDG heat exchanger tubes are currently plugged). With a 99 ^{0}F SW temperature, the EDGs are predicted to experience high temperature alarms.

Operating experience has shown that the Containment Air Recirculation Coolers (CARCs) maintain the containment temperature within 23 ^{0}F higher than the SW temperature. TS 3.6.5 "Containment Air Temperature", requires the containment temperature to be less than or equal to 120 ^{0}F. Therefore, TS 3.6.5 should not be limiting for 97 ^{0}F SW, but may be if the SW temperature exceeds 97 ^{0}F.

4.3.2 Component Cooling Water System

The CCW system loads includes the Spent Fuel Pool cooling system (SFPCS). As stated above, the SFPCS can maintain normal spent fuel pool discharges within a 150 ^{0}F limit with an elevated SW temperature. However, the CP&L analysis noted that the spent fuel pool will be unable to maintain this temperature in the event that a full core offload (i.e., 4/3 of a core stored in the pool) is required to begin immediately after plant shutdown. If the SFPCS cannot maintain pool temperature within limits during offloading operations, CP&L procedures require that the time interval between fuel discharges be increased to compensate.

4.4 SHUTDOWN RISK SIGNIFICANCE

The loss-of-offsite-power initiating event is significant not only during full power operation, but also during reduced power operation and shutdown modes. Because the emergency diesel generators (EDGs) are cooled by service water, there are potential concerns related to a loss of offsite power occurring during periods of elevated service water temperature, even if the plant is shut down.

The following discussion relates to conditions in which service water temperature is high enough to cause failure of the diesels. The event probabilities estimated below suggest that further evaluation is warranted if high SW temperature can

potentially cause failure of diesels. How high the SW temperature actually needs to be to cause EDG failure must be the subject of a separate evaluation.

Some information about system dependencies and accident sequence progression at full power is available from the H. B. Robinson IPE [Ref. 11]. The following discussion is based on information provided in that IPE, including the dependency matrix (pp. 3-95, 3-96), some frequencies and unavailabilities, and some accident sequence discussion. However, the information needed to verify all details of this discussion is not easily obtained from that IPE. In particular, the IPE does not address shutdown, and comments made here on shutdown are based on general background.

Given a loss of offsite power initiating event, if the temperature of the service water were high enough to cause failure of the diesel generators, then all AC would be lost. Once AC is lost, SW is lost (instead of merely being provided at excessively high temperature). Loss of all AC initiated during power operation would leave at least the steam turbine driven train of AFW potentially operable, provided that its alignment has been changed so that it is self-cooled rather than being cooled by SW (p.3-113). If steam generator cooling were not available, this event would leave all AC-reliant, SW-reliant cooling methods without either AC or SW. This comment applies to the RHR cooling arrangement at a typical plant. Some plants are able to apply passive cooling methods to core heat removal in some shutdown modes, and if the necessary alignments could be accomplished without power, cooling could be maintained even in this postulated event.

The following estimated event frequencies are provided for purposes of illustration. A more detailed evaluation would be needed to furnish a true estimate of core damage probability, conditional on SW temperature being very high.

The frequency of loss of offsite power used in the HBR2 IPE (page 3-14) is 6.1E-2 per year. This converts to an hourly rate of 7E-6 per hour. For the present purposes of establishing a general frequency/probability scale, this number will be used, but it should be noted that the actual frequency could be different during periods of extremely hot weather, and different during shutdown operation.

The effective unavailability of the steam turbine driven train will be estimated by summing the probability of failure to start (1.3 E-2, page 3-228) with the average maintenance unavailability (2.24E-2, page 3-229), giving 3.54 E-2. Additional event probabilities would contribute to the overall unavailability of this pump train; in particular, if a human action is needed to realign cooling after the initiating event, an additional contribution should be assessed (this is mentioned in passing on page 3-264, in connection with the loss of SW sequence). Some valve operation is also required. However, this estimate of unavailability will be used for the present illustration.

Given a SW temperature high enough to fail the diesels, the conditional probability of losing all AC in a 24-hour period is 1.7E-4 (obtained by multiplying 7E-6 by 24). At shutdown, this is essentially the probability of losing active RHR in a 24-hour period due to loss of offsite power, conditional on very high SW temperature. This is not a core damage probability by itself; nonrecovery factors need to be applied (allowing for recovery of offsite power), and depending on mode, steam generator cooling or other means of heat removal might be available.

During full power operation or before substantial cooldown, when steam generator cooling would still be available, the pertinent probability is that of loss of AC in conjunction with failure of the steam-driven pump train. Combining the numbers provided above gives a probability of 5.9E-6 per 24-hour period, for loss of all AC combined with failure of the steam turbine driven pump, resulting from high SW temperature.

This estimate is not a core damage probability by itself, because nonrecovery factors would still need to be applied. In some respects, this sequence would qualitatively resemble a sequence discussed in the IPE [Ref. 11]. Sequence 3, (pages 3-263 and 3-264), is initiated by a loss of service water. This causes failure of the motor driven main and auxiliary feedwater pumps. The loss-of-service water also causes failure of the high-head safety injection system, (which requires cooling that depend on service water), preventing bleed-and-feed cooling and thereby leading to a loss of all core heat removal. The point of this comparison is not that the high-temperature SW would necessarily fail these systems in the way that loss of SW fails them, but simply that losing both auxiliary feedwater and high-head safety injection is modeled in the IPE as leading to core damage.

The above discussion relates to core heat removal. For events initiated at power or very soon after shutdown, it is also possible for RCP seal cooling to be an issue if all AC is lost. For example, sequence 5, discussed on page 3-264, is a blackout-induced RCP seal LOCA.

While these numbers should not be taken directly as core damage probabilities, they do suggest that the core damage potential from high-temperature SW conditions is not insignificant, and does not vanish upon plant shutdown.

5.0 COMPLIANCE WITH REGULATORY REQUIREMENTS

5.1 INTRODUCTION

CP&L presented analysis to show that 97 ^{0}F was an acceptable operating limit for safety-related component performance for normal operation and for limiting post-accident situations (i.e., LOCA and MSLB inside containment). Carolina

Power and Light also performed an analysis to demonstrate that there is sufficient water in Lake Robinson to feed the service water pumps for 22 days.

In order to minimize the calculated peak containment pressure, analytical credit was also taken for a faster required closing time for the main steam isolation valves (MSIVs).

Carolina Power and Light also performed an analysis to demonstrate that there is sufficient water in Lake Robinson to feed the service water pumps for 22 days. The natural evaporation rate used was reasonable and the forced evaporation (due to the excess heat introduced into the lake) rate was conservatively estimated.

5.2 FINDINGS

The analyses performed by Carolina Power and Light have demonstrated that

1) The safety-related components cooled by the SW and CCW systems can perform their safety-related function during normal operation and under accident conditions if the SW temperature does not exceed 97 ^{0}F.

2) There is sufficient water in Lake Robinson to feed the service water pumps for 22 days if the UHS temperature does not exceed 97 ^{0}F. The natural evaporation rate used was reasonable and the forced evaporation (due to the excess heat introduced into the lake) rate was conservatively estimated.

3) The increase in UHS temperature to will not result in containment peak pressures or temperatures exceeding their design limits, if the UHS temperature does not exceed 97 ^{0}F.

4) Approved methodologies were used for the containment and LOCA analyses. These methodologies were applied in an appropriate and conservative manner.

6.0 CONCLUSIONS

This evaluation concludes that while the proposed TS change decreases the heat capacity of the UHS and the cooling capacity of safety-related equipment attached to the service water and closed cooling water systems, analysis has shown that containment peak pressure (GDC 44) requirements will still be met when the SW temperature is 97 ^{0}F or less.

However, if the lake temperature is permitted to closely approach the 99 ^{0}F limit during the course of a particular day, meteorological effects may cause the SW temperature to exceed the 99 ^{0}F temperature limit on a subsequent day. Because of the long response time and the relative insensitivity of the lake temperature to a change in reactor power level, there are no definitive actions that the operator could take to control the lake temperatures within the prescribed limit. Specific safety-related components, most notably the emergency diesel generators (EDGs), are not qualified for operation above 99 ^{0}F. Loss of the EDGs increases the risk of core melt whether or not the plant is promptly shut down.

7.0 REFERENCES

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