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NLS2015040
April 15, 2015

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Technical Specification Bases Changes
Cooper Nuclear Station, Docket No. 50-298, DPR-46

Dear Sir or Madam:

The purpose of this letter is to provide changes to the Cooper Nuclear Station (CNS) Technical Specification Bases implemented without prior Nuclear Regulatory Commission approval. In accordance with the requirements of CNS Technical Specification 5.5.10.d, these changes are provided on a frequency consistent with 10 CFR 50.71(e). The enclosed Bases changes are for the time period from April 2, 2013, through March 10, 2015. Also enclosed are filing instructions and an updated List of Effective Pages for the CNS Technical Specification Bases.

This letter contains no commitments.

If you have any questions regarding this submittal, please contact me at (402) 825-2788.

Sincerely,

Jim Shaw
Licensing Manager

/lb

Enclosure: Technical Specification Bases Changes

cc: Regional Administrator, w/enclosure
USNRC - Region IV

Cooper Project Manager, w/enclosure
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector, w/enclosure (per controlled document distribution)
USNRC - CNS

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NRR

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NLS2015040
ENCLOSURE

TECHNICAL SPECIFICATION
BASES CHANGES

FILING INSTRUCTIONS

TECHNICAL SPECIFICATION BASES

REMOVE	**INSERT**
List of Effective Pages - Bases	
1 through 7 (dated 02/07/13)	1 through 7 (dated 11/13/14)
Bases Pages	
B 3.3-77 (dated 11/25/12)	B 3.3-77 (dated 02/24/14)
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B 3.4-46 (dated 04/11/06)	B 3.4-46 (dated 04/23/13)
B 3.4-51 (Rev. 0)	B 3.4-51 (dated 04/23/13)
B 3.4-52 (dated 08/11/04)	B 3.4-52 (dated 04/23/13)
B 3.7-10 (dated 11/25/12)	B 3.7-10 (dated 09/18/14)

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B 3.9-10	0	B 3.10-27	6/10/99
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B 3.9-17	0	B 3.10-34	0
B 3.9-18	12/18/03	B 3.10-35	0
B 3.9-19	10/05/06	B 3.10-36	0
B 3.9-20	0	B 3.10-37	0
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B 3.9-24	0		
B 3.9-25	0		
B 3.9-26	0		
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B 3.10-9	0		
B 3.10-10	0		
B 3.10-11	0		
B 3.10-12	0		
B 3.10-13	0		
B 3.10-14	0		
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B 3.10-22	0		

Table B 3.3.3.2-1 (page 1 of 2)
Alternate Shutdown System Instrumentation/Controls

	FUNCTION	REQUIRED NUMBER OF CHANNELS
Instrument Parameter		
1.	HPCI Turbine Steam Inlet Pressure (Reactor Pressure)	1
2.	HPCI Pump Discharge Flow	1
3.	Fuel Zone Level	1
4.	Wide Range Level	1
5.	Torus Level	1
6.	Emergency Condensate Storage Tank (ECST) Level	1
7.	RHR System Loop B Flow	1
8.	Torus Temperature	1
Transfer/Control Parameter for ASD Room/DG Rooms		
9.	HPCI Turbine	1
10.	HPCI Auxiliary Lube Oil Pump	1
11.	HPCI Fan Coil Unit	1
12.	HPCI Flow Controller	1
13.	HPCI Flow Transmitter (FT-82)	1
14.	HPCI Turbine Steam Supply Isolation, HPCI-MO-14	1
15.	HPCI Steam Supply Inboard Isolation, HPCI-MO-15	1
16.	HPCI Steam Supply Outboard Isolation, HPCI-MO-16	1
17.	HPCI Pump Suction ECST, HPCI-MO-17	1
18.	HPCI Injection HPCI-MO-19	1
19.	HPCI Pump Discharge, HPCI-MO-20	1
20.	HPCI Test Bypass to ECST, HPCI-MO-21	1
21.	HPCI Test Bypass Shutoff, HPCI-MO-24	1
22.	HPCI Minimum Flow Bypass, HPCI-MO-25	1
23.	HPCI Pump Suction from Suppression Pool, HPCI-MO-58	1
24.	RHR HX B Outlet, RHR-MO-12B	1
25.	RHR Pump D Suction from Torus, RHR-MO-13D	1
26.	RHR Pump D Shutdown Cooling Suction, RHR-MO-15D	1
27.	RHR Pump B and D Minimum Flow, RHR-MO-16B	1
28.	RHR Loop B Injection Outboard Throttle, RHR-MO-27B	1
29.	Suppression Chamber Cooling Loop B Inboard Isolation, RHR-MO-34B	1
30.	Suppression Chamber Cooling Loop B Outboard Isolation, RHR-MO-39B	1

(continued)

Table B 3.3.3.2-1 (page 2 of 2)
Alternate Shutdown System Instrumentation/Controls

	FUNCTION	REQUIRED NUMBER OF CHANNELS
Transfer/Control Parameter for ASD Room/DG Rooms (continued)		
31.	RHR HX B Inlet, RHR-MO-65B	1
32.	RHR HX B Bypass Throttle, RHR-MO-66B	1
33.	REC Pump 1C	1
34.	REC Pump 1D	1
35.	Safety Relief Valve Main Steamline-C, MS-RV-71E	1
36.	Safety Relief Valve Main Steamline-C, MS-RV-71F	1
37.	Safety Relief Valve Main Steamline-D, MS-RV-71G	1
38.	Diesel Generator No. 2 Engine - Local Isolation, Control and Indication	1
39.	Diesel Generator No. 2 Generator Relay and Control - Local Isolation, Control and Indication	1
Transfer/Control Parameter for Local Auxiliary Shutdown Panels (LASPs)/DC Starters		
40.	RHR-MO-20, RHR Crossheader Shutoff	1
41.	RHR-MO-25B, RHR Loop B Inboard Injection	1
42.	RHR-MO-26B, RHR Loop B Drywell Spray Outboard Throttle	1
43.	RHR-MO-57, RHR Discharge to Radwaste Outboard Throttle	1
44.	REC-MO-695, REC Critical Loop Supply Crossover	1
45.	REC-MO-714, REC South Critical Loop Supply	1
46.	SW-MO-37, SW Loop Crosstie	1
47.	SW-MO-89B, RHR HX B Service Water Outlet	1
48.	SW-MO-651, REC HX B Service Water Outlet	1
49.	SW-MO-887, SW Emergency Supply to REC South Critical Loop	1
50.	SW-MO-889, SW Return From REC South Critical Loop	1
51.	MS-MO-74, Steam Line Drain Inboard Isolation	1
52.	RCIC-MO-15, RCIC Steam Supply Inboard Isolation	1
53.	RWCU-MO-15, RWCU Supply Inboard Isolation	1

BASES

SURVEILLANCE REQUIREMENTS (continued)

Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. USAR, Section VII-9.5.4.2.
2. 10 CFR 50.36(c)(2)(ii).
3. GENE-770-06-1, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," February 1991.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

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

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100 (Ref. 1) for the Control Rod Drop and Main Steam Line Break accidents, and 10 CFR 50.67, "Accident Source Term," (Ref. 6) for the Fuel Handling and Loss-of-Coolant accidents.

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 100 limit for the Control Rod Drop and Main Steam Line Break accidents, and within the 10 CFR 50.67, "Accident Source Term," (Ref. 6) limit for the Fuel Handling and Loss-of-Coolant accidents.

APPLICABLE SAFETY ANALYSES

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

This MSLB release forms the basis for determining offsite and control room doses (Refs. 2 and 3). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary,

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APPLICABLE SAFETY ANALYSES (continued)

P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).

LCO

The elements of this LCO are:

a. RCS pressure and temperature (Beltline, Bottom Head, and Upper Vessel) are within the applicable limits of Figure 3.4.9-1 and Figure 3.4.9-2, and heatup or cooldown rates are ≤ 100°F when averaged over a one hour period during RCS heatup, cooldown, and inservice leak and hydrostatic testing (The Adjusted Reference Temperature (ART) beltline region must be determined from Figure 3.4.9-2. During RCS heatup and cooldown operation (i.e., not critical and not performing inservice leak or hydrostatic testing) verify RCS pressure and temperature are within the applicable limits specified in Figure 3.4.9-1. During RCS inservice leak and hydrostatic testing verify RCS pressure and temperature are within the applicable limits specified in Figure 3.4.9-2;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ≤ 145°F during recirculation pump startup;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup;

d. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-3, prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are > 70°F when tensioning the reactor vessel head bolting studs.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.3 is to compare the bottom head drain temperature to the RPV steam dome saturation temperature.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

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

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7

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

The flange temperatures must be verified to be above the limits within 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 90°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the specified limits.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be

BASES

SURVEILLANCE REQUIREMENTS (continued)

initiated 30 minutes after RCS temperature ≤ 80°F in MODE 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature ≤ 90°F in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be within the specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. BWRVIP-86-A, October 2002.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. USAR, Section IV-2.6.
7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
8. 10 CFR 50.36(c)(2)(ii).
9. USAR, Appendix G.
10. ASME XI Code Case N-640.

BASES

SURVEILLANCE REQUIREMENTS (continued)

position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR is modified by a Note indicating that isolation of the SW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the SW System. As such, when all SW pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the SW System is still OPERABLE.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.2.4

This SR verifies that the automatic isolation valves of the SW System will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. The initiation signal is caused by low SW header pressure (approximately 20 psig). This SR also verifies that the SW pumps with their mode selector switch in AUTO will automatically start on a low SW header pressure.

Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NEDC 94-255, "Hydraulic Evaluation of Opening in Intake Structure Guide Wall," June 14, 1995.
2. USAR, Chapter V.
3. USAR, Chapter XIV.
4. 10 CFR 50.36(c)(2)(ii).
5. NEDC 00-095E, "CNS Reactor Building Post-LOCA Heating Analysis," May 28, 2010.