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NLS2006002
January 16, 2006

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

Subject: Revised and Supplemental Pages to License Amendment Request for Application of the Alternative Source Term for Reevaluation of the Fuel Handling Accident Dose Consequences
Cooper Nuclear Station, Docket No. 50-298, DPR-46

Reference: Letter to U. S. Nuclear Regulatory Commission from R. Edington (Nebraska Public Power District) dated September 29, 2005, "License Amendment Request for Application of the Alternative Source Term for Reevaluation of the Fuel Handling Accident Dose Consequences" (NLS2005075).

The purpose of this letter is for the Nebraska Public Power District to provide revised and supplemental pages to the referenced License Amendment Request. Enclosure 1 contains a revision to Pages 52 and 53 of the License Amendment Request to address 1967 Draft GDCs that were found to pertain to containment and monitoring of accident releases. License Amendments 212 and 213 were received subsequent to the submittal of the referenced License Amendment Request and affected several of the associated Technical Specification (TS) pages provided in that application. Accordingly, replacement markup and clean TS pages are provided for the affected pages in the Enclosure.

Several TS Bases changes were identified that are not associated with TSTF-51 Revision 2, but should nonetheless be included for information. They address the superseding requirements of 10 CFR 50.67 over 10 CFR 100 and General Design Criterion (GDC) 19 for offsite and onsite radiological dose limitations for the Fuel Handling Accident. The supplemental TS Bases pages are provided in the Enclosure.

These revised and supplemental pages have no impact on the No Significant Hazards or Environmental Considerations, or on the technical merits of the referenced License Amendment Request.

Should you have any questions concerning this matter, please contact Paul Fleming, Licensing Manager, at (402) 825-2774.

A001

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 01/16/2006
(Date)

Randall K. Edington
Vice President - Nuclear and
Chief Nuclear Officer

/wv

Enclosure

cc: Regional Administrator w/enclosure
USNRC - Region IV

Senior Project Manager w/enclosure
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/enclosure
USNRC – CNS

Nebraska Health and Human Services w/enclosure
Department of Regulation and Licensure

NPG Distribution w/o enclosure

CNS Records w/enclosure

Enclosure 1

Listing of Replacement/Supplemental Pages to NLS2005075

The following pages are revisions to the Attachments provided in NLS2005075:

Attachment 1:	Page 52 of 53 Page 53 of 53
Attachment 2:	Technical Specifications Page 3.3-57 Technical Specifications Page 3.3-63 Technical Specifications Page 3.6-40
Attachment 3:	Technical Specifications Page 3.3-57 Technical Specifications Page 3.3-63 Technical Specifications Page 3.6-40

The following pages are supplemental to the Attachment provided in NLS2005075:

Attachment 4:	Technical Specifications Bases Page B 3.3-185 Technical Specifications Bases Page B 3.7-25 Technical Specifications Bases Page B 3.7-27

Question 6: A TS change is proposed that will not require Secondary Containment operability during the movement of fuel assemblies that have not been "recently" irradiated. The FHA analysis assumes the release to the control room intake and the environment is through the Reactor Building vent. Justify that that release point is an appropriately conservative assumption given that the Secondary Containment may be inoperable.

Response: As discussed in Question 2, the NPPD approach to Secondary Containment breach control during fuel handling operations ensures that breaches can be sealed following an FHA, and furthermore, that a train of SGT can be placed in service to provide a filtered, monitored, and elevated release. Thus, the RG 1.183 assumption of a 2-hour release to the environment is only plausible if there is Reactor Building exhaust fan flow via the Reactor Building vent. As discussed in Question 5, ventilation exhaust flow will be in service during fuel handling operations to ensure CREFS initiation.

If there were no fan flow, there would not be a significant driving force from the refuel floor to the environment for the FHA release. A release from the Reactor Building railroad airlock doors was also considered under these conditions, but physical interlocks and security considerations prevent both Reactor Building railroad airlock doors from being open concurrently. Therefore, without forced ventilation flow, it would be expected that any FHA radioactive release would occur at a much slower rate than with forced ventilation flow via the Reactor Building vent. Also, as discussed in Question 5, fuel handling operations involving irradiated fuel with less than a 7 day decay time during periods without Reactor Building exhaust fan flow or without an Operable Secondary Containment with a SGT system in operation would require CREFS to be in operation, further mitigating the dose consequences of an FHA under these conditions. Accordingly, consistent with regulatory precedent (References 7.2-1 and 7.2-2) a ground level release from the Reactor Building vent when Secondary Containment and the SGT System are inoperable, or when a Reactor Building exhaust fan is not in operation, provides an appropriately conservative release point based on the information provided above.

Question 7: TS changes are proposed that eliminate the operability requirements for Secondary Containment and Standby Gas Treatment System operability during the movement of fuel assemblies that have not been "recently" irradiated. How will the intent of GDC 61 be met for controlling containment releases via confinement or filtering, and GDC 64 in monitoring releases?

Response: As discussed in Section 5.2, the construction of CNS predated the 1971 issuance of 10 CFR 50 Appendix A. NPPD has made no commitments to meet 1971 GDCs 61 and 64. However, 1967 GDCs 69 and 17 provide analogous criteria to 1971 GDCs 61 and 64 with respect to suitable containment and assuring a monitored release following an FHA. The FHA dose consequences demonstrate that undue amounts of radioactivity will not be released to the public environs, even with Secondary Containment inoperable. With a Reactor Building exhaust fan or SGT System fan in operation while moving recently irradiated fuel (see commitment in Question 5), a monitored release is assured. Otherwise, the NPPD Secondary Containment breach control strategy (see Question 2) is designed to ensure that the Secondary Containment function, including a confined, monitored and filtered release via the SGT System, can be restored following an FHA.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low (Level 2)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	$\geq$ - 42 inches
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	$\leq$ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	$\leq$ 49 mR/hr

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

(b) During ~~CORE ALTERATIONS and during~~ movement of recently irradiated fuel assemblies in secondary containment.

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Filter System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low (Level 2)	1,2,3, (a)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≥ - 42 inches
2. Drywell Pressure - High	1,2,3	2	SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 49 mR/hr

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

(b) During ~~CORE ALTERATIONS and during~~ movement of <u>recently</u> irradiated fuel assemblies in the secondary containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	18 months
SR 3.6.4.3.4	Verify the SGT units cross tie damper is in the correct position, and each SGT room air supply check valve and SGT dilution air shutoff valve can be opened.	18 months

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low (Level 2)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ - 42 inches
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 49 mR/hr

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

(b) During movement of recently irradiated fuel assemblies in secondary containment.

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Filter System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low (Level 2)	1,2,3, (a)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≥ - 42 inches
2. Drywell Pressure - High	1,2,3	2	SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 49 mR/hr

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

(b) During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for $\geq$ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	18 months
SR 3.6.4.3.4	Verify the SGT units cross tie damper is in the correct position, and each SGT room air supply check valve and SGT dilution air shutoff valve can be opened.	18 months

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Filter (CREF) System Instrumentation

BASES

BACKGROUND

The CREF System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The instrumentation and controls for the CREF System automatically isolate the normal ventilation intake and initiate action to pressurize the main control room and filter incoming air to minimize the infiltration of radioactive material into the control room environment.

In the event of a loss of coolant accident (LOCA) signal (Reactor Vessel Water Level — Low Low, Level 2 or Drywell Pressure — High) or Reactor Building Ventilation Exhaust Plenum Radiation — High signal, the normal control room inlet supply damper closes and the CREF System is automatically started in the emergency bypass mode. The air drawn in from the outside passes through a high efficiency filter and a charcoal filter in sufficient volume to maintain the control room slightly pressurized with respect to the adjacent areas.

The CREF System instrumentation has two trip systems. Each trip system includes the sensors, relays, and switches necessary to cause initiation of the CREF System. Each trip system receives input from each of the Functions listed above (each sensor sends a signal to both trip systems). The Reactor Vessel Water Level — Low Low, Level 2, Drywell Pressure — High, and Reactor Building Ventilation Exhaust Plenum Radiation — High are each arranged in a one-out-of-two taken twice logic for each trip system. The channels include electronic and electrical equipment (e.g., switches and trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREF System initiation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the control room is explicitly assumed for certain accidents as discussed in the USAR safety analyses (Refs. 1, 2, and 3). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents that assume CREF System operation, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A, or 10 CFR 50.67 (Fuel Handling Accident only).

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the USAR, Section X-3.0 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Section XIV-6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated ~~whole body and thyroid doses~~ total effective dose equivalent at the exclusion area and low population zone boundaries) are ~~≤ 25% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800~~ within 10 CFR 50.67 limits (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the USAR, Section XIV-6.1 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. USAR, Section X-3.0.
2. USAR, Section XIV-6.4.
3. ~~10 CFR 100~~ Not used.
4. ~~NUREG-0800, Section 15.7.4, Revision 1, July 1981~~ 10 CFR 50.67.
5. Regulatory Guide ~~1.25, March 1972~~ 1.183, July 2000.
6. USAR, Section XIV-6.1.
7. 10 CFR 50.36(c)(2)(ii).

Correspondence Number: NLS2006002

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		