September 5, 2006

Mr. Randall K. Edington
Vice President-Nuclear and CNO
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: APPLICATION OF THE ALTERNATIVE SOURCE TERM FOR REEVALUATION OF THE FUEL HANDLING ACCIDENT DOSE CONSEQUENCES AND RELATED TECHNICAL SPECIFICATION CHANGES (TAC NO. MC8566)

Dear Mr. Edington:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 222 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 29, 2005, as supplemented by letters dated January 16, and April 7, 2006.

The amendment replaces the current accident source term used in the fuel handling accident (FHA) design-basis radiological analysis with an alternative source term (AST) pursuant to Title 10 of the *Code of Federal Regulations*, Part 50.67, "Accident Source Term." The licensee performed the revised FHA analysis in support of TS changes related to refueling operations.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Brian Benney, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 222 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 222
License No. DPR-46

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

   A. The application for amendment by the Nebraska Public Power District (the licensee) dated September 29, 2005, as supplemented by letters dated January 16, and April 7, 2006, 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 license 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-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 222, are hereby incorporated in the license. The Nebraska Public Power District shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 5, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 222

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

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	INSERT
3.3-57	3.3-57
3.3-63	3.3-63
3.6-32	3.6-32
3.6-33	3.6-33
3.6-34	3.6-34
3.6-36	3.6-36
3.6-38	3.6-38
3.6-39	3.6-39
3.6-40	3.6-40
3.7-8	3.7-8
3.7-9	3.7-9

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated September 29, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052770499), as supplemented by letters dated January 16, and April 7, 2006 (ADAMS Accession Nos. ML060200292 and ML061020078, respectively), Nebraska Public Power District (the licensee) requested a license amendment for the Cooper Nuclear Station (CNS). The supplements dated January 16 and April 7, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published on January 3, 2006 (71 FR 149), in the *Federal Register*.

This amendment will replace the current accident source term used in calculating the fuel handling accident (FHA) dose consequences with the alternative source term (AST) in accordance with Title 10 of the *Code of Federal Regulations*, Part 50.67 (10 CFR 50.67), "Accident Source Term." The FHA re-analysis also involved several changes in selected analysis assumptions. The licensee performed the FHA re-analysis in support of technical specification (TS) changes related to refueling operations.

The following TSs are proposed to be revised:

- TS 3.3.6.2, Secondary Containment Isolation Instrumentation
- TS 3.3.7.1, Control Room Emergency Filter (CREF) System Instrumentation
- TS 3.6.4.1, Secondary Containment
- TS 3.6.4.2, Secondary Containment Isolation Valves (SCIVs)
- TS 3.6.4.3, Standby Gas Treatment (SGT) System
- TS 3.7.4, CREF System

The proposed TS changes would eliminate operability requirements for (1) the secondary containment, the SCIVs, the SGT system, and the secondary containment isolation instrumentation when handling irradiated fuel that has decayed for at least 24 hours since

critical reactor operations and (2) the CREF system and its initiation instrumentation when handling irradiated fuel that has decayed for at least 7 days. The proposed changes are based on Technical Specifications Task Force Traveler 51 (TSTF–51), "Revised Containment Requirements During Handling Irradiated Fuel and Core Alterations."

The licensee stated that this license amendment is being requested to reduce the duration and cost of planned plant outages.

2.0 REGULATORY EVALUATION

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed design-basis accident (DBA) radiological consequences and the acceptability of the revised analyses results. The regulatory requirements are the accident dose criteria in 10 CFR 50.67, "Accident Source Term," as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room," as supplemented by Section 6.4 of NUREG–0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." Except where the licensee proposed a suitable alternative, the U.S. Nuclear Regulatory Commission (NRC) staff utilized the regulatory guidance provided in SRP Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," in performing this review. The NRC staff also considered relevant information in the CNS updated safety analysis report, TSs, and TSTF–51.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of DBAs, performed by the licensee in support of its proposed license amendment. Information regarding these analyses was provided in Attachment 1 and Enclosure 1 of the submittal dated September 29, 2005, and in supplementary letters dated January 16, and April 7, 2006. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to evaluate the impacts of the revised design-basis FHA analyses. The findings of this SE are based on the descriptions of the licensee’s analyses and other supporting information docketed by the licensee. However, the NRC staff also performed independent calculations to confirm the conservatism of the licensee’s analyses.

TSTF–51, Revision 2, allows for removal of the TS requirements for engineered safeguards features, such as secondary containment and SGT, to be operable during movement of fuel, once sufficient radioactive decay has taken place to ensure that offsite doses remain well within (i.e., 25 percent of) 10 CFR Part 100 limits. TSTF–51 placed the following reviewer’s note in the basis for the standard TS 3.6.4.1, "Secondary Containment":

> The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by analysis that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10 CFR [Part] 100).

> Additionally, licensees adding the term "recently" must make the following commitment which is consistent with draft NUMARC 93–01, Revision 3, Section 11.2.6, "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment – Primary (PWR [pressurized-water reactor])/Secondary (BWR [boiling-water reactor])."
>
> The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel:
>
> - During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91–06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the technical specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.
>
> - A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.
>
> The purpose of the ‘prompt methods’ mentioned above are to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

When draft NUMARC 93–01, Revision 3, was issued in the final version, Section 11.2.6 was renumbered to Section 11.3.6 and the guidelines quoted above were numbered 11.3.6.5 and modified slightly as follows:

> . . . the following guidelines should be included in the assessment of systems removed from service:
>
> - During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91–06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS [reactor coolant system] decays fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.
>
> - A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw

the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

The NRC staff considers the final version and the draft version of this section of NUMARC 93–01, Revision 3, to be functionally equivalent. The licensee committed to implement the guidelines of Section 11.3.6.5 of NUMARC 93–01, Revision 3, within 30 days of the implementation of this license amendment. The NRC staff finds this meets the commitment required by TSTF–51, Revision 2.

Because the licensee has applied to implement an AST selectively for the FHA, the regulatory dose criteria of 10 CFR 50.67 are used in lieu of the 10 CFR Part 100 dose limits in determining acceptability of the changes.

3.1 Radiological Consequences of the Fuel Handling Accident

The licensee evaluated the dose consequences of an FHA following both a 24-hour decay time since reactor shutdown and a 7-day decay time since reactor shutdown. This evaluation was based on the AST guidelines outlined in NUREG–1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and RG 1.183, as well as the NRC computer code RADTRAD Version 3.03 (NUREG/CR–6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," including Supplements 1 and 2). To support its TS changes, the licensee's evaluation demonstrated that the radiological doses at the exclusion area boundary (EAB), low population zone (LPZ), and in the control room (CR) are within regulatory limits without crediting the operability of secondary containment, secondary containment isolation valves, the SGT system, or secondary containment isolation instrumentation after a 24-hour decay period following reactor shutdown. Likewise, the licensee's evaluation demonstrated that control room radiological doses are within regulatory limits without crediting the operability of the CREF system and CREF system instrumentation after a 7-day decay period following reactor shutdown.

The licensee's limiting postulated FHA event assumed a fuel assembly is dropped into the reactor core during refueling operations from a height of 32.95 feet, which is the maximum height allowed by the fuel handling equipment. The resulting impact of the fuel assembly onto the top of the core is assumed to damage 151, GE14, 10×10 fuel rods (representing approximately 0.3133 percent of the core) causing a gap release of radionuclides to the water pool above the core. This event could also occur over the spent fuel pool. However, the licensee states that significantly fewer (i.e., 48) fuel rods would be damaged in the latter case, due to reduced drop height. Since both the reactor cavity and spent fuel pool are located in the reactor building, an FHA in either the reactor cavity or the spent fuel pool is assumed to have the same potential release pathways from the reactor building to the environment.

Given that the primary containment remains open during refueling, the licensee assumed the radionuclides released as a result of the FHA pass through the water in the reactor cavity and enter the reactor building refueling floor atmosphere instantaneously. The water pool above the core serves as a barrier to the release of a significant amount of radionuclides. The remaining radionuclides that become airborne in the reactor building are assumed to be released to the outside environment over a 2-hour period.

The licensee determined the inventory of fission products in the fuel rods based on the maximum full power operation of the core with an assumed core power equal to the current licensed rated thermal power of 2381 megawatts thermal (MWt). The licensee multiplied this value by a factor of 1.02 to account for maximum possible measurement uncertainty as required by Appendix K to 10 CFR Part 50 for nuclear reactor power operation. The licensee assumed the core had operated at this power level for an extended period of time such that fission product equilibrium was reached. For radionuclides that have not reached equilibrium, the core inventory at time of shutdown was used. To account for differences in power distribution across the core, the licensee applied a peaking factor of 2.0 to the average inventory.

The majority of the fission products produced during plant operation remain within the fuel pellet; however, some migrate to void spaces, known as "gap," within the fuel rods. The licensee calculated the activity in the gap using the non-loss-of-coolant-accident gap inventory fractions presented in Table 3 of RG 1.183.[1] The licensee assumed that as the resulting gap activity release rises through the overlaying water, the elemental halogen species (i.e., Bromine and Iodine) are scrubbed by the water column, resulting in an effective halogen decontamination factor of 200 (i.e., 99.5 percent of the total halogen release from the damaged fuel rods is retained by the water). No decontamination of noble gases (i.e., Krypton and Xenon) was assumed.

The guidance in RG 1.183 allows an effective halogen decontamination factor of 200 when the overlaying water column is at least 23 feet. This pre-condition is met for the reactor cavity but not for the spent fuel pool where the water depth is only 21 feet 6 inches. The licensee stated that the implied reduction in scrubbing efficiency is offset by the reduced number of fuel rods (i.e., 48 versus 151) that are projected to be damaged by a fuel assembly drop over the spent fuel pool. The effective decontamination factor in RG 1.183 is based on an exponential function where more scrubbing occurs at the bottom of the water column than at the top. As such, a pool level of 21 feet 6 inches (a reduction of about 6.5 percent in pool depth) would result in a reduction in scrubbing efficiency of less than 6.5 percent. This is less than the 68 percent reduction in the amount of damaged rods, and hence the radionuclides released from the spent fuel pool are less than from the reactor cavity pool. The NRC staff finds the licensee's conclusion, that the consequences of an FHA over the reactor cavity bounds those for an FHA over the spent fuel pool, to be acceptable.

The licensee initially identified two possible release points for the FHA when the reactor building is open to the environment and the SGT system and containment are inoperable: the reactor building vent and the reactor building railroad airlock doors. Since physical interlocks and security considerations prevent both reactor building railroad airlock doors from being open concurrently, the licensee modeled FHA releases as occurring through the reactor building vent.[2]

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1 Footnote 11 to RG 1.183 states that the release fractions listed in RG 1.183 Table 3 are acceptable for use with fuels having a peak burnup up to 62 gigawatt-days per metric ton uranium (GWD/MTU), provided that the maximum linear heat generation rate does not exceed the 6.3 kilowatt per foot peak rod average power for burnup exceeding 54 GWD/MTU. The licensee's response to Question 1 in Attachment 1 to its letter dated April 7, 2006, states that the current core will remain within the constraints of this footnote and core reload design control measures will ensure that future cores remain within the burnup limitation.

2 In Question #2 of its request for additional information (RAI) letter dated February 9, 2006, the NRC staff asked the licensee to (1) identify all the possible release points for the FHA when the reactor building door is open to the environment and the standby gas treatment system and secondary containment are inoperable, and (2) justify why the reactor building vent release pathway is assumed to bound all of these

The licensee's FHA evaluation of control room doses for the 24-hour decay time case credited the operability of the CREF system and CREF system instrumentation. During both normal and radiological emergency modes of operation, the control room envelope is positively pressurized and the return air from the control room envelope is recirculated without filtration. During the first 1 minute of the event, the licensee assumed a normal unfiltered inflow of 3235 cubic feet per minute (cfm). The CREF system was then assumed to actuate due to high radiation detected in the reactor building exhaust plenum.[3] For the remaining duration of the event, the licensee assumed an emergency filtered inflow of 810 cfm.[4] The licensee also assumed an unfiltered inleakage of 400 cfm throughout the entire duration of the event.[5]

The licensee's FHA evaluation of control room doses for the 7-day decay time case did not credit the availability of the CREF system. For this scenario, a normal unfiltered inflow of 3635 cfm (which includes 400 cfm inleakage) was used for the duration of the accident.

The licensee also qualitatively assessed the potential gamma shine dose from external sources to the control room during the FHA. The radiation sources external to the control room include the airborne external cloud and CREF system filters located within the control room envelope. The licensee concluded that the cloud shine and CREF system shine control room doses would be a fraction of the inhalation doses and the resulting total dose would still be below regulatory criteria.

The NRC staff reviewed the information provided in the licensee's submittal, as supplemented, and also performed an independent calculation that confirmed the licensee's dose results. The assumptions used by the licensee (which are listed in Table 1 of this SE) were found acceptable by the NRC staff. The licensee's calculated dose results are given in Table 2. The NRC staff has found the licensee's calculated doses acceptable because they are within the SRP 15.0.1 radiological dose acceptance criteria for an FHA. These criteria are 6.3 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the EAB for the worst 2 hours, 6.3 rem TEDE at the LPZ for the duration of the accident, and 5 rem TEDE in the CR for the duration of the accident.

3.2 Atmospheric Dispersion Estimates

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other possible release pathways for the control room dose assessment. In its letter response dated April 7, 2006, the licensee identified a number of additional potential release pathways to the environment assuming an FHA on the refuel floor but stated that the reactor building vent remained bounding because of its direct path to the outside environment and relatively short distance to the control room intake. For additional conservatism, the licensee made a commitment to keep the reactor building roof hatch closed during movement of irradiated fuel in the secondary containment.

3 The licensee evaluated the ability of the reactor building exhaust plenum radiation monitor to provide the necessary CREF system initiation signal for the FHA. This setpoint assessment is predicated on having ventilation flow in the exhaust plenum at the start of the FHA. Therefore, if the CREF system is not in operation, the licensee has committed to having either a reactor building exhaust fan or an SGT system fan in operation prior to conducting fuel handling operations when less than a 7-day decay time has elapsed.

4 Although the CREF system is currently specified as having a 95 percent efficiency for all iodine species, the licensee modeled a filtered efficiency of 89 percent for all iodine species in its analyses in order to provide for future analytical operational margin.

5 The unfiltered inleakage assumption of 400 cfm is conservative when compared to the inleakage value of 64 cfm measured by the licensee during tracer gas testing for the radiological emergency mode of operation as reported in its in September 30, 2004, response to NRC Generic Letter 2003–01, "Control Room Habitability."

3.2.1 Control Room χ/Q Values

The licensee's control room dose analyses modeled FHA releases as occurring through the reactor building vent and assumed a single receptor point at the CR air intake for both unfiltered and filtered inflow and unfiltered inleakage. The licensee used existing CR atmospheric dispersion factors (χ/Q values) that had been submitted for NRC staff review as part of the submittal for the current FHA of record (License Amendment No.187). These χ/Q values, which are presented in Table 1, were listed in the calculation provided in Enclosure 1 to the licensee's letter dated September 14, 2001. They were generated using the NRC computer code ARCON96 (NUREG/CR–6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and onsite meteorological data collected during calendar years 1994–1998.

The release assumptions used for the currently licensed FHA and the one proposed in this license amendment request are somewhat different. The previous FHA assumed a 90-second unfiltered ground level reactor building vent release followed by an elevated filtered stack release for the balance of the 30-day duration, while the new analyses assumes a 30-day unfiltered reactor building vent ground level release. In its letter dated September 14, 2001, the licensee presented χ/Q values calculated for the reactor building vent release assuming three different values for the reactor building vent flow rate (i.e., 51,333 cfm, 9500 cfm, and 1780 cfm). The three flow rates modeled a reactor building exhaust fan coast down after a primary containment isolation system trip on high radiation in the reactor building exhaust plenum. For the new analyses, the licensee elected to use the highest (i.e., most conservative) of the three sets of χ/Q values, which are the χ/Q values associated with the lowest (i.e.,1780 cfm) reactor building vent flow rate.[6] Although the 1780 cfm χ/Q values are not specifically cited in the License Amendment No. 187 Safety Evaluation (as they were not applicable until after the 90-second Reactor Building ground level release had terminated), they are based on the same previously approved 1994–1998 meteorological data and ARCON96 methodology. For these reasons, the NRC staff has concluded that the CR χ/Q values presented in Table 1 are acceptable for use in the CR dose assessment submitted in support of this amendment.

3.2.2 Offsite χ/Q Values

The χ/Q values used by the licensee for the EAB and LPZ dose consequence assessments are based on ground level releases from the reactor building vent calculated using site specific inputs and methodology described in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." These χ/Q values are presented in Table 1. The calculation documenting these χ/Q values was previously submitted to the NRC staff in Enclosure 6 to a letter dated February 28, 2001. A subset of these χ/Q values (i.e., the χ/Q values listed as the 0–2 hour EAB and LPZ values) was approved by the NRC staff for use in License Amendment No. 187. The remaining EAB and LPZ χ/Q values approved in License Amendment No. 187 are not applicable to the new analyses because they represent elevated stack releases. Since the remaining offsite χ/Q values listed in Table 1 represent ground level reactor building vent releases derived using the same previously approved RG 1.3 methodology, the NRC staff has concluded that the EAB and LPZ χ/Q values presented in Table 1 are acceptable for use in the offsite dose assessment submitted in support of this amendment.

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6 Each SGT system fan has a flow rate of 1780 cfm and the reactor building exhaust fan flow rates are higher. If the CREF system is not in operation, the licensee has committed to having either a reactor building fan or SGT system fan in operation prior to conducting fuel handling operations when less than a 7-day decay time has elapsed.

3.3 Technical Specification Changes

The licensee proposed revising the TS for selected systems to relax operability requirements during the movement of irradiated fuel that has decayed for at least 24 hours or for at least 7 days.[7]

a. <u>24-hour Decayed Fuel</u>

The licensee proposed changes to the applicability and action statements for the following TSs:

- TS 3.6.4.1, Secondary Containment
- TS 3.6.4.2, Secondary Containment Isolation Valves
- TS 3.6.4.3, Standby Gas Treatment System

For these TSs, the licensee proposes deleting "During CORE ALTERATIONS" from the applicability and action statements. In addition, the licensee is proposing adding the term "recently" in front of "irradiated" in the phrase "During movement of irradiated fuel assemblies in the secondary containment" in the applicability and action statements.

The licensee is also proposing modifying the applicability statement as described above for the following TS:

- TS 3.3.6.2, Secondary Containment Isolation Instrumentation

b. <u>7-day Decayed Fuel</u>

The licensee proposed changes to the applicability and action statements for the following TS:

- TS 3.7.4, CR Emergency Filter System

For this TS, the licensee proposes deleting "During CORE ALTERATIONS" from the applicability and action statements. In addition, the licensee is proposing adding the term "lately" in front of "irradiated" in the phrase "During movement of irradiated fuel assemblies in the secondary containment" in the applicability and action statements.

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7 In its original submittal, dated September 29, 2005, the licensee proposed introducing the term "recently irradiated fuel assemblies" in the applicability and action statements for the TSs related to fueling operations. In some cases (i.e., TSs 3.3.6.2, 3.6.4.1, 3.6.4.2, and 3.6.4.3), the licensee was defining this term in the corresponding TS bases as fuel that has occupied part of a critical reactor core within the previous 24 hours; in other cases (i.e., TSs 3.3.7.1 and 3.7.4), the licensee was defining this term in the corresponding TS bases as fuel that has occupied part of a critical reactor core within the previous 7 days. In Question #4 of its RAI letter dated February 9, 2006, the NRC staff asked the licensee how using the same term (i.e., recently irradiated fuel assemblies) with different definitions within the TS would not cause confusion. In its letter response dated April 7, 2006, the licensee agreed that having different definitions for "recently irradiated fuel assemblies" could be a source of confusion and decided to use the term "lately irradiated fuel assemblies" for the 7-day irradiation period.

The licensee is also proposing modifying the applicability statement as described above for the following TS:

- TS 3.3.7.1, CR Emergency Filter System Instrumentation

The FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release. The limiting condition for operation and required actions will remain applicable during activities that could result in an FHA with fuel damage and radiological release. Therefore, the deletion of CORE ALTERATIONS is acceptable. These changes are consistent with the current FHA analysis and TSTF–51. Also, in accordance with the reviewer's note in TSTF–51 mentioned above, the licensee committed to the containment closure guidelines located in NUMARC 93–01. Therefore, these proposed changes to the above listed TSs are acceptable, subject to the acceptance of the radiological consequences as noted below.

The re-analyses of the FHA following both a 24-hour decay time and a 7-day decay time did not credit action by any system addressed by the above listed TSs. The FHA re-analysis was consistent with the definitions of "recently irradiated fuel" and "lately irradiated fuel" and acceptable dose results were obtained with the subject systems inoperable. As such, the proposed revisions are acceptable from an accident radiological consequence perspective.

3.4 Licensee Commitments

The licensee has made the following commitment in the license amendment request dated September 29, 2005:[8]

a. *Within 30 days of issuance of the license amendment, the licensee commits to implement the guidelines of Section 11.3.6.5 of NUMARC 93–01, Revision 3, using a strategy outlined in the Response to Appendix B, Question 2. (Commitment Number NLS2005075–01).*

The licensee stated that the approach that will be taken to conform to the NUMARC guidance will be to develop secondary containment breach control guidance that will be a prerequisite to moving irradiated fuel when secondary containment, the SCIVs, the SGT system, or secondary containment isolation instrumentation are not operable. This breach control guidance will include:

- Ensuring that secondary containment access doors and hatches are closed (except for opening and closing for normal personnel travel). Fouling of an access door/hatch is not allowed unless an individual is on station to restore the closed configuration if an FHA occurs.

- Ensuring that there is normally one available train of SGT, such that if an FHA occurs, the SGT train can be started manually. If both SGT trains would be out of service (such as for work on a component in the common suction line),

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8 The licensee made a second commitment in its September 29, 2005, license amendment request that was subsequently withdrawn in its letter dated April 7, 2006.

pre-positioned means would be on hand with personnel stationed to ensure system integrity could be restored and one train placed in operation.

– Ensuring there is at least the capability of closing one SCIV manually for each secondary containment penetration.

– Ensuring there are no breaches of piping systems that penetrate both sides of secondary containment unless there are pre-positioned means to close the breach if an FHA occurs.

– Ensuring there is no maintenance in progress that compromises the structural integrity of secondary containment (e.g., removal of blowout panels, or core drills).

The licensee stated that the strategy outlined above is designed to ensure that the secondary containment function can be restored without the allowance of partial breaches. Therefore, no SGT system tests above the applicable TS surveillance testing requirements are needed.

The licensee has made the following commitments in its letter dated April 7, 2006:

a. Concurrent with 30-day implementation of issued license amendment, the reactor building roof hatch will be maintained closed during the movement of irradiated fuel in secondary containment. (Commitment Number NLS2006017-01).

In order to ensure the reactor building vent remains the bounding release point as modeled in the CR dose assessment analysis, the reactor building roof hatch will be maintained closed during the movement of irradiated fuel in secondary containment.

b. Concurrent with 30-day implementation of issued license amendment, the TS 3.3.7.1 bases will be revised as provided in Enclosure 2 to formalize this support feature [airflow in the reactor building exhaust plenum]. (Commitment Number NLS2006017-02).

The licensee has assessed the TS setpoint for the radiation monitor and has determined that with a reactor building exhaust fan or SGT system fan initially running, the design-basis AST FHA will cause reactor building exhaust plenum high radiation activation. The TS 3.3.7.1 basis is being revised to formalize that adequate ventilation exhaust airflow is a support function to the reactor building ventilation exhaust plenum high radiation monitor during the movement of lately irradiated fuel. In this manner, ventilation exhaust airflow will be a TS operability requirement for the radiation monitor. In the case of the election to operate with the CREF system manually initiated (in lieu of a running reactor building exhaust fan or SGT fan), TS 3.3.7.1 would prevent the start of fuel movement until Required Action C.1 had been completed ("Initiate CREF System"). Station procedures will ensure these TS-driven prerequisites are met prior to moving lately irradiated fuel.

The NRC staff has reviewed these licensee commitments and has concluded that these commitments meet the Section 11.3.6.5 of NUMARC 93–01, Revision 3, guidelines that (1) ventilation system and radiation monitor availability be maintained and (2) a single normal or contingency method to promptly close primary or secondary containment penetrations be developed.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (71 FR 149, January 3, 2006). 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 the amendment.

6.0 CONCLUSION

The NRC staff has reviewed the AST implementation proposed by the licensee for CNS. The NRC staff also reviewed the proposed changes to the TSs associated with this license amendment request. In performing this review, the NRC staff relied upon information placed on the docket by the licensee, staff experience in doing similar reviews, and, where deemed necessary, staff confirmatory calculations.

This licensing action is considered a selective implementation of the AST. With the approval of this amendment, the AST, the TEDE criteria, and the analysis methods, assumptions, and inputs become the licensing basis for the assessment of radiological consequences of the FHA DBA. This approval is limited to this specific implementation. Subsequent modifications based on the selected characteristics incorporated into the licensing basis by this action may be possible under the provisions of 10 CFR 50.59. However, use of other characteristics of an AST or use of TEDE criteria that are not part of the approved licensing basis, and changes to previously approved AST characteristics, requires prior staff approval under 10 CFR 50.67. The selected characteristics of the AST and the TEDE criteria may not be extended to other aspects of the plant design or operation without prior NRC review under 10 CFR 50.67. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address the selected characteristics of the AST and the TEDE criteria as described in the CNS licensing basis.

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The NRC staff finds that the licensee used analysis, methods, and assumptions consistent with the guidance of RG 1.183, with the exceptions discussed and accepted earlier in this SE. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee’s estimates of TEDE due to a design-basis FHA will comply with the requirements of 10 CFR 50.67 and are in accordance with the guidance of RG 1.183.

The NRC staff finds reasonable assurance that CNS will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. The NRC staff concludes that the proposed AST implementation and the associated TS changes are acceptable from the standpoint of radiological consequences.

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: R. Brad Harvey

Date: September 6, 2006

TABLE 1
FHA Analysis Assumptions

Reactor power	2429 MWt
Radial peaking factor	2.0
Fission product decay period	24 hours and 7 days
Number of fuel rods damaged	151
Number of rods in core (equivalent full length)	47,857
Fuel Gap fission product inventory	
I–131	8%
Kr–85	10%
Other iodines and noble gases	5%
Alkali metals	0.12%
Iodine species fractions	
Elemental	0.9985
Organic	0.0015
Particulates	none
Reactor cavity water depth	23 feet
Pool iodine effective decontamination factor	200
Chemical form of iodine above pool[9]	
Elemental	57%
Organic	43%
Release modeling	
Immediate release from fuel through reactor cavity pool to building	
100% release from building in 2 hours	
No credit for building holdup or filtration prior to release	
CR envelope volume	141,860 ft^3
Normal ventilation unfiltered fresh intake	3,235 cfm
CREF system filtered intake	810 cfm
CREF system start delay time, minutes	1 minute
Unfiltered outside inleakage	400 cfm
CREF system filter efficiency	89%

9 The licensee's input to the RADTRAD computer model specified the iodine fractions as 0.95 aerosol, 0.0485 elemental, and 0.0015 organic. This discrepancy in iodine species fractions had no impact on the RADTRAD results since the filter efficiencies for all three iodine species were set to the same value (89 percent).

TABLE 1 (continued)

CR occupancy factors	
0–24 hr	1.0
24–96 hr	0.6
96–720 hr	0.4
CR breathing rate	3.5×10^{-4} m^3/sec
Offsite breathing rate	
0–8 hr	3.5×10^{-4} m^3/sec
8–24 hr	1.8×10^{-4} m^3/sec
24–720 hr	2.3×10^{-4} m^3/sec

Atmospheric dispersion factors, sec/m^3
Ground level release from reactor building vent

Period	EAB	LPZ	CR
0–2 hr	5.2×10^{-4}	2.9×10^{-4}	4.15×10^{-3}
2–8 hr		2.9×10^{-4}	3.24×10^{-3}
8–24 hr		7.3×10^{-5}	1.32×10^{-3}
24–96 hr		2.5×10^{-5}	9.01×10^{-4}
96–720 hr		5.2×10^{-6}	7.22×10^{-4}

TABLE 2
Calculated FHA Radiological Consequences

	EAB	LPZ	CR
Calculated results, TEDE			
24-hr decay period	1.459	0.815	4.507
7 day decay period	0.627	0.350	4.446
Dose acceptance criteria, TEDE	6.3	6.3	5

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