Nebraska Public Power District

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NLS2005075
September 29, 2005

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

Subject: 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

References:

1. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.
2. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

The purpose of this letter is for the Nebraska Public Power District (NPPD) to request Nuclear Regulatory Commission (NRC) approval for adopting the Alternative Source Term (AST), in accordance with 10 CFR 50.67, for use in calculating the Fuel Handling Accident (FHA) dose consequences. This letter also requests approval of associated changes to the Cooper Nuclear Station (CNS) Technical Specifications (TS).

NPPD has completed a new design basis calculation (Enclosure 1) using the guidelines detailed in References 1 and 2. This calculation demonstrates that radiological dose consequences of an FHA using the AST are within regulatory limits. The associated TS changes would eliminate operability requirements for Secondary Containment, Secondary Containment Isolation Valves, the Standby Gas Treatment System, and Secondary Containment Isolation Instrumentation when handling irradiated fuel that has decayed for 24 hours since critical reactor operations, and when performing Core Alterations. Similar TS relaxations are proposed for the Control Room Emergency Filter System and its initiation instrumentation after a decay period of 7 days.

The proposed TS changes are based on TSTF-51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations" for equipment whose performance is no longer credited in the FHA. Submittal of this License Amendment Request for the FHA is a selective scope application of the AST, as provided for in Reference 2.

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Attachment 1 provides a description of the License Amendment Request, the basis for the amendment, the No Significant Hazards Consideration evaluation pursuant to 10 CFR 50.91(a)(1), and the environmental impact evaluation pursuant to 10 CFR 51.22. Additionally, Appendix A to Attachment 1 describes the conformance of this License Amendment Request to Reference 2. Appendix B to Attachment 1 provides the responses to applicable issues that have been typically raised by the NRC in reviewing the FHA AST applications of other licensees. Attachment 2 provides the proposed changes to the current CNS TS on marked up pages. Attachment 3 provides the revised TS pages in final typed format. Attachment 4 provides the corresponding changes to the current Bases on marked up pages for your information. NRC approval is requested by September 1, 2006, with a 30-day implementation period, to support Refueling Outage 23, which is scheduled to begin in October 2006.

These proposed TS changes have been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). Amendments to the CNS Facility Operating License through Amendment 211 issued March 22, 2005, have been incorporated into this request. NPPD has concluded that the proposed changes do not involve a significant hazards consideration and that they satisfy the categorical exclusion criteria of 10 CFR 51.22(c)(9). This request is submitted under oath pursuant to 10 CFR 50.30(b).

By copy of this letter and its attachments, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies to the NRC Region IV office and the CNS Resident Inspector are also being provided in accordance with 10 CFR 50.4(b)(1).

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

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

Executed on 9/29/05
(Date)

Sincerely,

Randall K. Edington

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

/wv

Attachments
Enclosure

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

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

Senior Resident Inspector w/attachments, enclosure
USNRC - CNS

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

NPG Distribution w/o attachments, enclosure

CNS Records w/attachments, enclosure

ATTACHMENT 1

License Amendment Request for Application of the Alternative Source Term for Reevaluation of the Fuel Handling Accident Dose Consequences

Cooper Nuclear Station, Docket 50-298, DPR-46

Revised TS Pages

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

1.0 Description

2.0 Proposed Change

2.1 Licensing Basis Change for Fuel Handling Accident
2.2 Technical Specification Changes

3.0 Background

4.0 Technical Analysis

4.1 Radiological Consequences of the Fuel Handling Accident
4.2 Atmospheric Dispersion (X/Q)

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration
5.2 Applicable Regulatory Requirements/Criteria

6.0 Environmental Consideration

7.0 References

7.1 General References
7.2 Precedent

Appendix A Regulatory Guide 1.183 Comparison

Appendix B Responses to Commonly Raised NRC Issues On FHA AST License Amendment Requests

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-46 for Cooper Nuclear Station (CNS).

The proposed change revises the CNS design basis accident (DBA) radiological assessment calculational methodology for the Fuel Handling Accident (FHA) through application of the Alternative Source Term (AST), in accordance with the provisions of 10 CFR 50.67, "Accident Source Term." This application represents a selective scope application of AST, as provided for in Regulatory Guide 1.183 (Reference 7.1-1). The Nebraska Public Power District (NPPD) requests Nuclear Regulatory Commission (NRC) review and approval of the AST FHA methodology for application to CNS. Approval of this License Amendment Request will supersede the previous design basis source term assumptions and radiological criteria for the FHA. Future revisions of this analysis, if any, will use the updated approved assumptions and criteria. In accordance with the AST FHA analysis results, revisions to the CNS Technical Specifications (TS) and TS Bases are proposed that reflect the revised safety analysis assumptions for a postulated FHA in the Reactor Building.

The CNS TS currently impose restrictions on plant operations when handling irradiated fuel assemblies or performing Core Alterations. These restrictions require that certain structures, systems or components be operable, and thereby assure that the radiological consequences of an FHA do not exceed those calculated in design basis analyses. The TS changes proposed in this application are based on NRC-approved TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations" (Reference 7.1-2). The proposed TS and TS Bases changes are consistent with this guidance with regard to the relaxation of Secondary Containment, Secondary Containment Isolation Valve (SCIV), Standby Gas Treatment (SGT) System, Secondary Containment Isolation Instrumentation, Control Room Emergency Filter System (CREFS), and CREFS Instrumentation operability for the movement of irradiated fuel after a sufficiently long decay time has passed, and during Core Alterations. A 24 hour decay period and a 7 day decay period are analyzed for the timing of these relaxations. NPPD is not requesting the TSTF-51 relaxations for AC/DC Power Sources - Shutdown and AC Distribution Systems – Shutdown in this License Amendment Request, which would otherwise be applicable to CNS.

The implementation of these changes will reduce the duration and cost of planned outages while maintaining an adequate safety margin. For example, windows of Secondary Containment inoperability of established time periods are currently built into outage schedules. During these windows, critical path fuel movements cannot be performed, which can result in extended outage durations.

NRC approval is requested by September 1, 2006, with a 30-day implementation period. This is to support utilization of this License Amendment Request for Refueling Outage 23, which is scheduled to begin in October 2006.

2.0 PROPOSED CHANGE

NPPD is proposing to modify (a) the licensing basis of the FHA as described in the CNS Updated Safety Analysis Report (USAR) and (b) Technical Specifications for Secondary Containment (TS 3.6.4.1), Secondary Containment Isolation Valves (TS 3.6.4.2), Standby Gas Treatment System (TS 3.6.4.3), Secondary Containment Isolation Instrumentation (TS 3.3.6.2), Control Room Emergency Filter System (TS 3.7.4), and Control Room Emergency Filter System Instrumentation (TS 3.3.7.1).

2.1 Licensing Basis Change for Fuel Handling Accident

NPPD is revising the licensing basis of the FHA described in Section XIV-6.4 of the CNS USAR. The proposed licensing basis change is the reevaluation of the FHA using the AST methodology and dose consequences analysis in accordance with 10 CFR 50.67, Regulatory Guide (RG) 1.183, and NUREG-1465 (Reference 7.1-3). Sections 3.0 and 4.0 of this evaluation provide the background and technical analysis in support of the licensing basis change. Approval of this License Amendment Request will result in the necessary revisions to the USAR, with revised USAR pages submitted pursuant to 10 CFR 50.71(e).

2.2 Technical Specification Changes

An AST analysis has been performed for the radiological dose consequences of a postulated FHA involving irradiated fuel assemblies for two cases: 1) after a 24 hour decay time since reactor shutdown, and 2) after a 7 day decay time. The 24 hour decay time case shows that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary (EAB), and low population zone (LPZ) are below the allowable TEDE limits established in 10 CFR 50.67 without reliance on the safety functions of Secondary Containment, the SGT System, SCIVs, or Secondary Containment isolation instrumentation. The 7 day decay time case shows that in addition to the above Engineered Safety Features, there is sufficient radioactive decay that CREFS and its initiation instrumentation need not be credited in order for the FHA dose consequences to remain within these regulatory limits. These conclusions form the basis for the proposed TS changes that allow, after 24 hours since reactor shutdown, movement of irradiated fuel assemblies without the operability requirements for Secondary Containment, the SGT System, SCIVs, or Secondary Containment Isolation Instrumentation. TS changes are similarly proposed that relax CREFS and CREFS Instrumentation operability requirements after 7 days have elapsed since reactor shutdown. Due to the inherent time required to disassemble the reactor vessel and prepare to move the fuel, it is not credible that fuel movement would commence prior to the 24-hr decay time elapsing. Thus, there is no need for a dose consequence analysis for an FHA occurring prior to that time. The Technical Requirements Manual (TRM), which is incorporated by reference into the USAR, will continue to provide the administrative controls to preclude handling irradiated fuel prior to the analyzed decay period. (Refer also to the response to Question 4 of Appendix B for additional supporting information).

The operability requirements for Secondary Containment, the SGT System, SCIVs, Secondary Containment Isolation Instrumentation, CREFS, and CREFS Instrumentation are also being deleted during Core Alterations as part of this License Amendment Request. The USAR Chapter XIV events postulated to occur during Core Alterations, in addition to the postulated FHA, are: a) Control Rod Removal Error During Refueling, b) Fuel Assembly Insertion Error During Refueling, and c) Fuel Assembly Loading Error (due to either a mislocated bundle or misoriented bundle). Except for the FHA, these other events are not analyzed as resulting in fuel cladding integrity damage. Since the FHA is the only event postulated to occur during Core Alterations that presents a challenge to a fission product barrier, the proposed TS changes omitting Core Alterations is justified per the criteria of 10 CFR 50.36(c)(2).

The Reviewer's Note in TSTF-51 requires that licensees adding the term "recently" make a commitment 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)/Secondary (BWR)." The commitment in the Reviewer's Note reads:

> 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 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 has considered the final version and the draft version of NUMARC 93-01, Revision 3, to be functionally equivalent (Reference 7.2-2). NPPD commits to implement the guidelines of Section 11.3.6.5 of NUMARC 93-01, Revision 3, prior to the implementation of this license amendment. (Refer to the response to Question 2 in Appendix B for further information on the NPPD approach to conformance with this commitment).

Approval of this License Amendment Request will result in the necessary revisions to the associated TS Bases in accordance with TS 5.5.10 (Technical Specifications Bases Control Program). These TS Bases changes are provided for information in Attachment 4 of this submittal.

3.0 BACKGROUND

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to voluntarily replace the traditional accident source term used in their DBA analyses. Regulatory guidance for the implementation of the AST is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." 10 CFR 50.67 requires a licensee seeking to use an alternative source term to apply for a license amendment and requires that the application contain an evaluation of the consequences of DBAs. This License Amendment Request addresses these requirements in proposing selectively to use an AST in evaluating the offsite and control room radiological consequences of an FHA. This reanalysis involves several changes in selected analysis assumptions including different atmospheric dispersion values for the control room outside air intake. As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11. This will also replace the whole body (and its equivalent to any part of the body) dose criteria of 10 CFR 50 Appendix A GDC 19.

4.0 TECHNICAL ANALYSIS

4.1 Radiological Consequences of the Fuel Handling Accident

NPPD has completed a calculation following both a 24 hour decay time since reactor shutdown and a 7 day decay time evaluating the potential dose consequences of the FHA. A copy of the calculation is provided in Enclosure 1. This calculation uses the AST guidelines outlined in NUREG-1465 and RG 1.183. This calculation demonstrates that after a 24 hour decay period following reactor shutdown, the radiological doses at the EAB, LPZ, and in the control room are within regulatory limits without crediting the operability of Secondary Containment, Secondary Containment Isolation Valves, the Standby Gas Treatment System, or Secondary Containment Isolation Instrumentation. After a 7 day decay time, CREFS and CREFS Instrumentation are additionally not credited to meet the regulatory dose limits. A loss-of-offsite-power (LOOP) is not assumed to occur during this accident. (Refer to the response to Question 5 of Appendix B for additional information about the LOOP assumption). Enclosure 1, Attachments A through F provide the RADTRAD files used for the 24 hour decay time case and the 7 day decay time case.

In the postulated FHA, a fuel assembly is assumed to be dropped, thereby damaging 151 fuel rods during fuel handling. To support the proposed TS changes, the calculation does not take credit for Secondary Containment isolation or filtration by the SGT System, assuming a 24 hour period has elapsed after reactor shutdown from full power prior to the FHA. The calculation does not take credit for CREFS or its initiation instrumentation after 7 days have elapsed since reactor shutdown prior to the FHA.

The entire gap activity from the damaged fuel is assumed to be released to the Reactor Cavity Pool, and then directly to the outside atmosphere from the Reactor Building refueling floor over a 2-hour period. The calculated activity in the gap of the fuel rods assumes the assembly has been operated at 102 percent of rated thermal power times a maximum radial peaking factor of 2.0. The analysis assumes the RG 1.183 Table 3 non-loss-of-coolant accident gap fractions. In accordance with RG 1.183, the assumed iodine species released from the fuel gap to the water is 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic.

The radionuclides are assumed to be released from the damaged fuel rods, pass through the water in the reactor cavity, and enter the Reactor Building atmosphere instantaneously. As the released gases rise through the overlaying water, halogens are scrubbed by the water column, resulting in an effective halogen decontamination factor of 200. No decontamination of noble gases or organic iodine forms was assumed. The iodine species above the pool were assumed to be 57 percent elemental and 43 percent organic, in accordance with RG 1.183. 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 Pool, but not for the Spent Fuel Pool where the water depth may be a minimum of 21 feet 6 inches per TS requirements. For the case of an FHA occurring in the Spent Fuel Pool, the implied reduction in scrubbing efficiency is offset by the reduced number of fuel rods (48 vs. 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. In this function, more scrubbing occurs at the bottom of the water column than at the top of the water column. As such, a pool level of 21 feet 6 inches results in a conservative reduction of about 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. (Refer to the response to Question 1 of Appendix B for further information on an FHA occurring in the Spent Fuel Pool).

The analysis assumes a release rate based on the release of essentially 100 percent of the radionuclides in the Reactor Building to the environment over a 2-hour period. This release rate is equivalent to a release flow rate of 45,764 cfm.

To calculate the dose in the Control Room for the 24 hour decay time case where CREFS is credited, two air intake flow rates were used, representing the two distinct operating modes of the Control Room Air Conditioning System. The first mode was for normal air intake of 3635 cfm for a 1-minute time period. This flow rate was the sum of the normal fresh air intake flow of 3235 cfm and an assumed unfiltered inleakage of 400 cfm. The unfiltered inleakage assumption is conservative relative to the inleakage values reported in the NPPD response to Generic Letter 2003-01 (Reference 7.1-4). (Refer to the response to Question 3 in Appendix B for additional information). The second mode was for CREFS operation. CREFS initiates on high radiation detected in the Reactor Building Exhaust Plenum. NPPD has assessed the TS setpoint for the radiation monitor, and has determined that with a Reactor Building exhaust fan or SGT fan initially running, the design basis AST FHA will cause the credited Reactor Building Exhaust Plenum high radiation actuation within the 1-minute delay time assumed for CREFS initiation. (Refer also to the response to Question 5 of Appendix B for additional information).

In calculating the dose in the Control Room for the 7 day decay time case where CREFS is not credited, the normal air intake of 3635 cfm (including 400 cfm unfiltered inleakage) was used for the duration of the accident.

The analysis used assumptions and inputs that follow the guidance in RG 1.183. The key assumptions are listed in Table 1 of this attachment. The calculated dose results are given in Table 2. The calculated doses are within the Standard Review Plan 15.0.1 radiological dose acceptance criteria for an FHA. These TEDE criteria are 6.3 rem at the EAB for the worst two hours, 6.3 rem at the LPZ for the duration of the accident and 5 rem in the Control Room for the duration of the accident.

4.2 Atmospheric Dispersion (X/Q)

The Control Room intake X/Q values used in this analysis were developed previously and were submitted for NRC staff review (Reference 7.1-5) as part of the approval of the current FHA of record in License Amendment 187. The release models 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 release via the Reactor Building vent followed by an elevated filtered release for the balance of the 30-day duration, while this analysis assumes a 30-day unfiltered ground level release from the Reactor Building vent. In the Reference 7.1-5 submittal, X/Qs were calculated for the vent release for different values of Reactor Building vent flow rate (51333 cfm, 9500 cfm, and 1780 cfm). The three flow rates modeled a Reactor Building exhaust fan coastdown after a Group 6 Primary Containment Isolation System trip on high radiation in the Reactor Building Exhaust Plenum. For conservatism, NPPD has elected to use X/Q values calculated for the 1780 cfm flow rate. Although the X/Q values for 1780 cfm were not specifically cited in the License Amendment 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 approved meteorological data, and were developed using ARCON96.

The X/Qs used for the EAB and LPZ are ground level releases from the Reactor Building vent calculated using site specific inputs and methodology described in Regulatory Guide 1.3. These X/Q values were previously submitted to the NRC (Reference 7.1-6). These X/Q values were approved for use, in part, in License Amendment 187. Because of the previously described modeling differences between the current FHA and the one proposed in this License Amendment Request, only the EAB and LPZ X/Qs for the 90-second unfiltered ground level release were cited in License Amendment 187.

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 percent
Kr-85	10 percent
Other iodines and noble gases	5 percent
Alkali metals	0.12 percent
Iodine species fractions	
Elemental	0.9985
Organic	0.0015
Particulates	none
Reactor cavity water depth	23 feet
Pool iodine effective decontamination factor	200

Table 1 (continued)

Chemical form of iodine above pool	
Elemental	57 percent
Organic	43 percent

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

Control Room Envelope volume	141,860 ft^3
Normal ventilation unfiltered fresh intake	3,235 cfm
CREFS filtered intake	810 cfm
CREFS start delay time, minutes	1 minute
Unfiltered outside inleakage	400 cfm
CREFS filter efficiency	89%
Control room occupancy factors	
0-24 hr	1.0
24-96 hr	0.6
96-720 hr	0.4
Control room breathing rate	3.5E-4 m^3/sec
Offsite breathing rate	
0-8 hr	3.5E-4 m^3/sec
8-24 hr	1.8E-4 m^3/sec
24-720 hr	2.3E-4 m^3/sec

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

Period	EAB	LPZ	Control Room
0-2 hours	5.2E-4	2.9E-4	4.15E-3
2-8 hours		2.9E-4	3.24E-3
8-24 hours		7.3E-5	1.32E-3
24-96 hours		2.5E-5	9.01E-4
96-720 hours		2.5E-6	7.22E-4

Table 2
Calculated FHA Radiological Consequences

	TEDE (rem) EAB	LPZ	Control Room
Calculated results (24-hr decay period)	1.459	0.815	4.507
(7 day decay period)	0.627	0.350	4.446
Dose acceptance criteria	6.3	6.3	5

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with 10CFR50.92, a proposed change to the operating facility involves no "significant hazards" if operation of the facility, in accordance with the proposed change, would not 1) involve a significant increase in the probability or consequences of any accident previously evaluated, 2) create the possibility of a new or different kind of accident from any accident previously evaluated, or 3) involve a significant reduction in a margin of safety.

The Nebraska Public Power District (NPPD) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed amendment involves implementation of the Alternative Source Term (AST) for the fuel handling accident (FHA) at Cooper Nuclear Station (CNS). There are no physical design modifications to the plant associated with the proposed amendment. The FHA AST calculation does not impact the initiators of an FHA in any way.

   The changes also do not impact the initiators for any other design basis accident (DBA) or events. Therefore, because DBA initiators are not being altered by adoption of the AST analyses the probability of an accident previously evaluated is not affected.

   With respect to consequences, the only previously evaluated accident that could be affected is the FHA. The AST is an input to calculations used to evaluate the consequences of the accident, and does not, in and of itself, affect the plant response or the actual pathways to the environment utilized by the radiation/activity released by the fuel. It does, however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. For the FHA, the AST analyses demonstrate acceptable doses that are within regulatory limits after 24 hours of radioactive decay since reactor shutdown, without credit for Secondary Containment, the Standby Gas Treatment System, Secondary Containment Isolation Valves, or Secondary Containment Isolation Instrumentation, and that the Control Room Emergency Filter System (CREFS) and CREFS Instrumentation need not be credited after a 7 day period of decay. Therefore, the consequences of an accident previously evaluated are not significantly increased.

Based on the above conclusions, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed amendment does not involve a physical alteration of the plant. No new or different types of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes. The proposed changes to the control of Engineered Safety Features during handling of irradiated fuel do not create new initiators or precursors of a new or different kind of accident. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed amendment.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

3. Do the proposed changes involve a significant reduction in the margin of safety?

   Response: No

   The proposed amendment is associated with the implementation of a new licensing basis for the CNS FHA. Approval of this change from the original source term to an AST derived in accordance with the guidance of Regulatory Guide (RG) 1.183 is being requested. The results of the FHA analysis, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The AST FHA analysis has been performed using conservative methodologies, as specified in RG 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analysis adequately bounds the postulated limiting event scenario. The dose consequences of the limiting FHA remain within the acceptance criteria presented in 10 CFR 50.67, the Standard Review Plan, and RG 1.183.

The proposed changes continue to ensure that the doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundary, as well as the Control Room, are within the corresponding regulatory limits. For the FHA, RG 1.183 conservatively sets the EAB and LPZ limits below the 10 CFR 50.67 limit, and sets the Control Room limit consistent with 10 CFR 50.67.

Since the proposed amendment continues to ensure the doses at the EAB, LPZ and Control Room are within corresponding regulatory limits, the proposed license amendment does not involve a significant reduction in a margin of safety.

Based on the above, NPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The construction of CNS predated the 1971 issuance of 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants." CNS is designed to be in conformance with the intent of the Draft General Design Criteria (GDCs), published in the Federal Register on July 11, 1967, except where commitments have been made to specific 1971 GDCs. The applicable GDCs are:

Draft GDC 10 – Containment

"Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public."

This Draft GDC is applicable to Secondary Containment when the Primary Containment is open. This License Amendment Request does not alter CNS commitments to conformance with this Draft GDC, although Secondary Containment will no longer be credited for the mitigation of an FHA.

Draft GDC 69 – Protection Against Radioactivity Release From Spent Fuel and Waste Storage

"Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs."

The consequences of an FHA, as presented in this License Amendment Request demonstrate that undue amounts of radioactivity are not released to the public.

1971 GDC 19 – Control Room

NPPD is committed to the provisions of 1971 GDC 19 as it applies to the radiological exposures to Control Room occupants:

"Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."

With NRC acceptance of this AST License Amendment Request, the 5 rem whole body (or equivalent) criteria for the FHA will be replaced with the 5 rem TEDE limits of 10 CFR 50.67 (as provided for in GDC 19).

1971 GDC 20 – Protection System Functions

NPPD is committed to the provisions of 1971 GDC 20 as it applies to the design of the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System:

"The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety."

The deletion of the TS operability requirement for the Reactor Building Ventilation Exhaust Plenum – High function of TS 3.3.6.2 during Core Alterations and movement of non-recently irradiated fuel does not alter NPPD commitments to GDC 20.

10 CFR 50.67, "Accident Source Term"

10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design basis radiological consequence analyses. NRC approval of this AST License Amendment Request will result in a revision to the evaluation and consequences of a design basis FHA previously reported in the Updated Safety Analysis Report.

10 CFR 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance"

This paragraph provides criteria for evaluating the radiological aspects of reactor sites. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. A similar footnote appears in 10 CFR 50.67.

In accordance with the provisions of 10 CFR 50.67(a), the radiation dose reference values in 10 CFR 50.67(b)(2) were used in these analyses in lieu of those prescribed in 10 CFR 100. (Refer to footnote 5 on page 1.183-7 of Reference 7.1-1).

In conclusion, based on the considerations discussed above, (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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 General References

1. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. TSTF-51, Rev. 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Excel Services Corporation.

3. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," L. Soffer et al., February 1995.

4. Letter from R. Edington (NPPD) to USNRC, dated September 30, 2004, "Initial Actions Summary Report and Response to NRC Generic Letter 2003-01, 'Control Room Habitability'" (NLS2004105).

5. Letter from J. Swailes (NPPD) to USNRC, dated September 14, 2001, "Proposed License Amendment Related to the Design Basis Accident Radiological Assessment Calculational Methodology – Supplemental Information" (NLS2001085).

6. Letter from J. Swailes (NPPD) to USNRC, dated February 28, 2001, "Proposed License Amendment Related to the Design Basis Accident Radiological Assessment Calculational Methodology" (NLS2001011).

7.2 Precedent

There are numerous NRC-approved Boiling Water Reactor (BWR) precedents for use of AST for the FHA and adoption of TSTF-51. The following examples are approved selective scope FHA AST submittals that were used in the development of this License Amendment Request.

1. Letter from G. Vissing (NRC) to M. Kansler (Entergy Nuclear Operations), dated September 12, 2002, "James A. Fitzpatrick Nuclear Power Plant – Amendment Re: Technical Specification Change to the Requirements for Handling Irradiated Fuel (TAC No. MB5328)."

2. Letter from J. Boska (NRC) to M. Kansler (Entergy Nuclear Operations), dated April 28, 2005, "Pilgrim Nuclear Power Station – Issuance of Amendment Re: Alternative Source Term for the Fuel Handling Accident Dose Consequences (TAC No. MC2705).

Appendix A

Regulatory Guide 1.183 Comparison

The Appendix provides a comparison table of the Cooper Nuclear Station FHA AST calculation (Enclosure 1) to the criteria of RG 1.183 that pertain to BWR guidance for AST analysis development and License Amendment Request content.

RG Section	RG Position	CNS Analysis
1.1.1	The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. Changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the initial AST implementation has been approved by the staff and has become part of the facility design basis, the licensee may use 10 CFR 50.59 and its supporting guidance in assessing safety margins related to subsequent facility modifications and changes to procedures.	Conforms- Adequate safety margins are maintained, as discussed in the No Significant Hazards Consideration. Future changes will be evaluated under the provisions of 10 CFR 50.59.
1.1.2	The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of	Conforms- Adoption of the AST for the FHA makes possible the TS relaxations of Secondary Containment, SGT System, SCIV, and Secondary Containment Isolation instrumentation operability after a 24 hour decay period, and CREFS and CREFS Instrumentation after a 7 day

RG Section	RG Position	CNS Analysis
	challenges to the system, and uncertainties. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions.	decay period, as well as during Core Alterations. These TS changes are based on TSTF-51, which was approved for use by the NRC. Compliance with the GDCs are maintained (both the 1967 draft GDCs and the 1971 GDCs to which NPPD has committed). No reliance is placed on compensatory programmatic actions (including manual operator actions) to maintain adequate defense-in-depth.
1.1.2	Proposed modifications that seek to downgrade or remove required engineered safeguards equipment should be evaluated to be sure that the modification does not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program.	Not applicable- Changes to the TS which remove the operability requirements for Secondary Containment, the SGT System, SCIV and Secondary Containment Isolation Instruments during movement of irradiated fuel assemblies 24 hours after shutdown, and CREFS and CREFS instrumentation after a 7 day decay period, have negligible affect on the assumptions made in the PRA or the severe accident management program. The affected systems have no affect on Core Damage Frequency. In addition, Large Early Release Frequency is not affected based on the assumed damage to 0.3% (151/47857) fuel rods in an FHA and the CNS definition of a large release as greater than 10% Cesium Iodine fraction.

RG Section	RG Position	CNS Analysis
1.1.3	The design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. Although a complete re-assessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses would generally not be necessary. Regulatory Position 1.3 of this guide provides guidance on which analyses need updating as part of the AST implementation submittal and which may need updating in the future as additional modifications are performed.	Conforms- See RG Section 1.3 discussions.
1.1.3	This approach would create two tiers of analyses, those based on the previous source term and those based on an AST. The radiological acceptance criteria would also be different with some analyses based on whole body and thyroid criteria and some based on TEDE criteria. Full implementation of the AST revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. Selective implementation of the AST also revises the plant licensing basis and may establish the TEDE dose as the new acceptance criteria. Selective implementation differs from full implementation only in the scope of the change. In either case, the facility design bases should clearly indicate that the source term assumptions and radiological criteria in these affected analyses have been superseded and that future revisions of these analyses, if any, will use the updated approved assumptions and criteria.	Conforms- The CNS FHA AST License Amendment Request is a selective scope AST application. Approval of this License Amendment Request will supersede the previous design basis source term assumptions and radiological criteria for the FHA. Future revisions of this analysis, if any, will use the updated approved assumptions and criteria.
1.1.3	Radiological analyses generally should be based on assumptions and inputs that are consistent with corresponding data used in other design basis safety analyses, radiological and nonradiological, unless these data would result in nonconservative results or otherwise conflict with the guidance in this guide.	Conforms- The FHA analysis is a selective scope application of the AST. It relies on assumptions and inputs that do not create a conflict with, or render non-conservative, other design basis safety analyses.

RG Section	RG Position	CNS Analysis
1.1.4	Although the AST provided in this guide was based on a limited spectrum of severe accidents, the particular characteristics have been tailored specifically for DBA analysis use. The AST is not representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient *by itself* as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.	Not Applicable- No changes are proposed in this License Amendment Request to Emergency Preparedness requirements.
1.2.1	Full implementation is a modification of the facility design basis that addresses all characteristics of the AST, that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in the application (which may only include a subset of the plant analyses), but also to all future design basis analyses. At a minimum for full implementations, the DBA LOCA must be re-analyzed using the guidance in Appendix A of this guide. Additional guidance on analysis is provided in Regulatory Position 1.3 of this guide. Since the AST and TEDE criteria would become part of the facility design basis, new applications of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59, "Changes, Tests, and Experiments," or unless the new application involved a change to a technical specification. However, a change from an approved AST to a different AST that is not approved for use at that facility would require a license amendment under 10 CFR 50.67.	Not Applicable- This License Amendment Request is a selective scope implementation of the AST to the FHA.
1.2.2	Selective implementation is a modification of the facility design basis that (1) is based on one or more of the characteristics of the AST or (2) entails re-evaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees flexibility in technically justified selective implementations provided a clear, logical, and consistent design basis is	Conforms- This License Amendment Request is a selective scope application of the AST to change the design basis radiological dose consequence analysis of the FHA.

RG Section	RG Position	CNS Analysis
	maintained. An example of an application of selective implementation would be one in which a licensee desires to use the release timing insights of the AST to increase the required closure time for a containment isolation valve by a small amount. Another example would be a request to remove the charcoal filter media from the spent fuel building ventilation exhaust. For the latter, the licensee may only need to re-analyze DBAs that credited the iodine removal by the charcoal media. Additional analysis guidance is provided in Regulatory Position 1.3 of this guide. NRC approval for the AST (and the TEDE dose criterion) will be limited to the particular selective implementation proposed by the licensee. The licensee would be able to make subsequent modifications to the facility and changes to procedures based on the selected AST characteristics incorporated into the design basis 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 design basis, and changes to previously approved AST characteristics, would require prior staff approval under 10 CFR 50.67. As an example, a licensee with an implementation involving only timing, such as relaxed closure time on isolation valves, could not use 10 CFR 50.59 as a mechanism to implement a modification involving a reanalysis of the DBA LOCA. However, this licensee could extend use of the timing characteristic to adjust the closure time on isolation valves not included in the original approval.	
1.3.1	There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. These requirements include, but are not limited to, the following. • Environmental Qualification of Equipment (10 CFR 50.49) • Control Room Habitability (GDC 19 of Appendix A to 10 CFR Part 50) • Emergency Response Facility Habitability (Paragraph IV.E.8 of	Conforms- This selective scope AST FHA request is salient to: a) Control Room Habitability (GDC 19 and NUREG-0737 Item III.D.3.4), b) AST (10 CFR 50.67), c) Emergency Response Facility Habitability (10 CFR 50 Appendix E Paragraph IV.E.8), and d) Facility Siting (10 CFR 100.11). Control

RG Section	RG Position	CNS Analysis
	Appendix E to 10 CFR Part 50) • Alternative Source Term (10 CFR 50.67) • Environmental Reports (10 CFR Part 51) • Facility Siting (10 CFR 100.11)[5] There may be additional applications of the accident source term identified in the technical specification bases and in various licensee commitments. These include, but are not limited to, the following from Reference 2, NUREG-0737. • Post-Accident Access Shielding (NUREG-0737, II.B.2) • Post-Accident Sampling Capability (NUREG-0737, II.B.3) • Accident Monitoring Instrumentation (NUREG-0737, II.F.1) • Leakage Control (NUREG-0737, III.D.1.1) • Emergency Response Facilities (NUREG-0737, III.A.1.2) • Control Room Habitability (NUREG-0737, III.D.3.4)	Room Habitability and compliance with the Alternative Source Term requirements are the principal subjects of this submittal and are discussed in Sections 3 and 4 of this License Amendment Request. Regarding Emergency Response Facility Habitability, CNS will continue to meet the NUREG-0654 Planning Standard for Emergency Facilities and Equipment as described in the CNS Emergency Plan. As stated in Footnote 5 of this RG, the dose guidelines of 10 CFR 100.11 are superseded by 10 CFR 50.67 for licensees that have implemented an AST.
1.3.2	Any implementation of an AST, full or selective, and any associated facility modification should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the re-evaluation will necessarily be a function of the specific proposed facility modification[6] and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is	Conforms- The License Amendment Request for this selective scope application of the AST to the FHA evaluated the impact of the proposed change against the Current Licensing Basis, mitigating system design basis requirements, and Technical Specifications. No facility modifications are proposed as part of this License Amendment Request and compliance with regulations and commitments are maintained.

RG Section	RG Position	CNS Analysis
	considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid. Generic analyses, such as those performed by owner groups or vendor topical reports, may be used provided the licensee justifies the applicability of the generic conclusions to the specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. The license amendment request should describe the licensee's re-analysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.	
1.3.2	The NRC staff has performed an evaluation of the impact of the AST on three representative operating reactors (Ref. 14). This evaluation determined that radiological analysis results based on the TID-14844 source term assumptions (Ref. 1) and the whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses. However, this does not exempt the licensee from evaluating the remaining radiological and nonradiological impacts of the AST implementation and the impacts of the associated plant modifications. For example, a selective implementation based on the timing insights of the AST may change the required isolation time for the containment purge dampers from 2.5 seconds to 5.0 seconds. This application might be acceptable without dose calculations. However, evaluations may need to be performed regarding the ability of the damper to close against increased containment pressure or the	Conforms- There are no plant modifications that are planned to implement the FHA AST analysis. The radiological and nonradiological impacts of the FHA AST have been considered and discussed in the License Amendment Request, as applicable.

RG Section	RG Position	CNS Analysis
	ability of ductwork downstream of the dampers to withstand increased stresses.	
1.3.2	For full implementation, a complete DBA LOCA analysis as described in Appendix A of this guide should be performed, as a minimum. Other design basis analyses are updated in accordance with the guidance in this section.	Not Applicable- This License Amendment Request is a selective scope application of the AST to the FHA.
1.3.2	A selective implementation of an AST and any associated facility modification based on the AST should evaluate all the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. Design basis analyses are updated in accordance with the guidance in this section. There is no minimum requirement that a DBA LOCA analysis be performed. The analyses performed need to address all impacts of the proposed modification, the selected characteristics of the AST, and if dose calculations are performed, the TEDE criteria. For selective implementations based on the timing characteristic of the AST, e.g., change in the closure timing of a containment isolation valve, re-analysis of radiological calculations may not be necessary if the modified elapsed time remains a fraction (e.g., 25%) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual basis. For longer time delays, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed.	Conforms- The radiological and nonradiological impacts of the FHA AST have been considered as they apply to the implementation.
1.3.3	It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a *sensitivity analysis* is an evaluation that considers how the overall results vary as an input parameter (in this case, AST characteristics) is varied. A *scoping analysis* is a brief evaluation that uses	Not Applicable- The FHA AST analysis does not rely on sensitivity or scoping analyses.

RG Section	RG Position	CNS Analysis
	conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but are otherwise largely based on generic assumptions and inputs. Such cases might include postaccident vital area access dose calculations, shielding calculations, and equipment environmental qualification (integrated dose). It may be possible to identify a bounding case, re-analyze that case, and use the results to draw conclusions regarding the remainder of the analyses. It may also be possible to show that for some analyses the whole body and thyroid doses determined with the previous source term would bound the TEDE obtained using the AST. Where present, arbitrary "designer margins" may be adequate to bound any impact of the AST and TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low population zone (LPZ), or control room dose.	
1.3.4	Full implementation of the AST replaces the previous accident source term with the approved AST and the TEDE criteria for all design basis radiological analyses. The implementation may have been supported in part by sensitivity or scoping analyses that concluded many of the design basis radiological analyses would remain bounding for the AST and the TEDE criteria and would not require updating. After the implementation is complete, there may be a subsequent need (e.g., a planned facility modification) to revise these analyses or to perform new analyses. For these recalculations, the NRC staff expects that all characteristics of the AST and the TEDE criteria incorporated into the design basis will be addressed in all affected analyses on an individual as-needed basis. Re-evaluation using the	Not Applicable- This License Amendment Request is a selective scope implementation of the AST to the CNS FHA.

RG Section	RG Position	CNS Analysis
	previously approved source term may not be appropriate. Since the AST and the TEDE criteria are part of the approved design basis for the facility, use of the AST and TEDE criteria in new applications at the facility do not constitute a change in analysis methodology that would require NRC approval.[7]	
1.3.4	This guidance is also applicable to selective implementations to the extent that the affected analyses are within the scope of the approved implementation as described in the facility design basis. In these cases, the characteristics of the AST and TEDE criteria identified in the facility design basis need to be considered in updating the analyses. Use of other characteristics of the AST or TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, requires prior NRC staff approval under 10 CFR 50.67.	Conforms- Subsequent updates to the FHA AST analysis after NRC approval will consider the characteristics of the AST and TEDE criteria in the facility design basis.
1.3.5	Current environmental qualification (EQ) analyses may be impacted by a proposed plant modification associated with the AST implementation. The EQ analyses that have assumptions or inputs affected by the plant modification should be updated to address these impacts. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue. The EQ dose estimates should be calculated using the design basis survivability period.	Not Applicable- The FHA is not a basis for the CNS Environmental Qualification program.
1.4	The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. Use of an AST alone cannot increase the core damage frequency (CDF) or the large early release	Not applicable. No facility modifications are proposed or planned as implementation actions of the FHA AST analysis.

RG Section	RG Position	CNS Analysis
	frequency (LERF). However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the impact on the existing PRAs should be evaluated.	
1.4	Consideration should be given to the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is a reason to question adequate protection of public health and safety.	Conforms- The TS and TS Bases changes which remove the operability requirements for Secondary Containment, Standby Gas Treatment, Secondary Containment Isolation Valves and Secondary Containment Isolation Instruments during movement of irradiated fuel assemblies 24 hours after shutdown have a negligible effect on risk due to Core Damage or Large Early Releases. The affected systems have no affect on Core Damage Frequency. In addition, Large Early Release Frequency is not affected based on the assumed damage to 0.3% (151/47857) fuel rods in an FHA and the CNS definition of a large release as greater than 10% Cesium Iodine fraction. Approval of this License Amendment Request will result in the necessary changes to shutdown risk modeling to reflect the proposed changes to the TS, as required.
1.4	The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff	Not Applicable- NPPD is not utilizing risk insights as a basis for this License

RG Section	RG Position	CNS Analysis
	positions. For guidance, refer to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 15).	Amendment Request.
1.5	According to 10 CFR 50.90, an application for an amendment must fully describe the changes desired and should follow, as far as applicable, the form prescribed for original applications. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (Ref 16), provides additional guidance. The NRC staff's finding that the amendment may be approved must be based on the licensee's analyses, since it is these analyses that will become part of the design basis of the facility. The amendment request should describe the licensee's analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation.	Conforms- The License Amendment Request is formatted in accordance with accepted NRC/industry guidance. The request describes the radiological and nonradiological impacts of the FHA AST analysis. Consistent with several of the most recently approved BWR precedents (Pilgrim, Vermont Yankee, and River Bend), affected USAR pages are not included in the analyses. Approval of this License Amendment Request will result in the necessary revisions to the USAR, with revised USAR pages submitted pursuant to 10 CFR 50.71(e).
1.5	If the licensee has used a current approved version of an NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used and submits the inputs that the licensee used in the calculations made with that code. In many cases, this will reduce the need for NRC staff confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.	Conforms- NPPD has used RADTRAD Version 3.03 in the performance of the FHA AST analysis. The inputs made are included with Enclosure 1. ARCON96 was used to develop the Control Room X/Q values. That calculation was submitted to the NRC as part of the License Amendment 187 review.
1.6	Requirements for updating the facility's final safety analysis report (FSAR) are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all	Conforms- Approval of this License Amendment Request will result in the necessary revisions to the USAR, with revised USAR pages submitted pursuant

RG Section	RG Position	CNS Analysis
	safety evaluations performed by the licensee in support of requests for license amendments or in support of conclusions that changes did not involve unreviewed safety questions. The analyses required by 10 CFR 50.67 are subject to this requirement. The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis source term by the AST. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. Regulatory Guide 1.70 (Ref. 16) provides additional guidance. The descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.	to 10 CFR 50.71(e).
2.1	The AST must be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events, that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.	Conforms- This License Amendment Request applies the AST to the FHA.
2.2	The AST must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.	Conforms- See Enclosure 1, Section 2.2.
2.3	The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.	Conforms- This License Amendment Request considers a number of scenarios involving an FHA during fuel movements or Core Alterations. The most limiting of these events is analyzed for radiological consequences.
2.4	The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be	Conforms- Enclosure 1 has been developed based on NUREG-1465 and

RG Section	RG Position	CNS Analysis
	documented in a scrutable form that facilitates public review and discourse.	this Regulatory Guide. The calculation, which utilizes RADTRAD Version 3.03 was developed in accordance with 10 CFR 50 Appendix B, Criterion III.
2.5	The AST must be peer-reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution should be part of the documentation supporting the AST.	Conforms- Enclosure 1 has been developed by industry experts and reviewed and accepted by CNS Engineering. The calculation was developed in accordance with 10 CFR 50 Appendix B program, Criterion III.
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[8] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9] The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms- See Enclosure 1 Section 2.2.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits	Conforms- See Enclosure 1 Section 2.1.

RG Section	RG Position	CNS Analysis
	report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms- See Enclosure 1 Section 2.2.
3.2	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.	Not Applicable- This License Amendment Request is a selective scope implementation of the AST to the CNS FHA.
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3.[11] Non-LOCA Fraction of Fission Product Inventory in Gap** Table 3 Group — Fraction I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12	Conforms- See Enclosure 1 Section 2.2.
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase	Conforms- See Enclosure 1 Section 4.1.

RG Section	RG Position	CNS Analysis
	immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[12] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.	
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable- This License Amendment Request is a selective scope implementation of the AST to the CNS FHA.
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. Table 5 Radionuclide Groups <u>Group</u> — <u>Elements</u> Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthenides — La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms- These radionuclide groups were considered in the FHA AST analysis, as described in Enclosure 1 Section 4.3.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine,	Conforms- See Enclosure 1 Section 2.2.

RG Section	RG Position	CNS Analysis
	and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases. The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.	Conforms- See Enclosure 1 Section 2.2.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]	Conforms- See Enclosure 1 Section 2.6.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal	Conforms- See Enclosure 1 Section 2.6.

RG Section	RG Position	CNS Analysis
	Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms- See Enclosure 1 Section 2.6.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms- See Enclosure 1 Section 2.6.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the	Conforms- See Enclosure 1 Section 2.5.

RG Section	RG Position	CNS Analysis
	progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms- See Enclosure 1 Section 2.6.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms- See Enclosure 1 Section 3.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms- See Enclosure 1 Section 5.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms- See Enclosure 1 Section 5.
4.2.3	The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose	Conforms- See Enclosure 1 Section 5.

RG Section	RG Position	CNS Analysis
	rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	Conforms- Under the design basis source term release and dispersion criteria of Appendix B to this Regulatory Guide, the Reactor Building Exhaust Plenum radiation monitor will detect high radiation in the plenum and start CREFS within the assumed 1-minute initiation time.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms- No credit is taken for the use of personal protective equipment or prophylactic drugs.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms- See Enclosure 1 Section 2.6.

RG Section	RG Position	CNS Analysis
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$ Equation 1	Conforms- See Enclosure 1, Section 2.6.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Not Applicable- This is a selective scope application of the AST to the FHA. The NUREG-0737 analyses are based on non-AST LOCA results. Therefore a conversion to AST methodology is not appropriate.
4.4	The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.	Conforms- See Enclosure 1, Section 5.
4.4	The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A	Not Applicable- This is a selective scope application of the AST to the FHA. The

RG Section	RG Position	CNS Analysis
	to 10 CFR Part 50 or specify criteria derived from GDC 19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).	NUREG-0737 analyses are based on non-TEDE LOCA results. Therefore a change from GDC 19 to TEDE methodology is not appropriate.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms- Enclosure 1 was prepared and accepted by NPPD under a 10 CFR 50 Appendix B Quality Assurance program.
5.1.1	These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences.	Not Applicable- NPPD is not proposing deviations to conformance with this Regulatory Guide.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that	Conforms- The accident mitigation features that are credited in the FHA AST analysis are: 1) 23 foot water level over fuel in the reactor (TS-controlled), and 2) CREFS and CREFS initiation

RG Section	RG Position	CNS Analysis
	results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	instrumentation (TS-controlled). A loss-of-offsite power is not assumed for accidents occurring during Modes 4 or 5 per the CNS TS Bases for AC Power Sources.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.[18] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value.	Conforms- The numeric values chosen provide a conservative dose result. There are no parameters in the FHA AST analysis that have conflicting credit within the analysis. Parameters that are controlled by TS either use the TS values, or are otherwise bounded by them (e.g. CREFS filter efficiency relaxation). The parameters are not based on surveillance testing results.
5.1.4	The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding	Conforms- The FHA analysis assumptions and methods are compatible with the AST and the TEDE criteria.

RG Section	RG Position	CNS Analysis
	of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	
5.2	The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST.	Conforms- See RG 1.183 Appendix B Sections of this appendix.
5.2	The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with	Conforms- See RG 1.183 Appendix B Sections of this appendix.

RG Section	RG Position	CNS Analysis
	the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency.	
5.2	The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.	Conforms- PRA was not used as a basis for acceptability of this FHA AST License Amendment Request.
5.3	Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28).	Conforms- X/Qs that are used were previously submitted for NRC review (see License Amendment Request Section 4.2).
5.3	References 22 and 28 should be used if the FSAR X/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use	Not Applicable- The X/Q values were previously submitted to the NRC and are based on approved meteorological data (see License Amendment Request Section 4.2).

RG Section	RG Position	CNS Analysis
	is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[19] (Ref. 26) is generally acceptable to the NRC staff for use in determining control room X/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident X/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in X/Q analysis methodology should be reviewed by the NRC staff.	
6.0	The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.	Not Applicable- 10 CFR 50.49 compliance is not affected by the Fuel Handling Accident.
B-1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms- See discussions in RG Section 3.

RG Section	RG Position	CNS Analysis
B-1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms- The FHA is based on GE14 fuel which assumes 151 fuel rods are damaged per Amendment 22 of NEDE-24011-P-A (GESTAR II).
B-1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms- See Enclosure 1 Section 2.2.
B-1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms- See Enclosure 1 Section 2.2.
B-2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms- See Enclosure 1 Section 2.2. Section 4.1 of this License Amendment Request discusses the case of an FHA occurring in the Spent Fuel Pool.

RG Section	RG Position	CNS Analysis
B-3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms- See Enclosure 1 Section 4.1.
B-4	For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff.	Not Applicable- CNS does not have a separate fuel building.
B-5.1	If the containment is isolated[2] during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable- NPPD is not proposing an isolated Secondary Containment during fuel handling operations.
B-5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[1] no radiological consequences need to be analyzed.	Not Applicable- NPPD is not taking any credit for Secondary Containment isolation after an FHA.
B-5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms- Radioactive material that escapes the Reactor Cavity Pool is released to the environment over a 2-hour period.
B-5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]	Not Applicable- The SGT System is not credited in the release from the Secondary Containment to the environment.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the	Not Applicable- No credit is taken for mixing or dilution.

RG Section	RG Position	CNS Analysis
	containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	
Footnote 6	For example, a proposed modification to change the timing of a containment isolation valve from 2.5 seconds to 5.0 seconds might be acceptable without any dose calculations. However, a proposed modification that would delay containment spray actuation could involve recalculation of DBA LOCA doses, re-assessment of the containment pressure and temperature transient, recalculation of sump pH, re-assessment of the emergency diesel generator loading sequence, integrated doses to equipment in the containment, and more.	Conforms- No facility modifications are proposed with this License Amendment Request. Future modifications will consider the effects upon the inputs and assumptions of the FHA AST.
Footnote 7	In performing screenings and evaluations pursuant to 10 CFR 50.59, it may be necessary to compare dose results expressed in terms of whole body and thyroid with new results expressed in terms of TEDE. In these cases, the previous thyroid dose should be multiplied by 0.03 and the product added to the whole body dose. The result is then compared to the TEDE result in the screenings and evaluations. This change in dose methodology is not considered a change in the method of evaluation if the licensee was previously authorized to use an AST and the TEDE criteria under 10 CFR 50.67.	Conforms- Approval of this License Amendment Request will result in the necessary implementation within the CNS 10 CFR 50.59 program.
Footnote 8	The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02.	Conforms- See Enclosure 1, Section 2.2.
Footnote 9	Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used.	Not applicable- Particulates are not applicable to the FHA, as they are 100% retained in the Reactor Cavity Pool.

RG Section	RG Position	CNS Analysis
Footnote 10	The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.	Conforms- The design of the current core will remain within the Footnote 10 constraints. NPPD is monitoring Industry/NRC progress in eliminating this footnote for future core design.
Footnote 11	The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.	Conforms- Using the CNS modeling code, NPPD has determined that the current core will remain within the constraints of Footnote 11. However, the fuel vendor has been contacted to provide confirmation with the licensing code/methodology. The results are expected first quarter 2006, and supplemental information will be provided to the NRC, as necessary. NPPD is monitoring Industry/NRC progress in eliminating this footnote for future core design.
Footnote 12	In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase, i.e., in step increases.	Conforms- See Enclosure 1, Section 2.3.
Footnote 13	The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.	Conforms- RADTRAD Version 3.03 was used which calculates offsite exposure TEDE for the AST.
Footnote 14	With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.	Conforms- Future changes made pursuant to 10 CFR 50.59 will be in the context of the maximum 2-hour dose consequences.

RG Section	RG Position	CNS Analysis
Footnote 15	The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements since it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.	Conforms- FHA AST calculation was developed using RADTRAD Version 3.03.
Footnote 16	This occupancy is modeled in the X/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.	Conforms- The required occupancy assumptions are incorporated into the dose calculation. See Enclosure 1, Section 2.6.
Footnote 18	Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.	Not Applicable- Although a lower CREFS filter efficiency is assumed than is required by the TS Ventilation Filter Testing Program, NPPD is not proposing changes to the TS efficiency.
Footnote 19	The ARCON96 computer code contains processing options that may yield X/Q values that are not sufficiently conservative for use in accident consequence assessments or may be incompatible with release point and ventilation intake configurations at particular sites. The applicability of these options and associated input parameters should be evaluated on a case-by-case basis. The assumptions made in the examples in the ARCON96 documentation are illustrative only and do not imply NRC staff acceptance of the methods or data used in the example.	Conforms- The X/Qs used in the FHA AST analysis were developed using ARCON96 and were previously reviewed by the NRC pursuant to License Amendment 187.
Footnote B-1	These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.	Conforms- With ventilation flow established at the start of the accident, the Reactor Building Exhaust Plenum radiation monitor will provide the

RG Section	RG Position	CNS Analysis
		necessary CREFS initiation. Refer to Section 4.1 and Appendix B Question 5 of this License Amendment Request.
Footnote B-2	Containment *isolation* does not imply containment integrity as defined by technical specifications for non-shutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the appropriate form of isolation should be addressed in technical specifications.	Not Applicable- NPPD is not proposing an isolated Secondary Containment during fuel handling operations.
Footnote B-3	The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.	Conforms- With the acceptance of TSTF-51, the NRC has accepted that TS administrative controls are not needed provided there is a commitment made to conform with draft NUMARC 93-01, Revision 3, Section 11.2.6.

Responses to Commonly Raised NRC Issues
On FHA AST License Amendment Requests

A review has been performed of previous issues raised by the NRC as evidenced by the Requests for Additional Information (RAIs) that have been asked for BWR licensee FHA AST License Amendment Requests. This Appendix provides the NPPD responses to these RAI issues.

Question 1: There is less than 23 feet of water above the Spent Fuel Pool. Justify why an FHA associated with fuel in the reactor is more limiting than an FHA occurring over the Spent Fuel Pool.

Response: CNS Technical Specifications require a Spent Fuel Pool water level to be ≥ 21 ft 6 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. NPPD has determined that the difference in kinetic energy between an FHA in Spent Fuel Pool versus the Reactor Cavity Pool results in damage to 48 fuel rods versus 151. This is a 68 percent reduction in damaged rods.

The effective Decontamination Factor (DF) in RG 1.183 is based on an exponential function. In this function, more scrubbing occurs at the bottom of the water column than at the top of the water column. Therefore, assuming a linear change in DF as a function of water level would be conservative. As such, a pool level of 21 feet 6 inches compared with 23 feet would result in about a 6.5 percent reduction in DF.

In summary, the implied reduction in scrubbing efficiency is more than offset by the reduced number of fuel rods that are projected to be damaged by a fuel assembly drop over the spent fuel pool. Accordingly, the dose consequences of an FHA occurring over the Spent Fuel Pool will be bounded by the FHA occurring in the Reactor Cavity Pool. Note- This approach is similar to the one accepted by the NRC for the James A. Fitzpatrick Nuclear Power Station FHA AST application (see Page 4 to the Safety Evaluation of Reference 7.2-1).

Question 2: Describe the approach that will be used to conform to the commitment to implement the provisions of NUMARC 93-01 Section 11.3.6.5 when Secondary Containment and SGTS are inoperable during the handling of non-recently irradiated fuel. How much of an open area to the environment would be permitted? Describe the ventilation systems that would be used to draw the release from the postulated FHA (i.e., are they Engineered Safety Features, do they have carbon adsorber filters and high-efficiency particulate air filters, and are they tested in accordance with Regulatory Guide 1.52 or other standards?). Justify how these means will ensure that the NUMARC guidance "[these prompt methods] enable the ventilation systems to draw from the postulated FHA such that it can be treated and monitored" is met. Will there be a test to determine that all air flow was into the Secondary Containment in the event that partial closure is allowed?

Response: The approach that will be taken by NPPD 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), 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 strategy outlined above is designed to assure 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.

Question 3: Discuss the Control Room unfiltered bypass leakage assumption used in the FHA AST calculation compared with leakage results of inleakage testing performed per Generic Letter 2003-01.

Response: There are two issues related to the Generic Letter (GL) 2003-01 inleakage testing results that factor into the FHA AST calculation: 1) the assumed inleakage in the FHA calculation must bound the testing results, and 2) the receptor location for the X/Qs used in the FHA calculation must remain appropriately conservative relative to the tested inleakage.

Regarding the first issue, the unfiltered inleakage assumed in the AST FHA calculation is 400 cfm. This bounds the testing results of 64 cfm (CREFS pressurization mode) and 223 cfm (CREFS isolation mode) reported to the NRC in the response to GL 2003-01.

Regarding the second issue, the FHA calculation assumes a single receptor point at the Control Building HVAC intake. As part of the GL 2003-01 testing, NPPD has not attempted to determine the actual locations of the inleakage, or to quantify the amount of inleakage that ingress at points that may be closer to the Reactor Building vent than the Control Building intake. There are potential points of inleakage that are both greater than and less than the distance between the Reactor Building vent and the Control Building intake. NPPD believes that the Control Room X/Q (which assumes a ground level ingress at the Control Building HVAC intake) already provides inherent conservatism. Furthermore, from a geometric standpoint, it would be difficult to envision the magnitudes of wind direction shifts needed over the 2-hour release period that would maximize both the radionuclide concentration at the Control Building intake and potential points of closer inleakage. Additionally, inleakage points would typically be from adjacent buildings which would make plume pathways resulting in higher atmospheric dispersion factors improbable. Moreover, the 400 cfm unfiltered inleakage assumed in Enclosure 1 to this License Amendment Request provides a great deal of margin to the actual inleakage. For these reasons, assuming ingress at the Control Room HVAC intake provides conservative Control Room X/Qs and dose consequences that are appropriately bounding for the GL 2003-01 testing results.

Question 4: The only dose analyses provided for the FHA were those in which the dose was associated with fuel which was not "recently" irradiated. If it is ever intended to handle fuel which is "recently" irradiated, then an analysis needs to be provided that demonstrates acceptable dose results, both offsite and in the control room, in the event of an FHA. Are there any Technical Specifications or administrative controls that apply to when fuel movement is allowed post shutdown?

Response: Enclosure 1 assumes a 24 hour decay time since reactor shutdown prior to the FHA. Similarly, under the CNS Custom Technical Specifications 24 hours were required to elapse after shutdown before irradiated fuel movements could commence. As part of the conversion to the Improved Standard Technical Specifications (License Amendment 178), this TS was relocated to the Technical Requirements Manual (a document incorporated by reference into the USAR). The basis for this relocation was:

> The 24 hour decay time following subcriticality will always be met for a refueling outage because of the operations required prior to moving irradiated fuel in the reactor vessel (e.g., containment entry, removal of drywell head, removal of vessel head, removal of vessel internals).

> Therefore, the requirement is not necessary to be in Technical Specifications and its relocation will not adversely affect the public health and safety.

As a result of License Amendment 187, the TRM was changed from the previously relocated 24 hour decay time to the new 67 hour decay period. Approval of this License Amendment Request will result in restoring the original 24 hour value. In summary, the TRM will continue to provide the appropriate administrative controls to ensure that the assumed FHA decay time will be met. A dose calculation for an FHA occurring prior to the 24 hour decay time is unnecessary.

Question 5: If a fan is not operating within the reactor building (or shuts off due to Loss-of Offsite Power or single failure), would this result in the release occurring over a period longer than 2 hours, and would it result in a higher control room operator dose than the analysis provided? If the release occurred over 2 hours without the reactor building fan operating, would it result in a larger dose?

Response: Regulatory Guide 1.183 requires the design basis assumption of a 2-hour release regardless of whether there is forced ventilation or not. Holding all other variables constant (including the assumed CREFS actuation time within 1 minute), a release occurring over a longer period of time will result in lower Control Room doses. NPPD has assessed the ability of the Reactor Building Exhaust Plenum Radiation Monitor to provide the necessary CREFS initiation signal for the AST FHA. For personnel comfort considerations on the Refueling Bridge during fuel handling operations, the Reactor Building Heating and Ventilating System will typically be in service, which includes at least one exhaust fan. A LOOP is not assumed for accidents occurring during Modes 4 or 5 per the CNS TS Bases for AC Power Sources.[1] However, if a single failure occurs that disables the operating Reactor Building exhaust fan, a coastdown air flow will result that is sufficient (even for the 7 day decay time case) for the Radiation Monitor to provide the credited high radiation CREFS actuation within the assumed one minute time period after the design basis FHA. Since this setpoint assessment is predicated on having ventilation flow in the exhaust plenum at the start of the FHA, NPPD commits to the following prior to conducting fuel handling operations when less than a 7 day decay time has elapsed: 1) a Reactor Building exhaust fan or SGT System fan is in operation, or 2) CREFS is in operation.

Based on the above discussion, the Control Room dose consequences of Enclosure 1 are bounding over release periods that exceed 2 hours.

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1. Nonetheless, a LOOP occurring concurrent with the FHA would deenergize the Group 6 PCIS logic, and cause a CREFS initiation signal to occur.

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, and there are no 1967 GDCs that provide analogous requirements for assuring monitored releases following an FHA. Notwithstanding this, 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 monitored and filtered release via the SGT System, can be restored following an FHA.

ATTACHMENT 2

Proposed Technical Specifications
Markup Format

Cooper Nuclear Station, Docket 50-298, DPR-46

Listing of Revised Pages

TS Pages

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

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)	4	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	4	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)	4	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 ~~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)	4	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	4	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)	4	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.

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs.	C.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ---------------------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment. AND	 Immediately (continued)

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is $\geq$ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.3	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.4	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 1780 cfm.	18 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTES-----
1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	8 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment~~;~~ ~~during CORE ALTERATIONS,~~ or during OPDRVs.	D.1 ------NOTE------ LCO 3.0.3 is not applicable. ------------------ Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	~~AND~~	
	~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	D.~~3~~2 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the secondary containment~~;~~ ~~during CORE ALTERATIONS,~~ or during OPDRVs.	---------------NOTE--------------- LCO 3.0.3 is not applicable. ------------------------------------ C.1 Place OPERABLE SGT subsystem in operation. OR	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of recently irradiated fuel assemblies in secondary containment. ~~AND~~ ~~C.2.2 Suspend CORE ALTERATIONS.~~ AND C.2.~~3~~2 Initiate action to suspend OPDRVs.	Immediately ~~Immediately~~ Immediately
D. Two SGT subsystems inoperable in MODE 1 2, or 3.	D.1 Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs.	E.1 ---------NOTE---------- LCO 3.0.3 is not applicable. -------------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment. AND	Immediately (continued)

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
-------TEMPORARY NOTE------- The next required performance of this SR may be delayed until the current cycle refueling outage, but no later than February 2, 2005. This temporary note expires upon startup from that refueling outage.		
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
-------TEMPORARY NOTE------- The next required performance of this SR may be delayed until the current cycle refueling outage, but no later than February 2, 2005. This temporary note expires upon startup from that refueling outage.		
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

3.7 PLANT SYSTEMS

3.7.4 Control Room Emergency Filter (CREF) System

LCO 3.7.4 The CREF System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CREF System inoperable.	A.1 Restore CREF System to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of <u>recently</u> irradiated fuel assemblies in the secondary containment~~;~~ ~~during CORE ALTERATIONS,~~ or during OPDRVs.	------------NOTE------------- LCO 3.0.3 is not applicable. -----------------------------	
	C.1 Suspend movement of <u>recently</u> irradiated fuel assemblies in the secondary containment.	Immediately
	~~AND~~	
	~~C.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	<u>AND</u>	
	C.~~3~~<u>2</u> Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Operate the CREF System for $\geq$ 15 minutes.	31 days
		(continued)

ATTACHMENT 3

Proposed Technical Specifications
Final Typed Format

Cooper Nuclear Station, Docket 50-298, DPR-46

Listing of Revised Pages

TS Pages

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

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)	4	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	4	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)	4	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 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)	4	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	$\geq$ - 42 inches
2. Drywell Pressure - High	1,2,3	4	SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	$\leq$ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	4	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	$\leq$ 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.

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	C.1 ------------NOTE------------- LCO 3.0.3 is not applicable. --------------------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment. AND	 Immediately (continued)

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is $\geq$ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.3	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.4	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 1780 cfm.	18 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. <u>AND</u>	8 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	D.1 --------------NOTE------------- LCO 3.0.3 is not applicable. ----------------------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	D.2 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	---------------NOTE--------------- LCO 3.0.3 is not applicable. --- C.1 Place OPERABLE SGT subsystem in operation. OR	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
	C.2.2 Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1 2, or 3.	D.1 Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	E.1 ---------NOTE---------- LCO 3.0.3 is not applicable. -------------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
		(continued)

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
-------------------------------TEMPORARY NOTE------------------------------- The next required performance of this SR may be delayed until the current cycle refueling outage, but no later than February 2, 2005. This temporary note expires upon startup from that refueling outage.		
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
-------------------------------TEMPORARY NOTE------------------------------- The next required performance of this SR may be delayed until the current cycle refueling outage, but no later than February 2, 2005. This temporary note expires upon startup from that refueling outage.		
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

3.7 PLANT SYSTEMS

3.7.4 Control Room Emergency Filter (CREF) System

LCO 3.7.4 The CREF System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CREF System inoperable.	A.1 Restore CREF System to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	------------NOTE------------- LCO 3.0.3 is not applicable. ---------------------------- C.1 Suspend movement of recently irradiated fuel assemblies in the secondary containment. <u>AND</u> C.2 Initiate action to suspend OPDRVs.	 Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Operate the CREF System for ≥ 15 minutes.	31 days
		(continued)

ATTACHMENT 4

Proposed Technical Specifications Bases Revisions
Markup Format

Cooper Nuclear Station, Docket 50-298, DPR-46

Listing of Revised Pages

TS Bases Pages

B 3.3-172	**B 3.6-74**
B 3.3-189	**B 3.6-76**
B 3.6-67	**B 3.6-80**
B 3.6-68	**B 3.6-81**
B 3.6-69	**B 3.6-82**
B 3.6-70	**B 3.6-83**
B 3.6.72	**B 3.7-18**
B 3.6-73	**B 3.7-19**
	B 3.7-20

BASES

APPLICABILITY SAFETY ANALYSES, LCO, AND APPLICABILITY

3. Reactor Building Ventilation Exhaust Plenum Radiation-High (continued)

instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Ventilation Exhaust Plenum Radiation — High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during ~~CORE ALTERATIONS,~~ OPDRVs, and movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, this Function is only required to isolate secondary containment during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Building Ventilation Exhaust Plenum Radiation — High (continued)

The Reactor Building Ventilation Exhaust Plenum Radiation — High Function is required to be OPERABLE in MODES 1, 2, and 3 and during movement of recently irradiated fuel assemblies in the secondary containment, ~~CORE ALTERATIONS,~~ and operations with a potential for draining the reactor vessel (OPDRVs), to ensure control room personnel are protected during a pipe break resulting in significant releases of radioactive steam and gas, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., ~~CORE ALTERATIONS~~OPDRVs), the probability of a pipe break resulting in significant releases of radioactive steam and gas or fuel damage is low; thus, the Function is not required. Due to radioactive decay, this Function is only required to initiate the CREF System during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

ACTIONS

A Note has been provided to modify the ACTIONS related to CREF System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREF System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREF System instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the common interface with the Secondary Containment isolation Instrumentation, allowable out of service time of 12 hours for Functions 1 and 2, and 24 hours for Function 3, has been shown to be acceptable (Refs. 5, 6, and 7) to permit restoration of any inoperable channel to

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA) to limit fission product release to the environment. In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products released to the environment and to limit fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump and motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSIS

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1) and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of acritical reactor core within the previous 24 hours) inside secondary containment (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that fission products entrapped within the secondary containment structure following secondary containment isolation will be treated by the SGT System prior to discharge to the environment.

BASES

APPLICABLE SAFETY ANALYSES

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, following secondary containment isolation can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), ~~during CORE ALTERATIONS,~~ or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of critical reactor core within the previous 24 hours).

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary

BASES

ACTIONS

A.1 (continued)

containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status 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 12 hours and to MODE 4 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.

C.1~~,~~ and C.2~~, and C.3~~

Movement of recently irradiated fuel assemblies in the secondary containment~~, CORE ALTERATIONS,~~ and OPDRVs can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. ~~CORE ALTERATIONS and~~ Therefore, movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable in MODES 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,~~ Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend

BASES

ACTIONS

C.1~~,~~ and C.2~~, and C.3~~ (continued)

movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. Momentary transients on installed instrumentation due to gusty wind conditions are considered acceptable and are not cause for failure to meet this SR. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations during the applicable MODES and the low probability of a DBA occurring between surveillances.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. SR 3.6.4.1.2 also requires equipment hatches to be sealed. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. However, each secondary containment access door is normally kept closed, except when the access opening is being used for normal transient entry and exit or when maintenance is being performed on an access. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND	The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary. The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices. Automatic SCIVs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents. Other penetrations are isolated by the use of valves in the closed position or blind flanges.
APPLICABLE SAFETY ANALYSES	The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 3) and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) (Ref. 4). The secondary containment performs no active function in response to either of these limiting events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment System (SGT) System following secondary containment isolation, before being released to the environment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment following secondary containment isolation so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO are listed in Reference 6.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference 6.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE~~

BASES

APPLICABILITY (continued)

~~ALTERATIONS,~~ or during movement of recently irradiated fuel assemblies in the secondary containment. Moving recently irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3. Due to radioactive decay, SCIVs are only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

The second Note provides clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic SCIV, a closed manual valve, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to secondary containment. The Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to

BASES

ACTIONS B.1 (continued)

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in multiple penetrations.

C.1 and C.2

If any Required Action and associated Completion Time cannot be 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 12 hours and to MODE 4 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.

D.1, D.2, and D.3

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable~~, CORE ALTERATIONS~~ and the movement of recently irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,~~ Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

BASES

BACKGROUND (continued)

g. A centrifugal fan.

The capacity of the SGT System is sufficient to reduce and maintain the reactor building at a subatmospheric pressure of -0.25 inches water gauge (under neutral wind conditions of greater than 2 mph but less than 5 mph) with an air infiltration rate of no more than 100% of the reactor building volume per day.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both charcoal filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) (Ref. 2). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure. An OPERABLE SGT

BASES

LCO (continued)

subsystem consists of a demister, prefilter, HEPA filter, charcoal adsorber, a final HEPA filter, exhaust fan, and associated ductwork, dampers, valves and controls.

When the required decay heat removal flow through the cross tie damper is not met, only ONE SGT subsystem may be considered OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of <u>recently</u> irradiated fuel assemblies in the secondary containment. <u>Due to radioactive decay, the SGT System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).</u>

ACTIONS

<u>A.1</u>

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

<u>B.1 and B.2</u>

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, 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 12 hours and to MODE 4 within

BASES

ACTIONS

B.1 and B.2 (continued)

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.

C.1, C.2.1, and C.2.2~~, and C~~.2.3

During movement of recently irradiated fuel assemblies, in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the~~ Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

BASES

ACTIONS (continued)

D.1

If both SGTS subsystems are inoperable in MODE 1, 2, or 3, the SGT system may not be capable of supporting the required radioactivity release control function. Therefore, actions are required to enter LCO 3.0.3 immediately.

E.1~~,~~ and E.2~~, and E.3~~

When two SGT subsystems are inoperable, if applicable, ~~CORE ALTERATIONS~~ and movement of recently irradiated fuel assemblies in secondary containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,~~ ~~r~~Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem, including each filter train fan, for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of

BASES

APPLICABLE SAFETY ANALYSES

The ability of the CREF System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the USAR, Chapters X and XIV (Refs. 1 and 2, respectively). The CREF System is assumed to operate following a loss of coolant accident and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

The CREF System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The CREF System is required to be OPERABLE, since total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

The CREF System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE. The system is considered OPERABLE when its associated:

a. Fans are OPERABLE (one supply fan, the emergency booster fan and the exhaust booster fan);

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that the pressurization limit of SR 3.7.4.4 can be met. However, it is acceptable for access doors to be open for normal control room entry and exit, and not consider it to be a failure to meet the LCO.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the CREF System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

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

~~b. During CORE ALTERATIONS; and~~

~~c~~b. During movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the CREF System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

ACTIONS

A.1

The inoperable CREF System must be restored to OPERABLE status within 7 days. With the unit in this condition, there is no other system to perform control room radiation protection. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CREF System cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

BASES

ACTIONS

C.1~~, C.2,~~ and C.~~3~~2

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of recently irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, if the inoperable CREF System cannot be restored to OPERABLE status within the required Completion Time, activities that present a potential for releasing radioactivity that might require isolation of the control room must be immediately suspended. This places the unit in a condition that minimizes risk.

If applicable, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the CREF System in a standby mode starts on demand and continues to operate. The system should be checked periodically to ensure that it starts and functions properly. As the environmental and normal operating conditions of this system are not severe, testing the system once every month provides an adequate check on this system. Since the CREF System does not contain heaters, the system need only be operated for $\geq$ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment.

Correspondence Number: NLS2005075

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
NPPD 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.	NLS2005075-01	Within 30 days of issuance of the license amendment.
NPPD commits to the following prior to conducting fuel handling operations when less than a 7 day decay time has elapsed: 1) a Reactor Building exhaust fan or SGT system fan is in operation, or 2) CREFS is in operation.	NLS2005075-02	Within 30 days of issuance of the license amendment.