CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

MAY 1 8 2001

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SERIAL: HNP-01-068
10CFR50.90

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
TECHNICAL SPECIFICATIONS 3/4.9.4 AND UNREVIEWED SAFETY QUESTION

Dear Sir or Madam:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L) requests a revision to the Technical Specifications (TS) for the Harris Nuclear Plant (HNP). The proposed amendment revises Technical Specifications (TS) 3/4. 9.4 "Containment Building Penetrations" and associated Bases. Specifically, HNP proposes; 1) incorporating an alternate source term methodology in the fuel handling accident analysis, 2) to revise the applicable TS to remove portions of a note restricting applicability of administrative controls with respect to containment penetrations, and 3) to include the use of administrative controls on the equipment hatch and other penetrations that provide access from containment atmosphere to outside atmosphere.

Enclosure 1 provides a description of the proposed changes and the basis for the changes. Enclosure 2 details, in accordance with 10 CFR 50.91(a), the basis for CP&L's determination that the proposed changes do not involve a significant hazards consideration. Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment is required for approval of this amendment request. Enclosure 4 provides page change instructions for incorporating the proposed revisions. Enclosure 5 provides the proposed Technical Specification pages. Enclosure 6 provides a summary of the Alternate Source Term Fuel Handling Accident Analysis.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of North Carolina with a copy of the proposed license amendment.

CP&L requests that the proposed amendment be issued prior to refueling outage 10 and such that implementation will occur within 60 days of issuance to allow time for procedure revision and orderly incorporation into copies of the Technical Specifications.

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New Hill, NC 27562

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A001

Please refer any questions regarding this submittal to Mr. E. A. McCartney at (919) 362-2661.

Sincerely,

James Scarola

James Scarola

MSE/mse

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Environmental Considerations
4. Page Change Instructions
5. Technical Specification Pages
6. Alternate Source Term Fuel Handling Accident Analysis

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Darlene S. Yarborough

Notary (Seal)
My commission expires: 2-21-2005

DARLENE S. YARBOROUGH NOTARY *** PUBLIC HARNETT COUNTY, NC

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. Mel Fry, Director, N.C. DEHNR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

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BASIS FOR CHANGE REQUEST

Background

On March 27, 2000, the Nuclear Regulatory Commission (NRC) issued License Amendment 97 to the Harris Nuclear Plant (HNP). That amendment revised Technical Specification (TS) 3/4.9.4, "Containment Building Penetrations", and its associated Bases, to allow the personnel airlock and certain other containment building penetrations to remain open during refueling operations provided specific administrative controls are met. That amendment was approved for use during refueling outage 9 and operating cycle 10.

The NRC specified in the evaluation for License Amendment 97 that the staff is currently working toward resolution of generic issues related to control room habitability, in particular the validity of control room infiltration rates assumed by the licensees in analyses of control room habitability. The staff approved License Amendment 97 based on the licensee's assertion that doses to the control room staff remain bounded by the loss-of –coolant accident (LOCA) analysis for the upcoming fuel cycle (refueling outage 9 and cycle 10). The generic industry issues described in License Amendment 97 have not yet been resolved. Therefore, HNP has performed an alternate source term analysis to provide a justification to continue to use the provisions of TS 3/4.9.4 beyond cycle 10. HNP understands that approval of this proposed license amendment will not exempt CP&L from regulatory actions that may be implemented in the future as the control room habitability generic issue is resolved. Additionally, HNP is requesting to implement the guidance of TSTF-312 to expand the use of administrative controls to all containment penetrations specified in TS 3/4.9.4.

Proposed Change

HNP proposes to revise Technical Specification (TS) 3/4.9.4, "Containment Building Penetrations", to allow use of administrative controls on open containment penetrations during core alterations and movement of irradiated fuel beyond cycle 10 and to implement the guidance of TSTF-312. Additionally, HNP proposes to revise the fuel handling accident analyses to adopt the alternate source term methodology using the guidance of NRC Regulatory Guide 1.183.

Basis for the Proposed Change

As documented in a draft NEI 99-03 (dated January 2000), several nuclear plants performed testing on control room unfiltered inleakage that demonstrated leakage rates in excess of amounts assumed in the accident analysis. HNP performed alternate source term analyses for a fuel handling accident in containment and the fuel handling building. These analyses provide for a control room unfiltered inleakage of in excess of 300 cfm, which is the design basis value that HNP is proposing. The unfiltered inleakage, assumed in the proposed fuel handling accident analyses, is 500 cfm, which is well above the 300 cfm design basis value that HNP proposes to establish. HNP plans to submit similar alternate source term analyses for other accidents, which, in some cases, will assume the more restrictive design basis value of 300 cfm unfiltered inleakage. The use of 300 cfm unfiltered inleakage as a design basis value is expected to be well above the unfiltered inleakage value determined through testing or analysis consistent with resolution of issues identified in NEI 99-03.

The current HNP TS 3/4.9.4 Bases describes administrative controls used to restrict opening certain penetrations that communicate between the Reactor Containment Building atmosphere and the Reactor Auxiliary Building Ventilation System atmosphere. The proposed analysis assumes that all activity resulting from the fuel handling accident in containment is released to the atmosphere. Therefore, there is no longer a need to restrict penetration openings.

TSTF-312 provides administrative controls for maintaining containment penetrations open. The proposed amendment provides a note that states that penetration flow path(s) providing direct access from the containment atmosphere to outside atmosphere may be unisolated under administrative controls. The proposed Bases state that the administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during core alterations or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

The proposed Bases also state that the allowance to have containment personnel airlock doors open and penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and core alterations is based on (1) confirmatory dose calculations of a fuel handling accident as approved by the NRC staff which indicates acceptable radiological consequences and (2) commitments from the licensee to implement acceptable administrative procedures that ensure, in the event of a fuel handling accident (even though the containment fission product control function is not required to meet acceptable dose consequences) that the open airlock or equipment hatch can and will be promptly closed.

HNP has performed alternate source term analyses for fuel handling accidents in containment and the fuel handling building. The analyses assume that activity produced by the accident escapes to the outside atmosphere. Other key assumptions used are included in Enclosure 6.

Administrative Controls

HNP committed by letter, dated February 24, 2000, to provide written procedures that:

(1) An individual or individuals shall be designated and available at all times, capable of closing the breached penetration.
(2) The breached penetration shall not be obstructed unless capability for rapid removal is provided (such as quick disconnects for hoses).
(3) For the Personnel Air Lock, at least one door must be capable of being closed. HNP incorporated these changes in the TS Bases and plant procedure OMP-003. The NRC approved the use of these administrative controls in the Safety Evaluation Report for License Amendment 97 dated March 27, 2000.

HNP proposes to retain these administrative controls and expand it to include the equipment hatch and other penetrations that provide direct access from the containment atmosphere and outside atmosphere. HNP has measured the length of time to close any penetration (including the equipment hatch) and demonstrated that any penetration can be closed in less than one hour. HNP commits to incorporating administrative controls into procedures to ensure that all penetrations during a fuel handling accident will be closed in less than the two-hour release assumed in the accident analysis.

Additionally, HNP is proposing to provide clarification (in the TS Bases) that equivalent isolation methods can be used for the air lock(s) and equipment hatch as well as other penetrations. The description of equivalent isolation was included (and subsequently approved) in the submittal for License Amendment 97.

Dose Consequences

The table below provides the fuel handling accident (both inside containment and inside the fuel handling building) dose consequences from the alternate source term analyses with respect to the site boundary or exclusion area boundary (EAB), low population zone (LPZ), and the control room.

	EAB Dose (TEDE)	LPZ Dose (TEDE)	Control Room (TEDE)
FHA- Containment	2.2	0.6	1.5
FHA- Fuel Handling Building	0.5	0.2	0.2
Limit (RG 1.1.83 & 10CFR50.67)	6.3	6.3	5.0

Conclusion

HNP is requesting this change to incorporate guidance from TSTF-312 as well as implementing the alternate source term analyses for fuel handling accidents in containment and the fuel handling building. HNP has demonstrated through analyses that doses to the public and to control room operators remain well below required limits. Additionally, HNP proposes administrative controls to provide defense-in-depth to limit the consequences of fuel handling accidents in containment.

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10 CFR 50.92 EVALUATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an 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. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards determination. The bases for this determination are as follows:

Proposed Change

HNP proposes to revise Technical Specification (TS) 3/4.9.4, "Containment Building Penetrations", to allow use of administrative controls on open containment penetrations during core alterations and movement of irradiated fuel beyond cycle 10 and to implement the guidance of TSTF-312. Additionally, HNP proposes to revise the fuel handling accident analyses to adopt the alternate source term methodology using the guidance of NRC Regulatory Guide 1.183.

Basis

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes modify TS requirements previously reviewed and approved by the NRC in improved Technical Specifications (ITS) and changes to ITS as described in TSTF-312. An alternate source term calculation has been performed for the HNP that demonstrates that dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed change does not modify the design or operation of equipment used to move spent fuel or to perform core alterations

   Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Containment penetrations are designed to form part of the containment pressure boundary. The proposed change provides for administrative controls and operating restrictions for containment penetrations consistent with guidance approved by the NRC staff. Containment penetrations are not an accident initiating system as described in the Final Safety Analysis Report. The proposed change does not affect other Structures, Systems, or Components. The operation and design of containment penetrations in operational modes 1-4 will not be affected by this proposed change.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

   The proposed changes modify required Actions and Surveillance Requirements previously reviewed and approved by the NRC in improved Technical Specifications (ITS) and changes to ITS, TSTF-312. Additionally, the implementation of the alternate source term methodology is consistent with NRC Regulatory Guide 1.183. The proposed change to containment penetrations does not significantly affect any of the parameters that relate to the margin of safety as described in the Bases of the TS or the FSAR. Accordingly, NRC Acceptance Limits are not significantly affected by this change.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; (3) result in a significant increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Proposed Change

HNP proposes to revise Technical Specification (TS) 3/4.9.4, "Containment Building Penetrations", to allow use of administrative controls on open containment penetrations during core alterations and movement of irradiated fuel beyond cycle 10 and to implement the guidance of TSTF-312. Additionally, HNP proposes to revise the fuel handling accident analyses to adopt the alternate source term methodology using the guidance of NRC Regulatory Guide 1.183.

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in Enclosure 2, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite.

   The change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite.

3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure.

   The proposed change does not result in any physical plant changes or new surveillances which would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either individual or cumulative occupational radiation exposure.

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PAGE CHANGE INSTRUCTIONS

Removed Page	Inserted Page
3/4 9-5	3/4 9-5
B3/4 9-1	B3/4 9-1
B3/4 9-2	B3/4 9-2

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TECHNICAL SPECIFICATION PAGES

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building ~~penetrations shall~~ be in the following status:

is capable of being - Add

a. The equipment door ~~closed and held~~ in place by a minimum of four bolts~~,~~ * - Add

b. A minimum of one door in each airlock is capable of being closed*, and

Delete

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Be capable of being* closed by a manual or automatic isolation valve, blind flange or equivalent, or

2. Be capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves~~.~~ * Add

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

Delete

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition, capable of being closed/isolated*, or capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are either closed/isolated or capable of being closed/isolated*, or

b. Testing the normal containment purge and containment pre-entry purge makeup and exhaust isolation valves per the applicable portions of Specification 4.6.3.2.

Add

flow path(s) providing direct access from the containment atmosphere to the outside atmosphere

Delete

Delete

* ~~Penetrations~~ may be opened under administrative controls~~, except for containment purge and exhaust penetrations. This allowance is permitted for refueling outage 9 and cycle 10 only. Operation under these administrative controls has not been approved for refueling outage 10.~~

Delete

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses and are specified in the cycle-specific COLR. The boron concentration limit specified in the COLR ensures that a core K_{eff} of ≤ 0.95 is maintained during fuel handling operations. The administrative controls over the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME - DELETED

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Penetrations applicable to Technical Specification 3.9.4.b and 3.9.4.c may be opened provided the following administrative controls are in effect:

1. An individual or individuals shall be designated and available at all times, capable of isolating the breached penetration.

2. The breached penetrations shall not be obstructed unless capability for rapid removal of obstructions is provided (such as quick disconnects for hoses).

3. For the Personnel Air Lock, at least one door must be capable of being closed and secured. Additionally, the equipment hatch must be capable of being closed and secured. Equivalent isolation methods may also be used. Add

4. Only penetrations that communicate between the Reactor Containment Building atmosphere and the Reactor Auxiliary Building Ventilation System atmosphere are permitted to be open under these administrative controls.

Delete

Insert A & B

Containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated, or capable of isolation via adminstratvie controls, on at least one side of containment. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movement.

3/4.9.5 COMMUNICATIONS - DELETED

Delete

Insert A

The LCO is modified by a Note allowing penetration flow paths providing direct access from the containment atmosphere to the outside atmosphere to be open under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

Insert B

The allowance to have containment penetration (including the airlock doors and equipment hatch) flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on (1) confirmatory dose calculations as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitments from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident that the airlock or equipment hatch can and will be promptly closed following containment evacuation (even though the containment fission product control function is not required to meet acceptable dose consequences) and that the open penetration(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.

Delete 1

3/4.9.6 REFUELING MACHINE - DELETED

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING - DELETED

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

The minimum RHR flow requirement is reduced to 900 gpm when the reactor water level is below the reactor vessel flange. The 900 gpm limit reduces the possibility of cavitation during operation of the RHR pumps and ensures sufficient mixing in the event of a MODE 6 boron dilution incident.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge makeup and exhaust penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

Delete Add

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door is cabable of being closed and held in place by a minimum of four bolts*,

b. A minimum of one door in each airlock is capable of being closed*, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Be capable of being* closed by a manual or automatic isolation valve, blind flange or equivalent, or

   2. Be capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves*.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition, capable of being closed/isolated*, or capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are either closed/isolated or capable of being closed/isolated*, or

b. Testing the normal containment purge and containment pre-entry purge makeup and exhaust isolation valves per the applicable portions of Specification 4.6.3.2.

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* Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be opened under administrative controls.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses and are specified in the cycle-specific COLR. The boron concentration limit specified in the COLR ensures that a core K_{eff} of ≤ 0.95 is maintained during fuel handling operations. The administrative controls over the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME - DELETED

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Penetrations applicable to Technical Specification 3.9.4.b and 3.9.4.c may be opened provided the following administrative controls are in effect:

1. An individual or individuals shall be designated and available at all times, capable of isolating the breached penetration.

2. The breached penetrations shall not be obstructed unless capability for rapid removal of obstructions is provided (such as quick disconnects for hoses).

3. For the Personnel Air Lock, at least one door must be capable of being closed and secured. Additionally, the equipment hatch must be capable of being closed and secured. Equivalent isolation methods may also be used.

The LCO is modified by a Note allowing penetration flow paths providing direct access from the containment atmosphere to the outside atmosphere to be open under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

CONTAINMENT BUILDING PENETRATIONS (Continued)

The allowance to have containment penetration (including the airlock doors and equipment hatch) flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on (1) confirmatory dose calculations as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitments from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident that the airlock or equipment hatch can and will be promptly closed following containment evacuation (even though the containment fission product control function is not required to meet acceptable dose consequences) and that the open penetration(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.

Containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated, or capable of isolation via administrative controls, on at least one side of containment. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movement.

3/4.9.5 COMMUNICATIONS - DELETED

3/4.9.6 REFUELING MACHINE - DELETED

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING - DELETED

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

The minimum RHR flow requirement is reduced to 900 gpm when the reactor water level is below the reactor vessel flange. The 900 gpm limit reduces the possibility of cavitation during operation of the RHR pumps and ensures sufficient mixing in the event of a MODE 6 boron dilution incident.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge makeup and exhaust penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

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ALTERNATE SOURCE TERM FUEL HANDLING ACCIDENT

Engineering Report for the Radiological Consequences of the Fuel Handling Accident Using Alternative Source Term Methodology (Regulatory Guide 1.183) for the Shearon Harris Nuclear Power Plant

Table of Contents

1.0 Fuel Handling Accident (FHA) Radiological Analysis

1.1 Introduction

The Shearon Harris Nuclear Power Plant (SHNPP) licensing basis for the fuel handling accident radiological consequences analyses for Chapter 15 of the UFSAR is currently based on methodologies and assumptions that are derived from TID-14844 (Reference 1) and other early guidance.

Regulatory Guide (RG) 1.183 (Reference 2) provides guidance on application of alternative source terms (AST) in revising the accident source terms used in design basis fuel handling accident radiological consequence, as allowed by 10CFR50.67 (Reference 3).

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed with assumptions selected for the accident occurring either inside containment or in the fuel handling building. Activity released from the damaged assembly is released to the outside atmosphere through either the containment openings (such as the personnel air lock door or the equipment hatch) or the fuel pool ventilation system.

1.2 Common Analysis Inputs and Assumptions

The assumptions and inputs described in this section are common to analyses discussed in this report. The accident specific inputs and assumptions are discussed in Sections 2 and 3.

The total effective dose equivalent (TEDE) doses are determined at the site boundary (SB), at the low population zone (LPZ) and to control room personnel (CR) for the duration of the event. The SB and LPZ doses are determined for the 0 to 2 hour time period since the releases are over a 2 hour time period. The control room dose is reported at 24 hours even though releases are terminated at 2 hours. This accounts for the additional dose to the operators in the control room, which will continue for as long as the activity is circulating within the control room envelope.

The dose conversion factors (DCFs) used in determining the committed effective dose equivalent (CEDE) or inhalation dose are from Reference 5 and are given in Table 1. The dose conversion factors used in determining the effective dose equivalent (EDE) or the whole body external exposure or the acute dose for the duration of exposure to the cloud are from Reference 12 and are listed in Table 2. The TEDE dose is equivalent to the CEDE dose plus the EDE dose for the duration of exposure to the cloud.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table 3.

Parameters modeled in the control room personnel dose calculations are provided in Table 4. These parameters include the normal operation flowrates, the emergency operation flowrates, control room volume, filter efficiencies and control room operator breathing rates. In the analyses presented in this report, the control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for release of activity from the release point to the control room intake are used to determine the activity available at the intake. These factors have been chosen to bound credible release points for these events, and are the same factors used in the current SHNPP control room dose calculations. The inflow (filtered and unfiltered) to the control room and the control room recirculation flow are used to calculate the activity introduced to the control room and cleanup of activity from that flow.

The core fission product activity is provided in Table 5 for all nuclides. The core activities in Table 5 are based on a core power of 2900 MWt increased to 2958 to cover 2% uncertainty. Decay constants for each nuclide are provided in Table 6 and are from Reference 6 for the iodine and noble gas nuclides. The core activities are converted to limiting FHA activities by dividing by the number of fuel assemblies in the core, applying the appropriate local peaking factors and reducing the activity to account for the appropriate decay.

2.0 Fuel Handling Accident In Containment Analysis

2.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 7. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours. No credit is taken for isolation of containment for the FHA in containment.

2.1.1 Source Term

Consistent with Regulatory Guide 1.183 Position 1.2 of Appendix B, the radionuclides considered are xenons, kryptons, halogens, cesiums and rubidiums. The list of xenons, kryptons, and halogens considered is given in Table 6 and is based on the combined effect of the initial amount of activity in the core, the dose conversion factor and the half life for the respective nuclide. The cesium and rubidium are not included because they are not assumed to be released from the pool as discussed later.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, 10% for Kr-85, and 5% for all other nuclides.

As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly (264 rods) are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.73 times the core average power.

The decay time used in the analysis is 100 hours. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement.

2.1.2 Fission Product Form

In accordance with Reference 2, Iodine species in the pool is 99.85% elemental and 0.15% organic. This is based on the split leaving the fuel of 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine. It assumed that all CsI is dissociated in the water and re-evolves as elemental. This is assumed to occur instantaneously. Thus, 99.85% of the iodine released is elemental.

2.1.3 Pool Scrubbing Removal of Activity

Reg. Guide 1.183 (Reference 2) provides that for 23 feet of water above the fuel, or greater, the DF for elemental and organic iodine are 500 and 1, respectively. The Reg. Guide goes on to say that this results in an overall effective DF of 200. Performing the arithmetic, in accordance with the formulas cited in the Reg. Guide Reference B-1, the numerical result for overall effective DF is approximately 286. The overall effective DF of 200, therefore, represents a conservative approximation of the results of the detailed calculation. Using other formulas in the cited Reg. Guide Reference B-1, it was determined that for the SHNPP specific water height above the failed fuel in the containment of 22 feet, the elemental DF would be 382, instead of the Reg. Guide allowable elemental DF of 500. Using the SHNPP specific value of elemental DF in the formulas discussed above, it was determined that the actual SHNPP specific DF for 22 feet of coverage would be 243. Since this continues to exceed the Reg. Guide cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the SHNPP dose calculations.

Using an overall DF of 200 gives an elemental DF of 286. The iodine chemical split above the pool is 70% elemental and 30% organic. This is different than the Reg. Guide 1.183 value for the iodine chemical split above the pool of 57% elemental and 43% organic. The Reg. Guide 1.183 uses an elemental DF of 500. A smaller overall pool DF results in less activity being removed from the pool which is conservative.

The split between elemental and organic iodine leaving the pool has no impact on the analysis since the control room filter efficiencies are the same, and no other filtration is credited.

The cesium and rubidium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

2.1.4 Isolation and Filtration of Release Paths

No credit is taken for removal of iodine by filters nor is credit taken for isolation of release paths.

Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be released to the outside atmosphere over a 2 hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open.

2.1.5 Control Room Isolation

It is assumed that the control room HVAC system is initially operating in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room enters pressurization mode with operator action at 2 hours after isolation signal.

2.2 Acceptance Criteria

The offsite dose limit for a fuel handling accident is 6.3 rem TEDE per RG 1.183. This is ~25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

2.3 Results and Conclusions

The amount of activity released to the atmosphere is given in Table 8.

The fuel handling accident in containment doses are:

Site Boundary (0-2 hr)	2.2 rem TEDE
Low Population Zone (0-2 hr)	0.6 rem TEDE
Control Room (0-2 hr)	1.3 rem TEDE
Control Room (0-24 hr)	1.5 rem TEDE

The reported fuel handling accident doses listed above have been increased by approximately 5% from the actual analysis results to provide margin to accommodate potential small changes in analysis parameters in the future without requiring a change in the reported doses.

The acceptance criteria are met.

3.0 Fuel Handling Accident In Fuel Building Analysis

3.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 9. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly onto 52 Brunswick BWR fuel assemblies. This analysis also includes 50 PWR rods additionally damaged in the accident. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours.

3.1.1 Source Term

Consistent with Regulatory Guide 1.183 Position 1.2 of Appendix B, the radionuclides considered are xenons, kryptons, halogens, cesiums and rubidiums. The list of xenons, kryptons, and halogens considered is given in Table 6 and is based on the combined effect of the initial amount of activity in the core, the dose conversion factor and the half life for the respective nuclide. The cesium and rubidium are not included because they are not assumed to be released from the pool as discussed later.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, 10% for Kr-85, and 5% for all other nuclides.

As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly plus 50 additional PWR rods (314 rods) plus 52 Brunswick BWR fuel assemblies are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the PWR fuel assembly has been operated at 1.73 times the core average power and the BWR fuel assembly has been operated at 1.5 times the core average power.

The BWR fuel inventory was conservatively evaluated at the IF-300 shipping cask limits recently approved in Reference 13. The decay time used in the analysis is 100 hours for the PWR fuel and 4 years for the BWR fuel. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement.

3.1.2 Fission Product Form

Iodine species in the pool is 99.85% elemental and 0.15% organic. This is based on the split leaving the fuel of 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine. It assumed that all CsI is dissociated in the water and re-evolves as elemental. This is assumed to occur instantaneously. Thus, 99.85% of the iodine released is elemental.

3.1.3 Pool Scrubbing Removal of Activity

Reg. Guide 1.183 (Reference 2) provides that for 23 feet of water above the fuel, or greater, the DF for elemental and organic iodine are 500 and 1, respectively. The Reg. Guide goes on to say that this results in an overall effective DF of 200. Performing the arithmetic, in accordance with the formulas cited in the Reg. Guide Reference B-1, the numerical result for overall effective DF is approximately 286. The overall effective DF of 200, therefore, represents a conservative approximation of the results of the detailed calculation. Using other formulas in the cited Reg. Guide Reference B-1, it was determined that for the SHNPP specific water height above the failed fuel in the fuel handling building of 21 feet, the elemental DF would be 291, instead of the Reg. Guide allowable elemental DF of 500. Using the SHNPP specific value of elemental DF in the formulas discussed above, it was determined that the actual SHNPP specific DF for 21 feet of coverage would be 203. Since this continues to exceed the Reg. Guide cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the SHNPP dose calculations.

Using an overall DF of 200 gives an elemental DF of 286. The iodine chemical split above the pool is 70% elemental and 30% organic. This is different than the Reg. Guide 1.183 value for the iodine chemical split above the pool of 57% elemental and 43% organic. The Reg. Guide 1.183 uses an elemental DF of 500. A smaller overall pool DF results in less activity being removed from the pool which is conservative.

The split between elemental and organic iodine leaving the pool has no impact on the analysis since the control room filter efficiencies are the same.

The cesium and rubidium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

3.1.4 Isolation and Filtration of Release Paths

Credit is taken for removal of iodine by filters with the spent fuel pool ventilation system operation. Credit is not taken for isolation of release paths.

The activity released from the damaged assembly is assumed to be released to the outside atmosphere over a 2 hour period.

3.1.5 Control Room Isolation

It is assumed that the control room HVAC system begins in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room enters pressurization mode at 2 hours after isolation signal.

3.2 Acceptance Criteria

The offsite dose limit for a fuel handling accident is 6.3 rem TEDE per RG 1.183. This is ~25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

3.3 Results and Conclusions

The amount of activity released to the atmosphere is given in Table 10.

The fuel handling accident in fuel building doses are:

Site Boundary (0-2 hr)	0.5 rem TEDE
Low Population Zone (0-2 hr)	0.2 rem TEDE
Control Room (0-2 hr)	0.1 rem TEDE
Control Room (0-24 hr)	0.2 rem TEDE

The reported fuel handling accident doses listed above have been increased by approximately 5% from the actual analysis results to provide margin to accommodate potential small changes in analysis parameters in the future without requiring a change in the reported doses.

The acceptance criteria are met.

4.0 Conclusions

Regulatory Guide 1.183 defines an alternate source term model for use in evaluating the radiological consequences of a fuel handling accident. The fuel handling accident analysis has been evaluated and found to meet all acceptance criteria.

Implementation of this alternative source term methodology into the Shearon Harris Nuclear Power Plant's design basis allows for the movement of fuel in the containment with the equipment hatch and/or personnel air lock open.

5.0 References

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3. NRC Final Rule 10CFR50.67, issued in Federal Register, Vol. 64, No. 246, pages 71990-72002, 12/23/99.
4. U.S. Nuclear Regulatory Commission NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.
5. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88—0202, September 1988.
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7. International Commission on Radiological Protection, "Radionuclide Transformations, Energy and Intensity of Emissions," ICRP Publication 38, 1983.
8. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, 1979.
9. NUREG-0800, Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988.
10. Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983
11. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Proceedings of the Thirteenth AEC Air Cleaning Conference held August 1974, published March 1975.
12. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water and Soil," EPA 402-R-93-081, September 1993.
13. IF-300 Cask License, Certificate of Compliance #9001, New Appendix D, NEDO-10084-5.

Table 1: Committed Effective Dose Equivalent Dose Conversion Factors

Isotope	DCF (rem/curie)
I-131	3.29E4
I-132	3.81E2
I-133	5.85E3
I-134	1.31E2
I-135	1.23E3
Kr-85m	N/A
Kr-85	N/A
Kr-87	N/A
Kr-88	N/A
Xe-131m	N/A
Xe-133m	N/A
Xe-133	N/A
Xe-135m	N/A
Xe-135	N/A
Xe-138	N/A

Table 2: Effective Dose Equivalent Dose Conversion Factors

Isotope	Energy (rem· m^3/Ci· sec)
I-131	6.734E-2
I-132	0.4144
I-133	0.1088
I-134	0.4810
I-135	0.2953
Kr-85m	2.768E-2
Kr-85	4.403E-4
Kr-87	0.1524
Kr-88	0.3774
Xe-131m	1.439E-3
Xe-133m	5.069E-3
Xe-133	5.772E-3
Xe-135m	7.548E-2
Xe-135	4.403E-2
Xe-138	0.2135

Table 3: Offsite Breathing Rates and Atmospheric Dispersion Factors

	Offsite Breathing Rates (m^3/sec)
0 - 8 hours	3.5E-4
8 - 24 hours	1.8E-4
>24 hours	2.3E-4

	Offsite Atmospheric Dispersion Factors (sec/m^3)
Site Boundary	6.17E-4
Low Population Zone	
0 - 8 hours	1.4E-4
8 - 24 hours	1.0E-4
1 - 4 days	5.9E-5
> 4 days	2.4E-5

Table 4: Control Room (CR) Parameters

Volume (ft^3)	71,000
Normal Ventilation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	0.0
Unfiltered Makeup Flow Rate	1050.0
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Post Accident Recirculation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	4000.0
Unfiltered Inleakage	500.0
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Pressurization Mode Flow Rates (cfm)	
Filtered Makeup Air Flow Rate	400.0
Filtered Recirculation Flow Rate	3600.0
Unfiltered Inleakage	500.0
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Filter Efficiencies (%)	
Elemental	99
Organic	99
Particulate	99
CR Radiation Monitor Sensitivity (μCi/ml for Xe-133)	3.0E-6
CR Radiation Monitor Location	Emergency & normal air intakes
Delay to Initiate Switchover of Post Accident signal Recirculation HVAC mode after radiation	15 seconds
Operator Action Time to Switch to Pressurization Mode	2 hours
Breathing Rate - Duration of the Event (m^3/sec)	3.5E-4
Atmospheric Dispersion Factors (sec/m^3)	
0 – 8 hours	4.08E-3
8 – 24 hours	1.16E-3
1 – 4 days	3.25E-4
4 – 30 days	1.23E-5
Occupancy Factors*	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

* These occupancy factors (from Reference 11) have been conservatively incorporated in the atmospheric dispersion factors. This is conservative since it does not allow the benefit of reduced occupancy for activity already present in the control room from earlier periods.

Table 5: Core Total Fission Product Activities

Based on 102% of 2900 MWt

Isotope	Activity (Ci)
I-131	8.02E+07
I-132	1.16E+08
I-133	1.64E+08
I-134	1.80E+08
I-135	1.53E+08
Kr-85	8.62E+05
Kr-85m	2.19E+07
Kr-87	4.22E+07
Kr-88	5.95E+07
Xe-131m	8.96E+05
Xe-133	1.60E+08
Xe-133m	5.12E+06
Xe-135	3.83E+07
Xe-135m	3.21E+07
Xe-138	1.37E+08

Table 6: Nuclide Decay Constants

Isotope	Decay Constant (hr^{-1})
I-131	0.00359
I-132	0.303
I-133	0.0333
I-134	0.791
I-135	0.105
Kr-85m	0.155
Kr-85	7.37E-6
Kr-87	0.547
Kr-88	0.248
Xe-131m	0.00241
Xe-133m	0.0130
Xe-133	0.00546
Xe-135m	2.72
Xe-135	0.0756
Xe-138	2.93

Table 7: Assumptions Used for FHA in Containment Dose Analysis

Radial peaking factor	1.73
Fuel damaged (number of assemblies)	1
Time from shutdown before fuel movement (hr)	100
Activity in the damaged fuel assembly (Ci)	
I-131	6.06E5
I-132	0.0
I-133	6.38E4
I-134	0.0
I-135	4.68E1
Kr-85m	0.0
Kr-85	8.82E3
Kr-87	0.0
Kr-88	0.0
Xe-131m	7.61E3
Xe-133m	1.49E4
Xe-133	9.97E5
Xe-135m	0.0
Xe-135	2.03E2
Xe-138	0.0
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Water depth	22 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0
Filter efficiency	No filtration assumed
Isolation of release	No isolation assumed
Time to releases all activity (hours)	2

Table 7 Continued: Assumptions Used for FHA in Containment Dose Analysis

Time to start crediting post accident recirculation control room HVAC (seconds)	15
Time to start crediting pressurization HVAC mode (hours)	2

Table 8: FHA in Containment – Activity Released to Atmosphere

Nuclide	Amount of Activity (Curies)
I-131	2.420E2
I-132	0.0
I-133	1.592E1
I-134	0.0
I-135	1.168E-2
Kr-85m	0.0
Kr-85	8.820E2
Kr-87	0.0
Kr-88	0.0
Xe-131m	3.805E2
Xe-133m	7.450E2
Xe-133	4.985E4
Xe-135m	0.0
Xe-135	1.015E1
Xe-138	0.0

Table 9: Assumptions Used for FHA in the Fuel Handling Building Dose Analysis

Radial peaking factor (PWR fuel)	1.73
(BWR fuel)	1.5
Fuel damaged (number of assemblies)	1.2 PWR (314 rods) + 52 BWR
Time from shutdown before fuel movement (PWR) (hr)	100
(BWR fuel) (yr)	4
Activity in the damaged fuel assemblies (Ci)	
I-131	7.21E5
I-132	0.0
I-133	7.59E4
I-134	0.0
I-135	5.57E1
Kr-85m	0.0
Kr-85	1.41E5
Kr-87	0.0
Kr-88	0.0
Xe-131m	9.06E3
Xe-133m	1.77E4
Xe-133	1.19E6
Xe-135m	0.0
Xe-135	2.41E2
Xe-138	0.0
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Water depth	21 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0

Table 9 Continued: Assumptions Used for FHA in the Fuel Handling Building Dose Analysis

Spent Fuel Pool Ventilation System Filter efficiency	
Elemental	95
Organic	95
Particulate	95
Isolation of release	No isolation assumed
Time to releases all activity (hours)	2
Time to start crediting post accident recirculation control room HVAC (seconds)	15
Time to start crediting pressurization HVAC mode (hours)	2

Table 10: FHA in Fuel Handling Building - Activity Released to Atmosphere

Nuclide	Amount of Activity (Curies)
I-131	1.439E1
I-132	0.0
I-133	9.471E-1
I-134	0.0
I-135	6.950E-4
Kr-85m	0.0
Kr-85	1.410E4
Kr-87	0.0
Kr-88	0.0
Xe-131m	4.530E2
Xe-133m	8.850E2
Xe-133	5.950E4
Xe-135m	0.0
Xe-135	1.205E1
Xe-138	0.0