

December 20, 2005

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation
McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370

License Amendment Request for Selective
Implementation of the Alternative Source Term and
Revision to Technical Specification 3.9.4,
Containment Penetrations.

Pursuant to 10 CFR 50.90, Duke Energy Corporation (Duke) is requesting an amendment to the McGuire Nuclear Station (McGuire) Facility Operating Licenses and Technical Specifications (TS). This amendment request proposes to revise the McGuire licensing basis by adopting the Alternative Source Term (AST) radiological analysis methodology as allowed by 10 CFR 50.67 for the fuel handling accidents. This amendment request represents selective implementation of the AST as described in NRC Regulatory Guide 1.183.

This amendment request will also revise TS 3.9.4, Refueling Operations, Containment Penetrations, and its associated Bases based on guidance contained within the NRC approved industry Technical Specifications Task Force (TSTF) Change Traveler TSTF-51A, Revision 2: "Revise containment requirements during handling irradiated fuel and core alterations," for a Westinghouse plant. The application of the AST methodology to the fuel handling accidents radiological analyses supports this Technical Specification revision.

A001

Duke is requesting this amendment to provide flexibility in scheduling outage tasks by relaxing the current overly conservative containment closure requirements.

As a result of the proposed change to AST methodology, coupled with the guidance provided by TSTF-51A, Technical Specification 3.9.4, Containment Penetrations, would be amended to revise the applicability of the specification to only apply during movement of recently irradiated fuel. McGuire is committing to develop administrative controls to adequately close containment penetrations if necessary during refueling operations.

The contents of this amendment request are as follows:

Attachment 1 provides a marked copy of the affected Technical Specification and Bases showing the proposed changes.

Attachment 2 provides reprinted pages of the affected Technical Specification and Bases with the proposed changes incorporated.

Attachment 3 provides a description of the proposed changes and the technical justification.

Attachment 4, pursuant to 10 CFR 50.92, provides the determination that this LAR contains No Significant Hazards Consideration.

Attachment 5, pursuant to 10 CFR 51.22, provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

This License Amendment Request (LAR) is similar to LARs submitted by Catawba and Oconee Nuclear Stations. Selective implementation of AST was approved for Catawba on April 23, 2002. Full implementation of AST was approved for Oconee on June 1, 2004 and for Catawba on September 30, 2005.

Implementation of this proposed amendment to the McGuire Technical Specifications will impact the McGuire Updated Final Safety Analysis Report (UFSAR). As a result, it will be necessary to revise various sections of the McGuire UFSAR in accordance with 10 CFR 50.71(e).

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this LAR has been reviewed and approved by the McGuire Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this LAR is being forwarded to the appropriate State of North Carolina official.

Duke is requesting NRC review and approval of this LAR by August 1, 2006 or as soon as practical. This LAR was simplified to facilitate an expedited NRC review in order to support the McGuire Unit 2 Refueling Outage in September 2006. Duke plans to submit a full scope AST LAR for McGuire in the near future. The NRC's standard 30 day implementation grace period will be adequate for this LAR.

Inquiries on this matter should be directed to Lee A. Hentz at 704-875-4187.

Sincerely,

A handwritten signature in black ink, appearing to read "Gary R. Peterson". The signature is fluid and cursive, with a long horizontal stroke at the end.

Gary R. Peterson

Attachments

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cc: w/attachments

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OATH AND AFFIRMATION

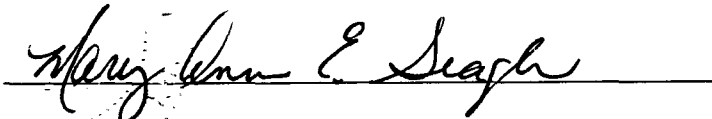
Gary R. Peterson affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.



Gary R. Peterson, Site Vice President

Subscribed and sworn to me: 12.20.05

Date



Notary Public

My commission expires: 2.26-07

Date

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bxc: w/attachments

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ATTACHMENT 1

MARKED PAGES OF AFFECTED TECHNICAL SPECIFICATION

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

- LCO 3.9.4 The containment penetrations shall be in the following status:
- a. The equipment hatch closed and held in place by a minimum of four bolts;
 - b. A minimum of one door in each air lock closed; and
 - c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
 - 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
 - 2. exhausting through an OPERABLE Containment Purge Exhaust System HEPA filter and charcoal adsorber.

APPLICABILITY: ~~During CORE ALTERATIONS,~~
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

recently

recently

A.1

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During ~~CORE ALTERATIONS~~ or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or exhausting through an OPERABLE containment purge exhaust HEPA filter and charcoal adsorber. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During ~~CORE ALTERATIONS~~ or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During

recently

50.67 (ref. 4)

recently

BASES

BACKGROUND (continued)

recently

~~CORE ALTERATIONS~~ or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

involving recently irradiated fuel

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge Supply and Exhaust is a subsystem of the Containment Purge and Ventilation System. Purge air is supplied to the Containment through two 50 percent capacity fans and their associated filters and heating coils. Purged air is exhausted through two 50 percent capacity fan and filter networks to the unit vent where it is monitored during release to the atmosphere. The purge air supply and exhaust fans and filters are located in the Auxiliary Building.

There are five purge air supply penetrations and four purge air exhaust penetrations in the Containment. These penetrations are in the upper compartment and lower compartment. Two normally closed isolation valves in each penetration provide Containment isolation.

The upper compartment purge exhaust ductwork is so arranged to draw exhaust air into a plenum around the periphery of the refueling canal, effecting a ventilation sweep of the canal during the refueling process. The lower compartment purge exhaust ductwork is arranged as to sweep the reactor well during the refueling process.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.

recently irradiated

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS~~ or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated

involving recently irradiated fuel

BASES

APPLICABLE SAFETY ANALYSES (continued)

Replace with
Insert A

fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 2), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

involving recently irradiated fuel

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for penetrations exhausting through an OPERABLE Containment Purge Exhaust System HEPA filter and charcoal adsorber.

APPLICABILITY

Replace with
Insert B

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

recently

INSERT A

TECH SPEC BASES 3.9.4

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with irradiated fuel minimum decay time of 72 hours, ensure that the release of fission product radioactivity, subsequent to the limiting fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 50.67 (Ref.4) and Regulatory Guide 1.183 (Ref.5).

INSERT B

TECH SPEC BASES 3.9.4

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. Recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. In Modes 1,2,3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In Modes 5 and 6, when movement of irradiated fuel assemblies is not being conducted, the potential for a fuel handling accident does not exist.

Additionally, due to radioactive decay, a fuel handling accident involving irradiated fuel that has not occupied part of a critical reactor core within the previous 72 hours will result in doses that are within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are exhausting through an OPERABLE Containment Purge Exhaust System HEPA filter and charcoal adsorber.

recently

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS~~ ~~OR~~ movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

significant

SR 3.9.4.2

involving recently irradiated fuel

This SR verifies that the required testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The Containment Purge Exhaust System filter tests are in accordance with Reference ~~4~~ ³. The VFTP includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. UFSAR, Section 15.7.4.
- ~~2. NUREG 0800, Section 15.7.4, Rev. 1, July 1981.~~
- ~~2~~ ² 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
- ~~3~~ ³ Regulatory Guide 1.52 (Rev. 2).
- ~~6~~ ⁶ PIP M-05-1608

4. 10 CFR 50.67, Accident Source Term.
5. Regulatory Guide 1.183, Rev. ~~0~~.

ATTACHMENT 2

REPRINTED PAGES OF AFFECTED TECHNICAL SPECIFICATION

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

- LCO 3.9.4 The containment penetrations shall be in the following status:
- a. The equipment hatch closed and held in place by a minimum of four bolts;
 - b. A minimum of one door in each air lock closed; and
 - c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
 - 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
 - 2. exhausting through an OPERABLE Containment Purge Exhaust System HEPA filter and charcoal adsorber.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or exhausting through an OPERABLE containment purge exhaust HEPA filter and charcoal adsorber. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 50.67 (Ref. 4). Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During

BASES

BACKGROUND (continued)

movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident involving recently irradiated fuel during refueling.

The Containment Purge Supply and Exhaust is a subsystem of the Containment Purge and Ventilation System. Purge air is supplied to the Containment through two 50 percent capacity fans and their associated filters and heating coils. Purged air is exhausted through two 50 percent capacity fan and filter networks to the unit vent where it is monitored during release to the atmosphere. The purge air supply and exhaust fans and filters are located in the Auxiliary Building.

There are five purge air supply penetrations and four purge air exhaust penetrations in the Containment. These penetrations are in the upper compartment and lower compartment. Two normally closed isolation valves in each penetration provide Containment isolation.

The upper compartment purge exhaust ductwork is so arranged to draw exhaust air into a plenum around the periphery of the refueling canal, effecting a ventilation sweep of the canal during the refueling process. The lower compartment purge exhaust ductwork is arranged as to sweep the reactor well during the refueling process.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during recently irradiated fuel movements.

**APPLICABLE
SAFETY ANALYSES**

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction

BASES

APPLICABLE SAFETY ANALYSES (continued)

with irradiated fuel minimum decay time of 72 hours, ensure that the release of fission product radioactivity, subsequent to the limiting fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 50.67 (Ref. 4) and Regulatory Guide 1.183 (Ref. 5)

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO limits the consequences of a fuel handling accident involving recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for penetrations exhausting through an OPERABLE Containment Purge Exhaust System HEPA filter and charcoal adsorber.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. Recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. In Modes 1,2,3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In Modes 5 and 6, when movement of irradiated fuel assemblies is not being conducted, the potential for a fuel handling accident does not exist.

Additionally, due to radioactive decay, a fuel handling accident involving irradiated fuel that has not occupied part of a critical reactor core within the previous 72 hours will result in doses that are within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

BASES

**SURVEILLANCE
REQUIREMENTS**

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are exhausting through an OPERABLE Containment Purge Exhaust System HEPA filter and charcoal adsorber.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident involving recently irradiated fuel that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment.

SR 3.9.4.2

This SR verifies that the required testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The Containment Purge Exhaust System filter tests are in accordance with Reference 3. The VFTP includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. UFSAR, Section 15.7.4.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. Regulatory Guide 1.52 (Rev. 2).
4. 10 CFR 50.67, Accident Source Term.
5. Regulatory Guide 1.183, Rev. 0.
6. PIP M-05-1608

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

DESCRIPTION OF PROPOSED CHANGES

SUMMARY OF TSTF-51A, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations"

Following reactor shutdown, decay of the short lived nuclides greatly reduces the fission product inventory present in irradiated fuel. After sufficient radioactive decay, the mitigation of a fuel handling accident primarily depends on water level rather than active ventilation or containment systems. Therefore, the Operability requirements of certain ESF Technical Specifications may be relaxed, if justified by radiological analysis.

CORE ALTERATIONS is defined in Technical Specifications as the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. As described in TSTF-51A, accidents postulated to occur during core alterations include inadvertent criticality, fuel handling accident, and the loading of a fuel assembly or control component in an incorrect location. Generically, it was concluded that of these off normal occurrences, only the fuel handling accident could result in cladding damage and possess the potential for radiological releases. Consequently, it is being proposed that the phrase "during CORE ALTERATIONS" be deleted from the applicable Technical Specifications.

These changes will allow increased flexibility during outages for moving personnel and equipment in and out of containment while not affecting Operability requirements. These changes only affect containment requirements during periods of relatively low shutdown risk.

Technical Specification 3.9.4, REFUELING OPERATIONS, Containment Penetrations

The APPLICABILITY is being revised to delete "During CORE ALTERATIONS" and add "recently" to irradiated fuel assemblies.

Required Action A.1, "suspend CORE ALTERATIONS," is being deleted.

Required Action A.2 is being revised to add "recently" to irradiated fuel assemblies and will be re-numbered.

The Bases for this Technical Specification is being revised per the guidance of TSTF-51A including the addition of the definition of "recently" irradiated fuel.

Containment closure will no longer be required for moving fuel that has not occupied a critical reactor core within the previous 72 hours ("non-recently" irradiated fuel). McGuire is committing to develop administrative controls to adequately close containment penetrations if necessary during refueling operations based on the following guidance:

During movement of non-recently irradiated fuel assemblies, ventilation system and radiation monitor availability (as defined by NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the reactor coolant system decays fairly rapidly. The goal of maintaining ventilation systems 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 exists. 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.

TECHNICAL JUSTIFICATION

Analyses of Radiological Consequences of the Fuel Handling
Accidents at McGuire Nuclear Station

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- Appendix C: McGuire Nuclear Station Site Meteorological Data
- Appendix D: Data for the Analysis of Radiological Consequences of Design Basis Fuel Handling Accidents and Weir Gate Drop at McGuire Nuclear Station

1 INTRODUCTION

With this License Amendment Request (LAR) Duke Energy Corporation (Duke) requests NRC approval of selective implementation of Alternative Source Terms for McGuire as defined in Regulatory Guide 1.183. Duke also requests NRC approval for the amendment of Technical Specification 3.9.4 pertaining to containment penetrations as described in TSTF-51A. Duke is requesting this amendment to provide flexibility in scheduling outage tasks by relaxing containment closure requirements during "non-recently" irradiated fuel movement.

In support of these requests, Duke has performed analyses of the radiological consequences of damaging a single fuel assembly both in Containment and in the Spent Fuel Pool (Design Basis Fuel Handling Accidents - FHAs), and damaging multiple fuel assemblies in the Weir Gate Drop Accident (WGD) in the Spent Fuel Pool (SFP). These analyses employ the method of Alternative Source Terms (AST) and conform to the main text and Appendix B of Regulatory Guide 1.183 (Reference 1).

Values for several parameters were derived to support these analyses, including the atmospheric dispersion factors for the transport of radioactivity to the control room outside air intakes (the control room χ/Q values). These χ/Q calculations conformed to Regulatory Guide 1.194 (Reference 2). Methods developed for similar analyses of the radiological consequences of design basis fuel handling accidents at Catawba and Oconee Nuclear Stations in support of similar Alternative Source Term implementation amendments (References 3 and 4) were used where practical and appropriate for McGuire Nuclear Station. For each design basis accident, total effective dose equivalent (TEDE) values were calculated for the following locations and time spans:

- 1) Exclusion Area Boundary (EAB) during the 2 hour time span of maximum releases.
- 2) Outer boundary of the Low Population Zone (LPZ) during the 30 day period after accident initiation.
- 3) Control room during the 30 day period after accident initiation.

Details of the FHA and WGD accident analyses are presented below. They include a comparison of the results using Regulatory Guide 1.183 and TEDE criteria against the results using current source term technology and the calculation of an equivalent TEDE.

2 OVERVIEW OF THE FUEL HANDLING ACCIDENTS

The analysis of the radiological consequences of design basis fuel handling accidents and a Spent Fuel Pool weir gate drop at McGuire Nuclear Station was performed using the Alternative Source Term methodology pursuant to 10 CFR 50.67 and Regulatory Guide 1.183 (Reference 1). Radiation doses (TEDE values) at the EAB, LPZ, and in the control room were calculated. The scenarios modeled include the FHA in containment, the FHA in the spent fuel pool area, and a SFP weir gate drop.

The events analyzed involve a cladding breach for all fuel pins in the population of affected fuel assemblies (either a single fuel assembly for the FHA or an array of 7 fuel assemblies for the weir gate drop). The single fuel assembly drop scenario is modeled to occur either in the SFP or in containment. Bounding source terms are used: the total amount of noble gases released from the breached fuel assemblies and all of the iodine that reach the pool surface are transported to the release points (unit vent and equipment hatch) and then to the environment.

A realistic, yet conservative, exponential exhaust model is used to release almost all of the radioactive source term for these design basis accidents in the first two hours. No credit is taken for containment integrity or for filtration by the Containment Purge Exhaust System (CPES) for the FHA in containment. For SFP building scenarios, no credit is taken for the Fuel Handling Ventilation Exhaust System (FHVES) filtration. In all scenarios, the activity released to the environment is not filtered by plant ventilation systems.

Additionally, no credit is taken for radioactive decay during the transit time after its release from the fuel to its arrival at a receptor. No credit is taken for filtration through the equipment or personnel hatches, deposition, plateout, or holdup.

For the FHA in containment, control room operator dose is investigated using release pathways from the equipment hatch or the unit vent (via the personnel airlock). The FHA and WGD in the SFP Building exhaust activity to the environment via the unit vent. It is then transported to the control room.

Radiation doses to the control room operators are calculated using site-specific atmospheric dispersion factors for single control room air intake locations. Control room doses are calculated using the methodology from Regulatory Guide 1.183 and dose conversion factors taken from Federal Guidance Reports No. 11 and 12 (References 1, 5 and 6).

3 ANALYTICAL METHOD AND COMPUTER CODES

The Bechtel proprietary computer code LOCADOSE (References 7 - 9) was used to model the transportation of radioactivity and to calculate off-site and control room radiation doses (TEDE values). This code utilizes a network of user defined nodes and flows to model the transport of activity. The generalized transport equations used in the codes are the same as those recently endorsed by the Staff (Reference 10). The time dependent Murphy-Campe equation is solved by LOCADOSE to calculate radioactivity buildup in the control room and the resulting radiation doses to the control room operators.

The LOCADOSE code requires that an initial isotopic inventory be entered into the source node. This source term inventory is produced external to the LOCADOSE code using the SCALE computer code system which was developed by the Oak Ridge National Laboratory (Reference 11). It provides fuel depletion models which were used to derive the inventory of fission products in a fuel assembly at reactor shutdown. This inventory comprises the source term entered into the LOCADOSE code. Radioactive decay before accident initiation is applied as part of the LOCADOSE transport model.

Control room atmospheric dispersion factors (χ/Q values) were calculated with the computer code ARCON96 (Reference 12). Individual control room χ/Q values were computed using the same method as that reported in support of

analysis of the AST design basis fuel handling accidents for Catawba Nuclear Station (Reference 3).

LOCADOSE, SCALE, and ARCON96 are standard tools used by Duke in the analysis of radiological consequences of design basis accidents.

4 RADIOLOGICAL SOURCE TERMS AND RELEASES

4.1 Source Term Development

Source terms used in the FHA analyses were based upon bounding fuel assembly depletion models. The isotopic inventory used for the FHA (single assembly) and the weir gate drop (7 fuel assemblies) accident is based on maximum activity over the full range of burnup. The fuel assembly isotopic inventory bounds an enrichment range of 3.5 to 5.0 wt% ²³⁵U and an assembly averaged burnup range up to 62,000 MWD/MTU. A constant radial peaking factor of 1.65 is applied over the full range of burnup. An allowance is also made for a thermal power uncertainty of 2%. The core design process checks ensure that the peaking modeled in the accident analyses bounds the projected power history of the cores being designed. This is the same basic (conservative) approach that is used in the analysis of radiological consequences of the design basis FHAs at Oconee (Reference 4).

"Recently irradiated" fuel refers to fuel assemblies that have occupied part of a critical reactor core within the previous 72 hours. As this is the earliest that fuel movement (and hence a fuel handling accident) could occur, the single assembly FHAs assume a decay time of 72 hours prior to accident initiation.

The WGD applies the same fuel assembly isotopics as the FHA, but scales them by the 7 fuel assemblies affected. 17.5 days of decay time (420 hours) was used (Reference 13). This time reflects the minimum post-shutdown time that must elapse prior to weir gate movement over the spent fuel. This decay time represents the earliest the weir gate could be dropped on fuel assemblies in the spent fuel pool.

The resulting source term inventory for a fuel assembly involved in a design basis fuel handling or weir gate drop accident is provided in Appendix A.

4.2 Iodine Species Composition

In accordance with Regulatory Guide 1.183, the iodine released from the fuel into the surrounding water (in either the reactor vessel or the spent fuel pool) was assumed to be composed of 95% of cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine compounds. The CsI released from the fuel is assumed to dissociate in the water, and the iodine instantaneously re-evolves as elemental iodine due to the low pH of the water. Therefore, all iodine released from the water is assumed to take the form of elemental iodine (57%) and organic iodine compounds (43%). In addition, the effective decontamination factor for all forms of iodine released from the water was set to 200. These assumptions are consistent with the values given in Regulatory Guide 1.183 (Reference 1). Values for iodine transport parameters are listed in Appendix D.

4.3 Spent Fuel Pool and Containment Release Models

The source term is released from the gaps of the affected fuel assemblies into the surrounding water. The particulates (alkali metals and some iodines) are retained by the water and the remaining iodines undergo scrubbing as they pass through the water. The noble gases pass through the water and are released from the water surface without mitigation. Thus, the atmosphere above the water where the event occurs contains noble gases and iodines. This mixture is released to the environment.

The time dependant activity release from the space is modeled as an exponential release. For the purpose of conservatism, it is desirable to release as much of the activity as quickly as possible. Since the EAB dose is computed over the first two hours, it is desirable to release as much of the activity as practicable over that time frame. The exponential release model will ensure that a great deal of the activity is released in the initial phases of the accident prior to the start of control room pressurization. This serves to maximize control room doses.

This model results in the release of almost all of the activity over the first two hours. Since it is an exponential release model, the highest rate of discharge occurs at the beginning of the release with a continual decrease in the activity release rate. The high initial release rate is conservative for control room doses since

unfiltered in-leakage is also higher at this time. This model is more conservative than a linear release model.

This modeling is very similar to that used for the FHA AST submittals for Catawba (Reference 3) and Oconee (Reference 4).

5 CONTROL ROOM TRANSPORT AND ATMOSPHERIC DISPERSION

Atmospheric dispersion factors are used to model the transport and dispersion of the released radioactivity from the release point to the receptor (off-site doses) or ventilation intake (control room doses) location. Off-site doses were calculated with the existing atmospheric dispersion factors for the EAB and LPZ. These are presented in Appendix D.

Atmospheric dispersion factors (χ/Q values) for the transport of radioactivity to the outside air intakes of the Control Room Area Ventilation System (CRAVS) at McGuire have been calculated. Plant data used to construct these models is presented in Appendix B. Historical meteorological data is provided in Appendix C. The computed control room χ/Q values are listed in Appendix D.

5.1 Control Room Atmospheric Dispersion Modeling

5.1.1 Control Room Area Ventilation System Design

The design of the CRAVS and its intakes form part of the basis for calculating control room χ/Q values.

The Control Room Area Ventilation System (CRAVS) at McGuire is equipped with four outside air intakes grouped into two pairs. Each pair is located near the corner of a reactor building and the outboard steam generator (S/G) doghouse. One pair is located at Unit 1 and the other pair located at Unit 2. The two CRAVS outside air intakes at each location are separated by only a few feet. Thus, McGuire is assumed to be a dual intake plant with each intake pair taken as one intake location.

Each intake is equipped with two separate and independent Class 1E motor operated isolation valves in series. The

isolation valves are open and fail "as is." The only controls for these valves are manual controls located in the control room; no automatic controls exist for any CRAVS outside air intake isolation valves. Each CRAVS outside air intake pair has an associated radiation monitor. However, these radiation monitors are non Class 1E. Therefore, McGuire is considered to possess "dual inlet design without manual or automatic selection controls" (Reference 14). In particular, no credit is taken for closure of any CRAVS outside air intake pair to reduce or eliminate the intake of radioactivity into the control room with outside airflow from the CRAVS Outside Air Pressurized Filter Trains.

The CRAVS outside air intakes are Seismic Category I. They are protected from turbine missiles by virtue of their location and are designed to withstand tornado wind loading. The ability of the CRAVS outside air intakes (in conjunction with the remainder of the CRAVS) to achieve and maintain a positive pressure in the control room is not degraded by tornado missiles. The NRC Staff reviewed and accepted the design of the CRAVS outside air intakes as a basis for classifying McGuire as a dual inlet plant (Reference 15). In conclusion, no failure mode in the McGuire license basis will cause the CRAVS outside air intakes to close.

5.1.2 Development of Control Room Atmospheric Dispersion Factors

The development of the control room atmospheric dispersion (χ/Q) values was performed by evaluating potential release points applicable to the three fuel handling accident scenarios, and modeling releases from these locations to the control room intakes. Two release points germane to these accidents were identified. One release point was associated with the spent fuel pool based accidents: the unit vent. Both of the release points were associated with releases in containment: the equipment hatch and the unit vent (from the personnel air lock or the Containment Purge Exhaust system).

Releases via the Containment Purge Exhaust system were considered. Like the releases from the personnel air lock, the Containment Purge Exhaust system would also discharge from the unit vent. Since the personnel air lock does not

possess filtration capability, this release path was considered to be more limiting than the Containment Purge Exhaust system. Both employ the same ultimate discharge point at the unit vent.

Control Room χ/Q values for radioactivity transport from each of the two potential release points for each nuclear unit to either CRAVS outside air intake were calculated using the computer code ARCON96. These calculations conformed to regulatory guidance and assumed ground releases (Reference 2).

Plant data for release point modeling is given in Appendix B. The meteorological data for McGuire used in the calculations are included on the CD in Appendix C. The control room χ/Q values for transport of activity from a release point to a Control Room Outside Air Intake for the fuel handling and weir gate drop accidents use the most restrictive combination. They are summarized below. For the period of 2-8 hours, the 0-8 calculated χ/Q is used. This is conservative compared to what would be computed for the 2-8 hour period using the 0-2 and 0-8 hour ARCON96 computed χ/Q s.

Table 5.1

Control Room Atmospheric Dispersion Factors (χ/Q , sec/m³)

Release Location	0-2 hour	0-8 hour	8-24 hour	1-4 day	4-30 day
Equipment Hatch	4.06E-03	3.57E-03	1.45E-03	1.14E-03	7.82E-04
Unit Vent	1.68E-03	1.47E-03	5.85E-04	4.54E-04	3.20E-04

Table 5.1 comprises the most conservative χ/Q s from the possible combinations of release and intake (receptor) locations. For the Equipment Hatch releases the worst case combination for each time increment is a release from the Unit 1 hatch to a Unit 1 intake. For the Unit Vent releases, the 0-2 hour χ/Q reflects a Unit 2 release to a Unit 2 intake. The rest of the Unit Vent release time intervals reflect a Unit 1 release to a Unit 1 intake.

5.2 Control Room Radioactivity Transport

Radioactivity can enter the control room either with outside airflow or as (unfiltered) in-leakage. The CRAVS at McGuire routes airflow from the control room outside air intakes, passing it through the Outside Air Pressurized Filter Trains to the Control Room Air Handling Units and into the control room. Each CRAVS filter train can start automatically following either a Safety Injection (SI) signal or a signal indicating undervoltage (loss of off-site power) conditions on the Class 1E 4160 volt switchgear to which it is aligned, or it may be started manually by the operators. Since automatic initiation is not predicted for the fuel handling accident scenarios, manual operator initiation is modeled. This action occurs within the first thirty minutes after the release of radioactivity from the damaged fuel.

The design basis performance characteristics of a CRAVS filter train for the current configuration of the system is presented in Appendix D.

The following assumptions and features related to the performance of the CRAVS during a FHA or WGD were included in the modeling.

- 1) A single outside intake pair is available for providing outside air into the control room. This intake is located in the contaminated air stream from the release point. All activity released is assumed to flow to this intake while undergoing atmospheric dispersion.
- 2) No CRAVS recirculation airflow is assumed. The McGuire CRAVS is provided with recirculation ductwork, but the recirculation dampers are currently closed and the system is operated in a once-through configuration, therefore recirculation is not modeled or credited. Modeling the CRAVS in once-through mode provides higher control room doses than would modeling CRAVS in recirculation mode.
- 3) The control room outside airflow rate is specified to be 2000 cfm \pm 10%. The minimum air flow rate was assumed in the model (1800 cfm). This rate reflects a single fan operating at the lower portion of its acceptable air flow range.

Duke studied the dependency of radioactivity levels in the control room and control room operator radiation doses on CRAVS outside and total airflow rates to the control room. These studies included mathematical analyses of solutions to the time dependent Murphy-Campe equation (Reference 16) and computer based calculations of radioactivity levels in the control room for several design basis accidents.

They demonstrated that for a "once-through" CRAVS, with no recirculation airflow, radioactivity levels (all fission products except noble gases) in the control room, and radiation doses to the control room operators, decrease with increasing CRAVS outside airflow rates. Therefore, control room radiation doses following a design basis FHA would decrease with increasing values of outside airflow rate to the control room.

The flow of noble gases into the control room (no filtration) increases the complexity of the dependence of radiation doses on CRAVS outside airflow rates. The studies showed that the control room TEDE values increase with decreasing outside CRAVS airflow rates.

Based upon these studies, it was concluded that conservative control room TEDE values would be calculated using a lower bound CRAVS outside airflow rate into the control room.

- 4) The rate of unfiltered in-leakage into the control room was set to 210 cfm. This value is based upon the error adjusted results from tracer gas testing. This value includes 10 cfm for use of control room access doors (leaving 200 cfm for rate of unfiltered in-leakage through the CRAVS and other parts of the control room envelope). The modeled unfiltered in-leakage rate exceeds the highest error adjusted unfiltered in-leakage obtained (177 cfm) during the tracer gas tests conducted in November 2003 (Reference 17).
- 5) Prior to the start of control room pressurization (at 30 minutes), with the CRAVS filter trains in standby, the rate of unfiltered in-leakage is set to 625 cfm. This value is based upon control room in-leakage testing which determined that the error adjusted

in-leakage to the MNS control room was 520 cfm. The assumed value also includes 10 cfm to account for control room ingress and egress.

- 6) The CRAVS filter efficiencies were modeled as 98.05% for elemental and organic iodine, and 99% for particulate. Although Reference 18 reflects the current licensing basis for McGuire, based upon recent interactions with the NRC staff during the review of the recently approved Catawba AST amendment, some features of Reference 19 were adopted in deriving these efficiencies. Specifically, a safety factor of two was applied to the Ventilation Filter Test Program (VFTP) specified penetrations (Reference 20) for elemental and organic iodines. In addition, the maximum permitted filter bypass in the VFTP for CRAVS was also explicitly included in the computation of these values.

6 DOSE COEFFICIENTS

This submittal reports the calculation of TEDE values at the EAB, LPZ, and in the control room following the design basis fuel handling and weir gate drop accidents. The dose coefficients used in the analyses conform to Regulatory Guide 1.183 (Reference 1). In particular, the coefficients for Committed Dose Equivalents (CDE values) and Committed Effective Dose Equivalents (CEDE values) for inhalation were taken from Federal Guidance Report 11 (Reference 5). The coefficients for deep dose equivalents (DDE values) were taken from Federal Guidance Report 12 (Reference 6).

7 ANALYSIS AND RESULTS

Radiation doses have been calculated for design basis fuel handling and weir gate drop accidents. No credit was taken for mitigation or filtration by any system except for CRAVS.

7.1 Design Basis FHA Scenarios

The fuel handling accident begins with the drop of a spent fuel assembly into either the reactor cavity or the spent fuel pool. All rods in the dropped fuel assembly are assumed to be damaged releasing all of their gap activity to the surrounding water and subsequently to either the containment or the spent fuel pool building atmosphere. The initiating event is assumed to occur 72 hours after shutdown (based upon the definition of "recently irradiated").

The weir gate drop accident begins with the drop of a weir gate into the spent fuel pool. The dropped weir gate damages 7 spent fuel assemblies. The gap inventories from all of the pins in the 7 impacted fuel assemblies are released into the spent fuel pool water. Due to restrictions (Reference 13) on the movement of the weir gate, the damaged fuel assemblies are credited with 17.5 days (420 hours) of decay prior the movement of the weir gate and their potential impact on the spent fuel in the SFP.

As analyzed in support of this LAR, the accident sequence consists of the following events:

- 1) Drop of a fuel assembly into either the reactor cavity or spent fuel pool, or drop of a weir gate into the spent fuel pool.
- 2) Release of radioactivity from the reactor cavity or spent fuel pool water (with an overall effective pool decontamination factor of 200).
- 3) Transport with dispersion to the Exclusion Area Boundary (EAB), boundary of the Low Population Zone (LPZ), and the control room outside air intakes.
- 4) Transport to the control room with unfiltered in-leakage and filtered outside airflow into the control room.
- 5) The control room operators manually start the CRAVS to pressurize the control room thirty minutes after accident initiation. This action will be directed by procedures. Upon demand, only one intake and one train of CRAVS respond. This action changes the rate of unfiltered in-leakage to the control room.

No credit is taken for the mitigation of off-site radiation doses by any plant system. Credit is taken for the Class 1E CRAVS to mitigate radiation doses to the control room.

The parameters associated with the design basis fuel handling accident and weir gate and their design basis values are listed in Appendix D.

7.2 Post Accident Off-site Radiation Doses

The off-site TEDE values for the design basis fuel handling accidents and weir gate drop are presented below.

**Table 7.1
Off-Site Dose Results for Fuel Handling and
Weir Gate Drop Accidents Utilizing AST**

Design Basis Accident	EAB TEDE (Rem)	LPZ TEDE (Rem)	Limit (Rem)
Fuel Handling Accident in Containment	2.9	0.26	6.3
Fuel Handling Accident in the Spent Fuel Pool	2.9	0.26	6.3
Weir Gate Drop	5.4	0.49	6.3

The TEDE values for the EAB and LPZ doses listed above are within the acceptance criteria.

The corresponding EAB whole body and thyroid radiation doses are reported in the UFSAR for these design basis accidents. Corresponding equivalent TEDE values for these accidents can be computed by applying a 3% weighting factor to the thyroid dose and adding it to the whole body dose (as described in footnote 7 to Reference 1). This provides a basis for comparison with the TEDE values reported above. These results and comparison are provided in the table below.

Table 7.2
Comparison of Equivalent EAB TEDE from Classical Source
Term Analyses and EAB TEDE Dose from AST Analyses (Rem)

Design Basis Accident	Whole Body	Thyroid	Equivalent Classical TEDE	AST TEDE
FHA in Containment	0.80	27.0	1.6	2.9
FHA in the Fuel Building	0.80	9.1	1.1	2.9
Weir Gate Drop	0.91	19.0	1.5	5.4

Some differences in the classical and AST analyses modeling details exist. In the classical analyses reported in the UFSAR, credit was taken for the Containment Purge Exhaust System and the Fuel Handling Ventilation Exhaust System; no credit was taken for these systems in the AST analyses. This change more than offsets the benefits derived from use of the lower gap fractions in Regulatory Guide 1.183 relative to Regulatory Guide 1.25.

7.3 Post Accident Control Room Radiation Doses

Control room TEDE values for the design basis fuel handling and weir gate drop accidents are presented below.

Table 7.3
Control Room Dose Results for Fuel Handling and
Weir Gate Drop Accidents Utilizing AST

Design Basis Accident	Control Room TEDE (Rem)	Limit (Rem)
FHA in Containment via Equipment Hatch	3.4	5
FHA in Containment via Personnel Air Locks/ FHA in SFP	1.4	5
Weir Gate Drop	2.8	5

No control room radiation doses are reported for these accidents in the McGuire UFSAR. Therefore, equivalent TEDE values in the control room for these design basis accidents

are not estimated for comparison with the control room TEDE values reported above. The TEDE values for the control room doses listed above are within the acceptance criteria.

8 SUMMARY OF ANALYSES

Radiation doses have been calculated for the design basis fuel handling and weir gate drop accidents postulated to occur at McGuire Nuclear Station. The alternative source term methodology was used in these design basis analyses. The radioactive source terms were developed and the analyses conducted pursuant to Regulatory Guide 1.183.

The transport and release of radioactivity to the environment for the design basis fuel handling accidents and weir gate drop were modeled incorporating the following assumptions:

- 72 hours of decay were credited in the single assembly fuel handling accident analyses based upon the definition of "recently irradiated" fuel.
- 17.5 days of decay were credited in the multiple assembly weir gate drop accident analysis based upon the minimum permitted time to move a weir gate.
- No filtration credit was taken for either the Containment Purge Exhaust System or the Fuel Handling Ventilation Exhaust System.
- The retention of iodine in either the reactor cavity water or the spent fuel pool water was modeled. For both accident locations the effective decontamination factor was set to 200 (which was comprised of an organic iodine DF of 1 and a elemental iodine DF of 350).

New atmospheric dispersion factors (χ/Q values) for transport of radioactivity to the CRAVS outside air intakes were calculated. The method for this calculation conforms to current regulatory guidance.

Radiological consequences were calculated for the following scenarios:

- 1) Fuel Handling Accident in containment with the equipment hatch open (releases through the open equipment hatch).
- 2) Fuel Handling Accident in containment with the personnel air locks open (releases from the unit vent stack).
- 3) Fuel Handling Accident in the spent fuel pool (releases from the unit vent stack).
- 4) Weir Gate Drop in the spent fuel pool (releases from the unit vent stack).

Off-site and control room radiation doses were calculated for the scenarios listed above in Tables 7.1 and 7.3. Based on the study reported above, the CRAVS filter train airflow rate to the control room was set to its lower bound value. For the calculation of control room radiation doses following a design basis fuel handling accident or weir gate drop, manual start of one CRAVS filter train at thirty minutes after the initiating event was credited.

For all scenarios, the off-site (TEDE values at the EAB and LPZ) and control room operator dose consequences (TEDE values) were shown to be within their acceptance criteria.

9 REFERENCES

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Appendix A

Radioactive Source Term for the Design Basis Fuel Handling Accidents and Weir Gate Drop at McGuire Nuclear Station

**Isotopic Inventory for a Single Fuel Assembly Damaged in a
Fuel Handling Accident or Weir Gate Drop Accident**

Halogens	Activity (Ci)		Noble Gases	Activity (Ci)		Alkali Metals	Activity (Ci)
Br83	1.31E+05		Kr83m	1.32E+05		Rb86	2.54E+03
Br85	2.99E+05		Kr85m	2.98E+05		Rb88	8.89E+05
Br87	4.95E+05		Kr85	7.48E+03		Rb89	1.18E+06
I130	3.95E+04		Kr87	6.15E+05		Rb90	1.12E+06
I131	8.09E+05		Kr88	8.69E+05		Cs134	2.06E+05
I132	1.18E+06		Kr89	1.12E+06		Cs136	5.92E+04
I133	1.67E+06		Xe131m	1.24E+04		Cs137	9.23E+04
I134	1.95E+06		Xe133m	5.20E+04		Cs138	1.66E+06
I135	1.60E+06		Xe133	1.65E+06		Cs139	1.58E+06
			Xe135m	3.62E+05			
			Xe135	4.12E+05			
			Xe137	1.55E+06			
			Xe138	1.59E+06			

Notes:

- The isotopics for the fuel assembly described above are produced at reactor shutdown by the SCALE computer code. No radioactive decay is modeled in the inventory above. Credited radioactive decay is applied to this isotopic inventory for the particular scenario as part of the transport model by the LOCADOSE computer code.
- This depletion model included thermal power uncertainty and a constant peaking factor of 1.65. It bounds the full range of permitted burnup.

Appendix B

Site Specific Data for the Calculation of Control Room Atmospheric Dispersion Factors Applicable to Fuel Handling and Weir Gate Drop Accidents at McGuire Nuclear Station

**Site Specific Data for the Calculation of Control Room Atmospheric Dispersion Factors
At McGuire Nuclear Station**

Parameter ¹	Unit Vent	Equipment Hatch ²	Equipment Hatch ³
Source Type	Vertical Point Source	Horizontal or Capped Point	Vertical Area Source
Release Height (m)	40.2	8.3	0
Flow Rate (m ³ /sec)	8.6	0	0
Sigma-Y (m)	0	0	1
Sigma-Z (ms)	0	0	1
Building Cross Section Area (m ²)	1588	1588	1588
Source / Stack Radius (m)	0	0	0
Vertical Velocity (meters/sec)	0	0	0
Distance (m), Direction (°)			
U1 Release to U1 CR OAI	43, 62		36, 32 (arc)
U1 Release to U2 CR OAI	94, 299	116, 298	
U2 Release to U1 CR OAI	94, 83	137, 76	
U2 Release to U2 CR OAI	43, 323		61, 10 (arc)

Notes:

¹Abbreviations are as follows: U = Unit, CR = Control Room, OAI = outside air intake (outside air intake pair or outside air intake location).

²Release location set as a horizontal point source for transport of fission products from the Unit 1 release location to the Unit 2 CRAVS outside air intakes and from the Unit 2 release location to the Unit 1 CRAVS outside air intakes.

³Release location set as a vertical area source for transport of fission products from the Unit 1 release location to the Unit 1 CRAVS outside air intakes and from the Unit 2 release location to the Unit 2 CRAVS outside air intakes.

Appendix C

McGuire Nuclear Station Site Meteorological Data

The data is contained on compact disc.

Appendix D

Data for the Analysis of Radiological Consequences of Design Basis Fuel Handling Accidents and Weir Gate Drop at McGuire Nuclear Station

**Data for the Analysis of Radiological Consequences of
Design Basis Fuel Handling Accidents and Weir Gate Drop at
McGuire Nuclear Station**

<u>Source Parameters</u>	<u>Value</u>
Radioactive Source Term Inventory	Appendix A
Peaking Factor	1.65
Gap release fractions	
I-131	8%
Kr-85	10%
Other Noble Gases	5%
Other Iodines	5%
Alkali Metals	12%
<u>Transport Parameters</u>	
Spent Fuel Pool and Reactor Cavity Water Level (ft)	23
Effective Decontamination Factor	200
Elemental Iodine DF	350
Organic Iodine DF	1
Chemical Composition of Iodine Released from the Water	
Elemental	57%
Organic	43%
<u>Receptor Parameters</u>	
Off-site Breathing Rates (m ³ /sec)	
0 hr - 8 hr	3.5E-4
8 hr - 24 hr	1.8E-4
24 hr - 720 hr	2.3E-4
Control Room Occupancy Factors	
0 hr - 24 hr	1.0
24 hr - 96 hr	0.6
96 hr - 720 hr	0.4
Control Room Breathing Rate (m ³ /sec)	3.5E-4

<u>Control Room Parameters</u>	<u>Value</u>
Control room volume (ft ³)	1.07E+05
Rate of unfiltered in-leakage to the control room (cfm)	
With the CRAVS OAPFT in standby	625
With the CRAVS OAPFT in operation	210
CRAVS outside airflow rate - single train (cfm)	2000 ± 10%
CRAVS Filter Efficiencies (incl. bypass, %)	
Elemental	98.05
Organic	98.05
Particulate	99

Atmospheric Dispersion Factors

EAB χ/Q (sec/m ³)	9.00E-04
LPZ χ/Q_s (sec/m ³)	
0 - 8 hr	8.00E-05
8 - 24 hr	5.20E-06
24 - 96 hr	1.70E-06
96 - 720 hr	3.70E-07
Control Room χ/Q_s	
Equipment Hatch release (sec/m ³)	
0 - 2 hr	4.06E-03
0 - 8 hr (used for 2 - 8 hr period)	3.57E-03
8 - 24 hr	1.45E-03
24 - 96 hr	1.14E-03
24 - 720 hr	7.82E-04
Unit Vent release	
0 - 2 hr	1.68E-03
0 - 8 hr (used for 2 - 8 hr period)	1.47E-03
8 - 24 hr	5.85E-04
24 - 96 hr	4.54E-04
24 - 720 hr	3.20E-04

Dose Conversion Factors

Inhaled CEDE Coefficients	FGR-11
DDE Coefficients	FGR-12

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate this Duke License Amendment Request (LAR) does not involve a significant hazards consideration.

This LAR proposes to revise the McGuire licensing basis by adopting the Alternative Source Term (AST) radiological analysis methodology as allowed by 10 CFR 50.67 for the fuel handling accidents. This LAR will also revise Technical Specification 3.9.4, Refueling Operations, Containment Penetrations, and its associated Bases based on guidance contained within the NRC approved industry Technical Specifications Task Force (TSTF) Change Traveler TSTF-51A, Revision 2: "Revise containment requirements during handling irradiated fuel and core alterations," for a Westinghouse plant. The analyses of radiological consequences of the fuel handling accidents with the AST methodology support this Technical Specification revision.

Conformance of this LAR to the standards for a determination of no significant hazards, as defined in 10 CFR 50.92, is shown in the following:

1. Does this LAR involve a significant increase in the probability or consequences of an accident previously evaluated ?

No. AST is an updated methodology used to evaluate the dose consequences of the fuel handling accidents (FHAs). It has been demonstrated that the dose consequences of the re-analyzed accidents remain within the dose limits of 10 CFR 50.67 and Regulatory Guide 1.183. For the FHAs, this is contingent upon the irradiated fuel decaying a minimum of 72 hours (non-recently irradiated fuel).

The proposed LAR would allow core alterations and movement of non-recently irradiated fuel within the Containment Building with the equipment hatch, personnel air locks, and other containment penetrations open. Operation of the Containment Purge Exhaust System (CPES) is not required during movement of non-recently irradiated fuel. The CPES is not needed or credited in the revised analysis of the FHAs.

The proposed Technical Specification revision modeled after TSTF-51A has also been previously reviewed and approved by the NRC and is supported by McGuire's revised radiological analyses utilizing AST.

The proposed revisions to the FHA radiological analyses and Technical Specification do not adversely affect accident initiators or precursors nor alter design assumptions. The proposed revisions do not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an accident. Therefore, the proposed revisions will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does this LAR create the possibility of a new or different kind of accident from any accident previously evaluated ?

No. The proposed revisions do not involve an addition or modification to any plant system, structure, or component. AST is an updated methodology that was used to re-evaluate the dose consequences of the McGuire UFSAR previously analyzed accidents.

This LAR would increase the time during which the equipment hatch and personnel air locks could be open during movement of non-recently irradiated fuel as allowed by the dose analysis. Having these penetrations open does not create the possibility of a new or different accident.

This LAR also removes the operability requirements for the CPES during movement of non-recently irradiated fuel as allowed by TSTF-51A. It does not alter the operation of this system beyond its functional capabilities. Modifying the Technical Specification operability requirements of this system does not create the possibility of a new or different accident.

Therefore, no new or different accidents will be created by revising this Technical Specification per TSTF-51A or changing to the AST methodology per 10 CFR 50.67.

3. Does this LAR involve a significant reduction in a margin of safety ?

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed re-analysis of the FHA dose consequences will have no affect on the performance of these barriers.

The proposed LAR would allow movement of non-recently irradiated fuel within the Containment Building with the equipment hatch, personnel air locks, and other containment penetrations open. The re-analysis of the FHAs using AST with these penetrations open remains within the dose limits specified by 10 CFR 50.67.

Therefore, the proposed LAR will not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the preceding analysis, it can be concluded that this LAR does not involve a significant hazards consideration as defined in 10 CFR 50.92.

ATTACHMENT 5

ENVIRONMENTAL ASSESSMENT / IMPACT STATEMENT

This McGuire License Amendment Request which adopts the Alternative Source Term radiological analysis methodology per 10 CFR 50.67 has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations.

This LAR does not involve a significant hazards consideration, increase the types and amounts of effluents that may be released offsite, or result in a significant increase of individual or cumulative occupational radiation exposures. The revised radiation dose consequences remain well within the limits of 10 CFR 50.67 and Regulatory Guide 1.183.

Therefore, this McGuire License Amendment Request meets the criteria provided by 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.