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March 20, 2008

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

ATTENTION: Document Control Desk

Subject: Duke Power Company LLC d/b/a
Duke Energy Carolinas, LLC
McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370

License Amendment Request for Full Scope Implementation of the Alternative Source Term

Pursuant to 10 CFR 50.90, Duke Energy Carolinas (Duke) is requesting an amendment to the McGuire Nuclear Station (McGuire) Facility Operating Licenses. 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, Accident Source Term, for the Loss of Coolant Accident. This amendment request represents full scope implementation of the AST as described in NRC Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Revision 0. Selective implementation of AST for the McGuire Fuel Handling Accidents was approved by the NRC on December 22, 2006. There are no changes proposed to the McGuire Technical Specifications within this amendment request.

The application of the AST methodology to the Loss of Coolant Accident (LOCA) radiological analysis will allow McGuire to resolve the Control Room envelope degraded boundary condition as discussed in McGuire's response to NRC Generic Letter 2003-01, Control Room Habitability, dated February 19, 2004.

By separate amendment request dated January 22, 2008, Duke proposed to revise the McGuire Technical Specification (TS) requirements related to Control Room envelope habitability in TS 3.7.9, Control Room Area Ventilation System. The proposed changes are consistent with the Industry and NRC approved

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Technical Specification Task Force (TSTF) change TSTF-448, Control Room Habitability, Revision 3 and the NRC Consolidated Line Item Improvement Process (CLIIP).

Duke has performed a review of all McGuire License Amendment Requests (LAR) currently under review by the NRC for impacts to this AST LAR. None of these LARs impact any assumptions or results of the LOCA AST radiological analysis.

The contents of this amendment request are as follows:

Attachment 1 provides Duke's evaluation of the LAR which contains a description of the proposed changes, background information, the LOCA AST radiological technical analysis, the determination that this LAR contains No Significant Hazards Considerations, an applicable regulatory requirements evaluation, the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement, and the appendices.

This LAR is similar to LARs submitted by Oconee, Catawba, and San Onofre Nuclear Stations. Full Scope Implementation of AST was approved for Oconee on June 1, 2004, for Catawba on September 30, 2005, and for San Onofre on December 29, 2006.

Implementation of this proposed LAR to the McGuire licensing basis 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 North Carolina State officials.

Duke is requesting NRC review and approval of this LAR by April, 2009. Duke is also requesting a 60 day implementation grace period due to the extensive document changes necessary to implement this LAR.

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

Sincerely,

Bruce Hamilton

Bruce H. Hamilton

Attachment

cc: w/attachments

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OATH AND AFFIRMATION

Bruce H. Hamilton 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.

Bruce Hamilton

Bruce H. Hamilton, Site Vice President

Subscribed and sworn to me: March 20, 2008

Date

Lori C Gibby / Lori C. Gibby

Notary Public

My commission expires: July 1, 2012

Date

bxc: w/attachments

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ATTACHMENT 1

EVALUATION OF PROPOSED AMENDMENT
TABLE OF CONTENTS

ACRONYMS

AST	Alternative Source Term
BWST	Borated Water Storage Tank
CA	Auxiliary Feedwater System
CDE	Committed Dose Equivalent
CEDE	Committed Effective Dose Equivalent
CLB	Current Licensing Basis
CLIIP	Consolidated Line Item Improvement Process
CR	Control Room
DDE	Deep Dose Equivalent
DF	Decontamination Factor
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
FHA	Fuel Handling Accident
FWST	Refueling Water Storage Tank
GDC	General Design Criterion
GL	Generic Letter
KI	Potassium Iodide
L_a	Containment Leakage Rate
LAR	License Amendment Request
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
MHA	Maximum Hypothetical Accident (applicable to Oconee only)
NRC	Nuclear Regulatory Commission
NC	Reactor Coolant System
ND	Residual Heat Removal System
NS	Containment Spray System
ORNL	Oak Ridge National Laboratory
P_a	Peak Design Basis Loss of Coolant Accident Containment Pressure
RG	Regulatory Guide
SCALE	Standardized Computer Analyses for Licensing Evaluation
SER	Safety Evaluation Report
Sp	Containment High High Pressure Engineered Safeguards Signal
Ss	Safety Injection Engineered Safeguards Signal
SSF	Safe Shutdown Facility
TEDE	Total Effective Dose Equivalent
TS	Technical Specification
TSTF	Technical Specification Task Force
VA	Auxiliary Building Ventilation System
VC	Control Room Ventilation System
VE	Annulus Ventilation System
VFTP	Ventilation Filter Test Program
VP	Containment Purge System
VQ	Containment Air Release System
VX	Containment Air Return System

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Duke is requesting an amendment to the McGuire Facility Operating Licenses. This License Amendment Request (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, Alternate Source Term, for the Loss of Coolant Accident. This LAR represents full scope implementation of the AST as described in NRC Regulatory Guide 1.183 (Reference 1). Selective implementation of AST for the Fuel Handling Accidents was approved by the NRC on December 22, 2006 (Reference 2).

2.0 PROPOSED CHANGES

As stated above, this LAR proposes to revise the McGuire licensing basis by adopting the AST radiological analysis methodology Loss of Coolant Accident (LOCA). Per Regulatory Guide 1.183, full scope implementation requires as a minimum the re-analysis of the LOCA. New applications of AST would not require prior NRC approval unless stipulated by a 10 CFR 50.59 evaluation or involved a change to a Technical Specification.

Implementation of this LAR will require changes to the McGuire UFSAR Chapter 15 control room and off-site radiological consequence analyses for the LOCA. Following approval of this amendment request by the NRC, McGuire will provide the revised UFSAR sections to the NRC as part of the normal UFSAR update as required by 10 CFR 50.71(e).

This LAR does not propose any changes to the McGuire Technical Specifications.

3.0 BACKGROUND

McGuire is the last of the three Duke nuclear sites to request full scope implementation of AST. In performing this work, McGuire desired to adhere to the main text and Appendix A of Regulatory Guide 1.183 and to use the experience and features of the past Duke AST LARs as much as possible. Methods and features developed for the approved AST models of Duke's other two nuclear sites (Catawba and Oconee) were used, where applicable, to minimize the uniqueness or originality of the McGuire models. Because McGuire and Catawba are both Westinghouse four loop plants with ice condenser containments, a great deal of similarity between these plants could be presumed. However there are some specific differences in the response of the plant systems to a LOCA. Those differences are modeled and discussed where applicable in the technical analysis section.

The application of the AST methodology to the LOCA radiological analysis will allow McGuire to resolve the control room degraded boundary condition as discussed in

McGuire's response to NRC Generic Letter (GL) 2003-01 dated February 19, 2004 (Reference 3).

As requested by GL 2003-01, tracer gas testing performed at McGuire in October 2003 revealed that unfiltered in-leakage into the control room exceeded the limit in the current design basis accident analyses. As a result, an operability evaluation was performed. To maintain the control room Operable but Degraded, a new post LOCA radiation dose calculation was performed to include the contribution of potassium iodide (KI) usage by the control room operators to maintain dose within the regulatory limits. The revised calculation determined that KI use will maintain the operator dose within limits up to an unfiltered in-leakage of approximately 610 standard cubic feet per minute (scfm). The highest unfiltered in-leakage value measured at McGuire was approximately 177 scfm.

To completely resolve the control room degraded boundary condition, McGuire is submitting this LAR pursuant to 10 CFR 50.67 to incorporate AST methodology into the post LOCA dose consequence calculation and licensing basis. Adoption of the AST methodology will allow inclusion of testing based unfiltered in-leakage value of approximately 210 scfm (after control room pressurization) without exceeding the regulatory dose limits for the operators and without the need to credit KI.

By separate amendment request dated January 22, 2008, Duke proposed revising the McGuire Technical Specification (TS) requirements related to Control Room envelope habitability in TS 3.7.9, Control Room Area Ventilation System. The proposed changes are consistent with the Industry and NRC approved Technical Specification Task Force (TSTF) change TSTF-448, Control Room Habitability, Revision 3, and the NRC Consolidated Line Item Improvement Process (CLIIP). These changes include periodic testing of control room unfiltered in-leakage and a periodic habitability assessment.

Duke has also performed a review of all McGuire LARs currently under review by the NRC for impacts to this AST LAR. None of these LARs impact any assumptions or results of the LOCA AST radiological analysis.

4.0 TECHNICAL ANALYSIS

4.1 INTRODUCTION

With this License Amendment Request (LAR), Duke requests Nuclear Regulatory Commission (NRC) approval of full scope implementation of Alternative Source Terms (AST) for McGuire Nuclear Station (McGuire) as defined in Section 1.2 of Regulatory Guide (RG) 1.183 (Reference 1). Duke previously submitted an LAR for selective implementation of AST at McGuire which was reviewed and approved in Reference 2. This LAR requests that McGuire be permitted to completely replace the "classical" or TID-14844 based source term in its licensing basis with the alternative source term of RG 1.183.

Duke has performed an analysis of the radiological consequences of a McGuire Large Break Loss of Coolant Accident (LOCA) using Alternative Source Term methodology and computed the impact to off-site and control room doses. In keeping with Duke's "fleet approach" to nuclear standardization, the McGuire LOCA model has adopted the same or similar positions as the Catawba Nuclear Station (Catawba) model, where technically feasible. Duke has strived to maintain strict compliance with RG 1.183. The adherence of the McGuire LOCA analysis to RG 1.183 is summarized in Appendix C, including the one exception taken which is discussed in Section 4.6.2 and an alternative iodine partitioning methodology (from the Catawba submittal) employed which is discussed in Sections 4.5.4 - 4.5.6 of this LAR.

In response to NRC concerns stated in the McGuire Fuel Handling Accident (FHA) AST Safety Evaluation Report (SER) (Reference 2) related to the data used to compute the control room atmospheric dispersion factors, Duke has re-computed these factors using 5 years of data from the meteorological tower at its current location. The re-computed values are included in Appendix B.

The consequences for the design basis LOCA were computed in terms of total effective dose equivalent (TEDE) for the following locations and time spans:

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

The details of the analysis of the radiological consequences from a LOCA at McGuire using the RG 1.183 methodology are presented below. This represents a reanalysis of the McGuire LOCA accident and will replace the current analysis which is based on the "classical" source term technology upon approval and implementation of this amendment.

No physical plant modifications are required by the new analysis or to implement AST at McGuire. Implementation of AST will permit McGuire to retire a non-conforming issue and associated current operability evaluation related to measured control room unfiltered in-leakage and its impact on the radiological consequences to control room operators from a design basis LOCA.

4.2 COMPUTER CODES

The McGuire LOCA analysis is performed using the same methods and codes as the analyses supporting Duke's other nuclear plants (Catawba and Oconee, References 7 and 8), both of which have been approved for full scope implementation of AST. The LOCA analysis is complex due to the number of plant systems modeled. Many of these systems and some specific portions of the model require separate codes and analyses whose results are imported into the radiological consequences code. The radiological analysis is performed by the LOCADOSE code (Reference 9). This is the controlling code for the analysis and derives many of its inputs from the output of other codes. These inputs include source term isotopics and system response modeling. LOCADOSE provides the integrated, time-dependent response model used to transport activity within the plant and to receptors. It computes radiological consequences from effluent releases.

The main computer codes used are discussed below. All of the codes and methodologies described have been used to support the Oconee Nuclear Station (Oconee) Maximum Hypothetical Accident (MHA) and Catawba LOCA analyses and their corresponding approved AST submittals. New computer codes are not used in the McGuire LOCA analysis.

4.2.1 LOCADOSE

LOCADOSE is a Bechtel proprietary code (Reference 9) which is used by others in the commercial nuclear industry for similar application. It is the primary radiological effluent analysis code used by Duke to analyze design basis accidents, including those supporting the AST LARs, for all of Duke's nuclear plants. LOCADOSE was used to compute the off-site and control room doses for the McGuire AST Fuel Handling Accident (Reference 10) and for the Catawba AST LOCA and FHA analyses (References 6 and 11). It is a nodal code. Activity is transferred between the nodes to simulate the progression of the accident and the transport of activity within the plant and to the environment and receptors. This code uses plant systems and component performance taken from the codes described below in this section to compute TEDE doses off-site (EAB and LPZ) and in the control room.

4.2.2 SCALE

The Standardized Computer Analyses for Licensing Evaluation (SCALE) code system (version 4.4) was produced by Oak Ridge National Laboratory (ORNL) for the NRC (Reference 12). It is a compilation of a large number of individual codes which perform a variety of functions using sequences or modules that act as interfaces to simplify the input. While SCALE can perform criticality and shielding analyses (among other functions), it is used for source term depletion and decay through the SAS2H module (using the ORIGEN-S code) in radiological analyses.

SCALE is used to model fuel assembly depletion over the fuel cycle and to produce fuel assembly isotopics which are then used to model the activity in the core available for release as a result of the accident. These isotopics are produced using inputs which were chosen to bound expected fuel designs, core designs, and core operational parameters to produce a conservative source term. The isotopics produced by SCALE are used to derive the source term input to the LOCADOSE model.

SCALE was used to generate the source terms used in the Catawba FHA and LOCA submittals (References 6 and 11) and in the McGuire AST FHA submittal (Reference 10). McGuire and Catawba share the same source term analysis.

4.2.3 ARCON96

ARCON96 (Reference 13) is used to compute control room habitability atmospheric dispersion factors (χ/Q) at Duke. It produces dispersion factors between potential release locations and the control room air intakes. Adjustments can be made to these factors after they have been produced by ARCON96 (post processing) but prior to input into the LOCADOSE code to reflect scenario specific modeling (typically related to the Control Room Ventilation System modeling).

4.2.4 CANVENT

The CANVENT computer code is Duke proprietary. It is used to analyze post LOCA conditions in the secondary containment (annulus) and to model the response and continued operation of the Annulus Ventilation System to the LOCA. CANVENT computations utilize a mass balance in the annulus given post LOCA containment and reactor building leakage and Annulus Ventilation System exhaust airflow rates. It calculates the energy balance in the annulus using the energy in the annulus air, leakage entering the annulus, airflow leaving the annulus, heat transfer (convective and radiant) to the annulus and reactor building, and thermal and pressure induced expansion of the containment shell into the annulus. The program calculates the time dependent annulus pressure response and Annulus Ventilation System exhaust and recirculation airflow rates. This data is used to model the Annulus Ventilation System performance in LOCADOSE. This code was employed in the same manner for the Catawba AST LAR.

4.2.5 PHSC

PHSC was developed internally at Duke to model the post LOCA time-dependent sump pH profile. Its calculation methodology utilizes correlations and data for boric acid/sodium hydroxide solutions and other common acids and bases from EPRI reports NP-5561-CCML (Reference 14) and TR-105714 (Reference 15). It produces time-dependent sump pH profiles at the projected sump temperature and at the normalized temperature (25 °C). It has been benchmarked against test data.

The PHSC methodology for predicting the time-dependent response of containment sump pH in a post LOCA environment is based on NUREG/CR-5950 (Reference 5). It was originally developed to support the Oconee AST LAR (Reference 8) and was modified to incorporate of the effects of ice melt prior to being used to support the Catawba AST LOCA submittal (Reference 7).

This code and its methodology were discussed on the phone with the staff as Chemistry RAIs during the Catawba AST LAR review and documented in the second part of Reference 16.

4.2.6 IODEX

IODEX is a Duke proprietary code. It is based upon PHSC and is used to model the partitioning of Emergency Core Cooling System back-leakage to the Refueling Water Storage Tank (FWST). It calculates the water temperature and the buildup of iodine, sodium, and other solutes (nitrates, chlorides, and lithium) in the Refueling Water Storage Tank, given the solute concentration and water temperature in the containment sump and an ECCS back-leakage rate. It also calculates the pH in the tank, using the method of NUREG/CR-5950 (Reference 5) to determine the formation of volatile elemental iodine, the release of iodine to the environment, and the equivalent partition fractions. It takes into account the displacement of air from ECCS leakage into the FWST and diurnal expansion of the air in the tank. This model relies mainly on sump chemistry, sump temperature, and Auxiliary Building convective air transfer.

IODEX was used to model the partitioning of the ECCS releases to the Refueling Water Storage Tank for the Catawba LOCA LAR (Reference 6) and the same NUREG/CR-5950 (Reference 5) based methodology was used to model partitioning of Borated Water Storage Tank (BWST) releases for the Oconee AST MHA LAR submittal (Reference 17).

4.2.7 WASHOUT

WASHOUT is a Duke proprietary code. It is used to calculate spray lambdas to quantify the mitigation provided by the Containment Spray (NS) System during injection from the Refueling Water Storage Tank and during sump recirculation. The code models the removal of elemental and particulate iodine from the containment atmosphere. During recirculation it also considers the effects of radiolysis and the potential for iodine revolitilization. This code's methodology is consistent with that used for Oconee and Catawba.

Iodine washout (elemental and particulate) during the injection phase is modeled using the methodology from NUREG/CR-0009 (Reference 18). Iodine (particulate and elemental) removal from the containment atmosphere and the amount of elemental iodine retention in the sump solution during recirculation is modeled using the NUREG/CR-5950 (Reference 5) methodology. WASHOUT methodology is based on and is consistent with Standard Review Plan Section 6.5.2, *Containment Spray as a Fission Product Cleanup System* (Reference 19).

4.3 OVERVIEW OF ACCIDENT AND INTEGRATED RESPONSE

4.3.1 Design Basis Scenario

The plant's response to a Loss of Coolant Accident involves an integrated and coordinated response of individual systems and components. An analysis of the radiological consequences of the accident needs to include the individual system responses and their coordination into the integrated plant response.

The bounding design basis scenario includes a limiting single failure. Potential single failures of individual pieces of equipment and of whole trains of systems were postulated, including ventilation component failures, and mitigating system component failures to determine the bounding scenario. For McGuire, this scenario is referred to as the "Minimum Safeguards" scenario. In this scenario a power failure results in the loss of one train of the following LOCA mitigation systems:

- Containment Spray System (NS)
- Containment Air Return System (VX)
- Annulus Ventilation System (VE)
- Control Room Ventilation System (VC)

This failure results in the loss of one fan or pump or an entire train of equipment. Thus, the modeled plant response to the LOCA only credits one Containment Spray pump and train, one Containment Air Return fan, one Annulus Ventilation System fan and train, and one Control Room Ventilation fan and filter. In addition, maintenance is assumed in progress on one of the four Control Room Ventilation System inlet valves and that this maintenance has resulted in the blockage of this suction path, leaving one of the four intake paths unavailable. Thus, the Control Room Ventilation System is modeled in its minimum alignment for normal plant operation with two inlets at one intake location open and only one inlet at the other intake location open. Control Room Ventilation System configuration is controlled via several administrative means including Technical Specifications and Selected Licensee Commitments.

This scenario affects several systems and greatly reduces the ability of the plant to mitigate radioactivity transportation and release, and it increases the activity available to off-site and control room receptors. It also increases the time that the activity is resident in the control room. The loss in mitigation functions serves to increase the consequences of the accident. Since this scenario reduces the effectiveness of multiple mitigation systems and equipment, it bounds any postulated scenario which would reduce the effectiveness of only one of these systems.

In addition, several pieces of mitigation equipment were assumed to exhibit their minimum acceptable performance characteristics, including assuming the latest possible start time to provide additional conservatism. The following systems or equipment are modeled at minimal performance in the McGuire LOCA analysis:

- Containment Air Return (VX) System and Spray (NS) start times are based upon on the worst case (latest) diesel generator sequencer loading. This prolongs the delay until these systems start to mitigate the LOCA.
- NS and VX flows are reduced to reflect the operation of the diesel generator at its lowest (Technical Specification) allowed frequency (2% reduction) and voltage (10% reduction). This reduces pump and fan performance.
- Control Room Ventilation (VC) and Annulus Ventilation (VE) fans are modeled at minimal acceptable performance (10% flow reduction from nominal). The modeled start times bound the minimal acceptable performance times for these fans to establish pressure and vacuum conditions (respectively). Low VC fan flow produces higher control room doses than high fan flow for a once-through VC system like McGuire's. The amount of flow reduction for both systems' fans bounds the reduction associated with diesel generator under-frequency and under-voltage conditions.
- Filter efficiencies reflect a design margin factor of 2 to offset potential degradation between filter tests and explicitly include maximum Technical Specification and Ventilation Filter Test Program (VFTP) filter bypass flow.
- VC System inlet flow streams may not be perfectly balanced. 65% of the outside air flow is assumed to come from the contaminated stream. As will be discussed in Section 4.6.9, this value is based upon plant testing and bounds all system alignments for normal operation.

The sequence of events for this scenario is summarized in Table 1.

4.3.2 McGuire Integrated Response to a Loss of Coolant Accident / Sequence of Events

Initially, the reactor is at full power (plus 2% thermal power uncertainty) at the end of cycle. One inlet path at one intake location for the VC System has been removed from service (flow path blocked). The Containment Air Release (VQ) System is in service. The Reactor Coolant (NC) System activity level is the maximum allowed by Technical Specifications.

The event begins with a double-ended guillotine rupture of a large NC System pipe releasing reactor coolant to lower containment. Engineered Safeguards signals, including Safety Injection (Ss) and reactor trip, are modeled as generated nearly instantaneously with the initiation of the LOCA. This activates one train (other trains lost due to single failure) of the Emergency Core Cooling System (ECCS), Auxiliary Feedwater (CA) System, VE, Auxiliary Building Ventilation (VA), and VC Systems. It also starts containment isolation. The equipment affected by the single failure is rendered useless for the duration of the Minimum Safeguards LOCA scenario. The hi-hi containment pressure (Sp) signal is initiated almost simultaneously with the Safety Injection (Ss) signal and activates the NS System which starts spray flow within 2 minutes of accident initiation.

Table 1
Sequence of Events for LOCA with Minimum Safeguards and One VC Outside Air Inlet Valve Closed at One Inlet Location

Events	Time (sec)
LOCA initiation Reactor trip Safety injection signal Begin containment leakage Minimum Safeguards failure (at scenario initiation) One closed VC outside air inlet (initial condition) Containment Isolation begins Release reactor coolant water	0
VQ System isolated	4
Begin control room pressurization	11
Begin gap source release Control room pressurized	30
Start VE System (in exhaust mode)	39
Annulus vacuum established	71
Start NS System (no effect/credit until VX start)	120
Start VX System (NS System becomes effective/creditable)	600
Begin early in-vessel source release	1800
End gap source release	1830
Begin ND auxiliary spray from sump (start ECCS leakage)	3000
ECCS and NS System transferred to sump recirculation	3240
End early in-vessel source release	6480
Particulate Iodine DF>50 (reduce particulate spray lambda by factor of 10)	7100
Elemental Iodine DF>200 (cease elemental spray credit)	46,000
Stop NS system (cease spray credit) Reduce Containment leakage rate to half	86,400
End 30 day releases, end scenario	2,592,000

Based upon the worst case stroke times, the VQ system is isolated as part of containment isolation four seconds after the start of the accident, resulting in a negligible and insignificant effect on accident consequences. Pressurization air flow to the control room automatically begins within 11 seconds after the accident and pressurization is accomplished by 30 seconds. VC fans are assumed to be performing at their minimum acceptable flow rate. NS provides spray flow to the upper containment compartment within two minutes after the start of the accident, but it is not effective until the VX system starts ten minutes after the rupture to mix the upper and lower containment atmospheres. The start times for the NS and VX systems bound the worst case (latest) start time based upon the diesel generator sequencer loading. The flow rates for these systems reflect the diesel generator operating at the minimum frequency and voltage as allowed by Technical Specification SR 3.8.1.2.

Radioactivity is not released from the core for the first 30 seconds (Reference 1) so any potential release or release path, prior to containment isolation, available during this first 30 seconds prior to gap release is insignificant. Gap activity release begins at this point and continues at a constant rate until it is completed 1830 seconds after accident initiation (Reference 1). At 1800 seconds, the fuel begins to release fission products (early in-vessel release) at a constant rate until the assumed fractions of core inventory are released completely at 1.8 hr after the initiating event (Reference 1). Fission products begin to accumulate in the containment atmosphere and in the sump during activity release. Releases from the containment sump to the environment, via ECCS, do not begin until the initiation of sump recirculation flow.

Figure 1 represents the transport model for the LOCA analysis.

4.3.3 Annulus Ventilation Response

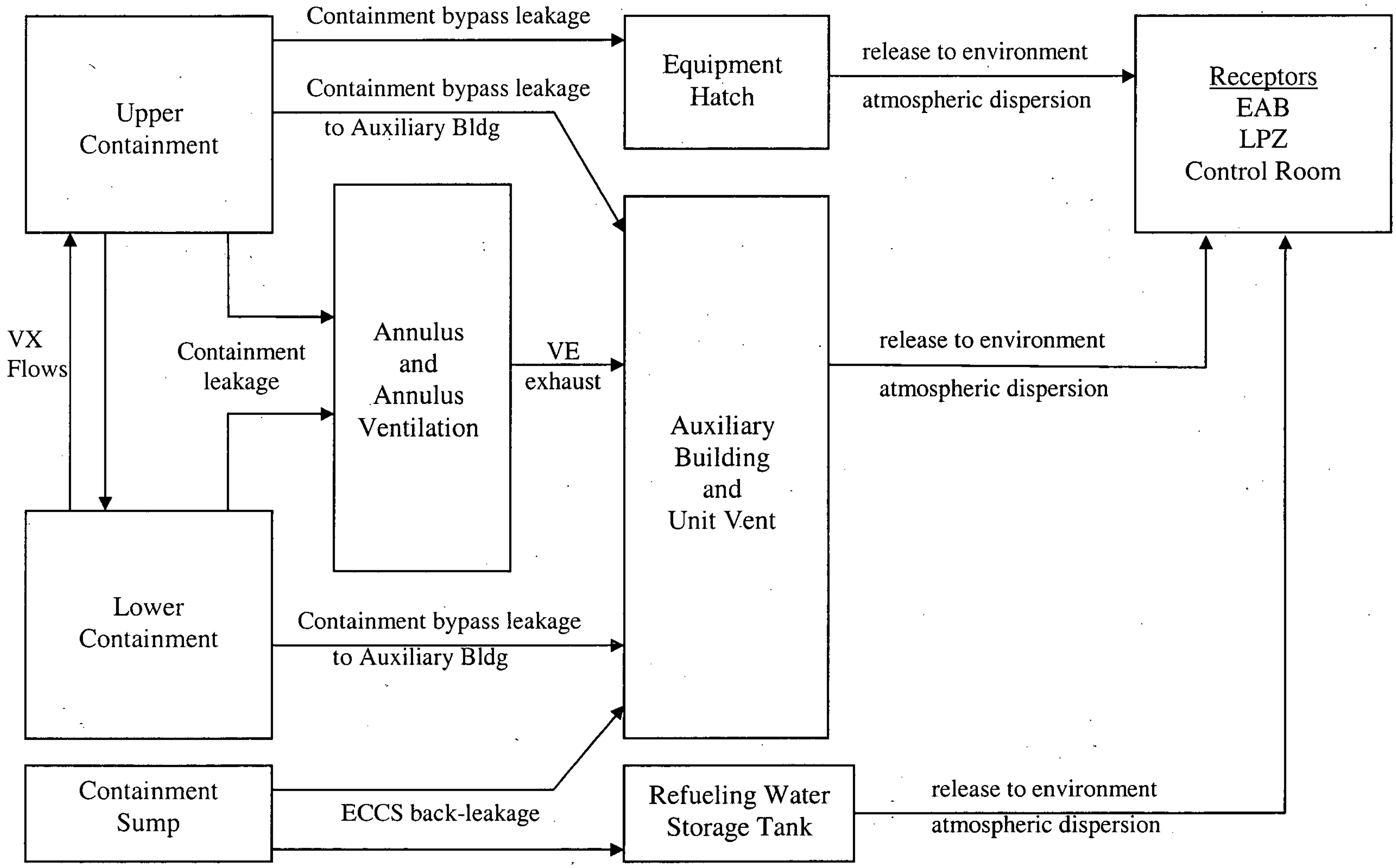
The VE System starts 39 seconds after the accident begins (based upon worst case diesel sequencer loading). It initially operates in exhaust mode to establish a vacuum in the annulus by discharging filtered annulus air to the environment. During this period, all containment leakage bypasses the VE System (100% containment bypass leakage). Once the annulus vacuum is established, 93% of the containment leakage rate is modeled to go into the annulus. It is filtered and exhausted to the environment through the unit vent stack until the VE System draws the annulus to the VE recirculation setpoint. The remaining 7% of the containment effluent bypasses the annulus. This is the maximum amount of bypass allowed per Technical Specification 5.5.2.

Radioactivity entrained with containment bypass leakage is postulated to escape to the environment through the following leak paths:

1) Equipment Hatch.
2) The Unit Vent Stack (via the Auxiliary Building, bypassing the VA filters).

Annulus pressure degrades from leakage into the annulus, primarily from the exterior environment. Once the annulus pressure reaches the VE exhaust setpoint, the system re-aligns to full exhaust mode. It then operates in the exhaust mode until the annulus pressure is lowered to the recirculation setpoint. The VE system exhaust airflow is routed to the unit vent stack.

Figure 1: Representation of Transport Model for LOCA Releases to Receptors



4.3.4 Emergency Core Cooling Systems Response

The ECCS responds in injection mode, taking suction from the Refueling Water Storage Tank (FWST). When that source of water is exhausted, the suction is swapped to the containment sump for recirculation (at 3240 seconds). ECCS leakage begins once sump recirculation flow begins with Auxiliary Spray initiation. For the Minimum Safeguards scenario this flow starts nearly an hour after the accident (at 3000 seconds).

Sump recirculation places contaminated water in the ECCS systems where leakage from pumps and valves in these systems could release activity to the Auxiliary Building atmosphere. This leakage will be filtered by the filtered exhaust portion of the VA System and discharged to the environment through the unit vent stack. Leakage from the ECCS could also be released into the FWST. Some of the iodine radioisotopes entrained in this leakage path could become volatile, enter the FWST airspace, and leak to the environment through the FWST vent. Only iodine radioisotopes are modeled to escape from ECCS leakage per Reference 1, however, iodine precursors are included in the sump source term to account for iodine produced by decay during the progression of the accident as they would be in the sump and available for release. Noble gases are not transported to the sump, as they are expected to remain in the containment atmosphere. The particulate isotopes (including iodine precursors) remain in particulate form and are retained in the sump water.

4.3.5 Containment Spray System Response

The Containment Spray System starts and provides spray flow within 2 minutes after accident initiation. This bounds the worst case diesel generator sequencer loading. NS provides spray flow to upper containment at a rate reduced from nominal based upon the possibility for the diesel generator to operate at a reduced frequency and voltage. Full spray flow coverage is provided to upper containment, and spray effectiveness is reduced for flow from nozzles which point toward containment walls.

Initially, spray provides no radiological benefit since the release is made to lower containment. This compartment is separated from upper containment by the operating deck which essentially denies communication between upper and lower containment without a motive force. This separation sets up conditions to promote the flow of the post LOCA lower containment atmosphere to upper containment through the ice condensers. With all of the activity initially in lower containment, spray into upper containment (which has no activity in its atmosphere) has no mitigation impact. Once the VX System starts, however, it forces air flow from upper containment to lower containment which promotes airflow in the opposite direction through the ice condensers. When activity is discharged into upper containment from the ice condensers, it can be mitigated by spray. No credit is taken for any potential natural air flow into the ice condensers prior to VX start due to the thermal conditions in lower containment. Therefore, the McGuire LOCA model does not credit spray flow removal until the start of the VX System. The VX System start time bounds the worst case diesel loading sequence.

Spray mitigation is modeled through the use of spray lambdas. Elemental iodine and particulates removals are modeled. Each has its own spray lambdas and time dependent model. Particulate spray credit is reduced by a factor of 10 when the iodine Decontamination Factor (DF) reaches 50 and elemental iodine spray credit is discontinued when its DF reaches 200 (Reference 19). Removal of particulates by spray is credited until 24 hours after accident initiation.

4.3.6 Control Room Ventilation System Response

McGuire was categorized as a dual intake plant during initial licensing (Reference 24). It has no manual or automatic selection controls. The VC system has two trains, each train has one intake location with two separate inlets for a total of four intakes. The intake locations (see Appendix A) are sufficiently separated to eliminate the possibility of both locations being exposed to a concentrated plume of activity (Reference 24).

In the Minimum Safeguards scenario, one VC fan is lost and air flow through one of the two inlet paths at one intake location is assumed to be restricted or closed off. The air flow rate from the two control room intake locations is not perfectly balanced, so the flow split between the inlets is based upon plant testing in the potential configurations. The flow split value modeled bounds all allowable normal operational configurations of intakes and operating fans. The majority of the intake flow is assumed to come from the contaminated stream.

4.3.7 Calculation of Doses

Releases are made continually to the environment after initiation of the postulated LOCA. Radiation doses to the EAB are calculated over a period of two hours following the initiating event. The two hour period for the calculation of EAB doses is selected to coincide with the maximum release of radioactivity to the environment. Radiation doses to the LPZ and the control room operators are calculated over a period of 30 days following the initiating event. The control room dose is computed assuming the period of worst release is concurrent with a period of the most unfavorable dispersion per Reference 27. These time periods of calculation are in accordance with Reference 1.

4.4 SOURCE TERM

Catawba and McGuire share the same core isotopics analysis. This analysis was produced to bound both sites. The LOCA core isotopics were derived using the SAS2H module of the SCALE 4.4 code package. This module is a standard industry tool for source term depletion analyses and has been used in support of other Duke submittals. SAS2H uses the XSDRNPM fuel and assembly neutronics models along with ORIGEN-S which performs the depletion, transmutation, and decay calculations (although no decay is credited in the isotopics generation analysis). The 44 group ENDF/B-V library was chosen for this problem. Both SAS2H and this library have received extensive use and validation for the generation of LWR fuel isotopics.

The isotopics analysis models were based upon a 17 x 17 fuel assembly with 0.374 inch diameter fuel pins. Conservatisms were built into the significant parameters of the analysis. Additionally, this analysis investigated a large number of combinations of fuel enrichments, batch loadings, and fuel burn-up to bound current and potential future core designs. Conservative input values are used to generate the LOCA isotopics. These input values envelope the associated source term parameters including fuel burn-up, burnable poisons, peaking factors, and heavy metal loading. Proposed core designs are reviewed as part of Duke's reload design process to ensure that the design basis accident radiological analyses remain bounding.

4.4.1 Burn-up

Fuel assembly burn-ups are varied in the LOCA cases up to and including 62 GWd/MTU. Therefore, the results of this LOCA analysis are valid for individual fuel assembly burn-up levels up to 62 GWd/MTU (for fuel assemblies to be discharged at the end of the cycle). Burn-up can also be examined on a core averaged basis. The LOCA cases bound core average burn-ups above 44 GWd/MTU, which is in excess of anticipated core designs which are expected to have a peak core averaged burn-up of about 40 GWd/MTU.

4.4.2 Enrichment

The cases upon which the LOCA source term was constructed bound enrichments from 3.5% to 5% ^{235}U. This band of enrichments bounds expected enrichments and includes the McGuire Technical Specification 3.7.15 limit of 5% enrichment. Therefore, this source term bounds current and projected enrichment requirements.

4.4.3 Peaking

The three batches of fuel are modeled with bounding peaking factors for each cycle. Realistically, the source term cannot exceed an overall power peaking of 1.0 (except for thermal power uncertainty, discussed in the next section), but the combination of the conservatively applied individual batch peaking factors results in a power level greater than nominal rated power. Thus, the source term is conservative from the standpoint of batch and overall peaking.

4.4.4 Thermal Power Uncertainty

A thermal power uncertainty of 2% is included in the derivation of the source term. In effect, this increases the "nominal" or expected power level of the core to 102% (3479 MW thermal). This factor bounds the uncertainties in the plant thermal power calorimetric calculation. This is a separate parameter from power peaking discussed in the preceding section.

4.4.5 Source Term Release Model

Release fractions and timings are in strict accordance with RG 1.183 Tables 2 and 3.

4.4.6 Gap Release to Reactor Building Atmosphere Phase (30 – 1830 seconds)

The release of activity in the fuel/clad gap begins 30 seconds after the accident initiates. All of the activity in the gap is modeled to be released over the next 30 minutes. Thus, the gap release begins at 30 seconds and ends at 1830 seconds. A linear rate is used to model the release of all of the available gap activity over this time frame. The total core isotopic source term activity for each of the three isotope groups released during this phase is multiplied by the 5% release fraction to compute the amount of activity released over this time period. This is then divided by the time duration of the release to derive the release rate of the isotopes from the gap.

Table 2 lists the nuclides released during the gap release phase and their release rates.

4.4.7 Early In-Vessel Release to Reactor Building Atmosphere Phase (0.5 – 1.8 hours)

The early in-vessel release phase models the relocation of core materials. It begins one-half hour after the onset of the accident and lasts for 1.3 hours. The release rate for the early in-vessel phase is computed in the same manner as the gap release rate. The release fractions from RG 1.183 Table 2 for this phase are applied to the core inventory to determine the amount of activity released during this period. This activity is divided by the duration of the release phase to derive the linear release rate.

Table 3 lists the nuclides modeled to be released during this phase and their release rate.

4.4.8 Overlap of Releases to Reactor Building Atmosphere (1800 – 1830 seconds)

From 1800 to 1830 seconds, both the gap and early in-vessel releases occur and there is overlap between them. Only the Halogens, Noble Gases, and Alkali Metals are released in the gap and early in-vessel phases, and so their release rates are the only ones affected during this time. The release rates for radionuclides in the other groups are the same as those from the early in-vessel phase (see Table 3). The release rates during this time are simply the sum of the gap (Table 2) and early in-vessel (Table 3) release rates.

Table 4 lists the release rates for the isotopes that are released in both the gap and early in-vessel phases.

4.4.9 ECCS Release Source Term

The ECCS release model uses the same release fractions and timing as the modeled release to the Reactor Building atmosphere. Since only the iodines are postulated to be released from ECCS leakage (Reference 1), only the iodines and the iodine precursors (Telluriums) are included in this source term. Addition of the precursors accounts for the production of iodine during the accident from radioactive decay in the sump. The release rates from the core for the Iodines and Telluriums are the same as those listed in Tables 2 through 4.

Table 2
Gap Release Phase Nuclides and Release Rates

Noble Gases	Release Rate (Ci/hr)	Halogens	Release Rate (Ci/hr)	Alkali Metals	Release Rate (Ci/hr)
Kr83m	1.56E+06	Br83	1.55E+06	Rb86	2.08E+04
Kr85m	3.40E+06	Br85	3.41E+06	Rb88	1.00E+07
Kr85	1.07E+05	Br87	5.56E+06	Rb89	1.33E+07
Kr87	6.96E+06	I130	2.96E+05	Rb90	1.25E+07
Kr88	9.79E+06	I131	1.04E+07	Cs134	2.09E+06
Kr89	1.25E+07	I132	1.52E+07	Cs136	5.60E+05
Xe131m	1.43E+05	I133	2.15E+07	Cs137	1.26E+06
Xe133m	6.72E+05	I134	2.47E+07	Cs138	2.09E+07
Xe133	2.08E+07	I135	2.06E+07	Cs139	1.96E+07
Xe135m	4.51E+06				
Xe135	6.65E+06				
Xe137	1.98E+07				
Xe-138	1.98E+07				

Table 3
Early In-Vessel Phase Nuclides and Release Rates

Noble Gases	Release Rate (Ci/hr)	Halogens	Release Rate (Ci/hr)	Alkali Metals	Release Rate (Ci/hr)	Barium, Strontium	Release Rate (Ci/hr)
Kr83m	1.14E+07	Br83	4.17E+06	Rb86	4.00E+04	Sr89	1.58E+06
Kr85m	2.48E+07	Br85	9.18E+06	Rb88	1.92E+07	Sr90	1.43E+05
Kr85	7.82E+05	Br87	1.50E+07	Rb89	2.56E+07	Sr91	2.55E+06
Kr87	5.09E+07	I130	7.97E+05	Rb90	2.40E+07	Sr92	2.60E+06
Kr88	7.15E+07	I131	2.80E+07	Cs134	4.02E+06	Sr93	2.82E+06
Kr89	9.13E+07	I132	4.09E+07	Cs136	1.08E+06	Ba139	3.08E+06
Xe131m	1.05E+06	I133	5.79E+07	Cs137	2.42E+06	Ba140	2.89E+06
Xe133m	4.91E+06	I134	6.65E+07	Cs138	4.02E+07	Ba141	2.80E+06
Xe133	1.52E+08	I135	5.55E+07	Cs139	3.77E+07		
Xe135m	3.30E+07						
Xe135	4.86E+07						
Xe137	1.45E+08						
Xe138	1.45E+08						

Table 3 (continued)
Early In-Vessel Phase Nuclides and Release Rates

Tellurium Group	Release Rate (Ci/hr)	Noble Metals	Release Rate (Ci/hr)	Cerium Group	Release Rate (Ci/hr)
Sb127	3.71E+05	Mo99	3.79E+05	Ce141	6.65E+04
Sb129	1.32E+06	Tc99m	3.35E+05	Ce143	6.88E+04
Te127m	6.08E+04	Tc101	3.38E+05	Ce144	5.08E+04
Te127	3.66E+05	Ru103	3.31E+05	Np237	1.63E-02
Te129	1.26E+06	Ru105	2.40E+05	Np238	1.94E+04
Te129m	2.55E+05	Ru106	1.23E+05	Np239	8.92E+05
Te131	3.34E+06	Rh103m	3.31E+05	Pu236	2.82E-02
Te132	5.73E+06	Rh105	2.15E+05	Pu238	1.65E+02
Te133	4.69E+06	Pd109	8.98E+04	Pu239	1.44E+01
Te133m	3.88E+06			Pu240	1.98E+01
Te134	8.15E+06			Pu241	5.58E+03
				Pu242	1.14E-01
				Pu243	2.16E+04

Table 3 (continued)
Early In-Vessel Phase Nuclides and Release Rates

Lanthanides	Release Rate (Ci/hr)	Lanthanides	Release Rate (Ci/hr)
Y90	1.49E+03	Pm148	2.89E+03
Y91	2.06E+04	Pm148m	4.55E+02
Y91m	1.50E+04	Pm149	1.02E+04
Y92	2.32E+04	Pm151	3.35E+03
Y93	1.89E+04	Sm153	8.82E+03
Y94	2.92E+04	Eu154	1.52E+02
Y95	2.98E+04	Eu155	5.94E+01
Zr95	2.74E+04	Eu156	4.88E+03
Zr97	2.74E+04	Pr143	2.40E+04
Nb95	2.75E+04	Pr144	2.05E+04
Nb95m	3.05E+02	Pr144m	2.86E+02
Nb97	2.74E+04	Am241	2.69E+00
La140	3.05E+04	Am242m	1.75E-01
La141	2.78E+04	Am242	1.38E+03
La142	2.80E+04	Am243	6.78E-01
La143	2.75E+04	Cm242	7.89E+02
Nd147	1.07E+04	Cm244	1.45E+02
Pm147	2.69E+03		

Table 4
Release Rates for Affected Nuclides During Overlap of the Gap and Early In-Vessel Phases

Noble Gases	Release Rate (Ci/hr)	Halogens	Release Rate (Ci/hr)	Alkali Metals	Release Rate (Ci/hr)
Kr83m	1.30E+07	Br83	5.72E+06	Rb86	6.08E+04
Kr85m	2.82E+07	Br85	1.26E+07	Rb88	2.92E+07
Kr85	8.89E+05	Br87	2.05E+07	Rb89	3.89E+07
Kr87	5.78E+07	I130	1.09E+06	Rb90	3.65E+07
Kr88	8.13E+07	I131	3.84E+07	Cs134	6.11E+06
Kr89	1.04E+08	I132	5.61E+07	Cs136	1.64E+06
Xe131m	1.19E+06	I133	7.94E+07	Cs137	3.68E+06
Xe133m	5.58E+06	I134	9.12E+07	Cs138	6.11E+07
Xe133	1.73E+08	I135	7.61E+07	Cs139	5.73E+07
Xe135m	3.75E+07				
Xe135	5.52E+07				
Xe137	1.64E+08				
Xe138	1.64E+08				

4.4.10 Iodine Specie Fractions and Sump pH

Iodine species are released to the Reactor Building atmosphere and to the sump using the fractions specified in RG 1.183. The iodine released to the Reactor Building atmosphere is 95% particulate (cesium iodide), 4.85% elemental, and 0.15% organic. The iodine in the sump is modeled to be 97% elemental and 3% organic (no particulates). RG 1.183 specifies that adoption of the iodine specie fractions modeled for the Reactor Building atmosphere is permitted if the sump pH is controlled at values of 7 or greater. The analysis of the McGuire post LOCA sump pH performance supports the adoption of the Regulatory Guide 1.183 values.

The McGuire sump pH analysis uses the same methodology (PHSC code discussed in Section 4.2.5) as used for the Catawba sump pH analysis. These are separate cases of the same analysis. The pH analysis incorporates elements of the NUREG/CR-5950 method and accounts for ice melt. The acids and bases tracked by this code include:

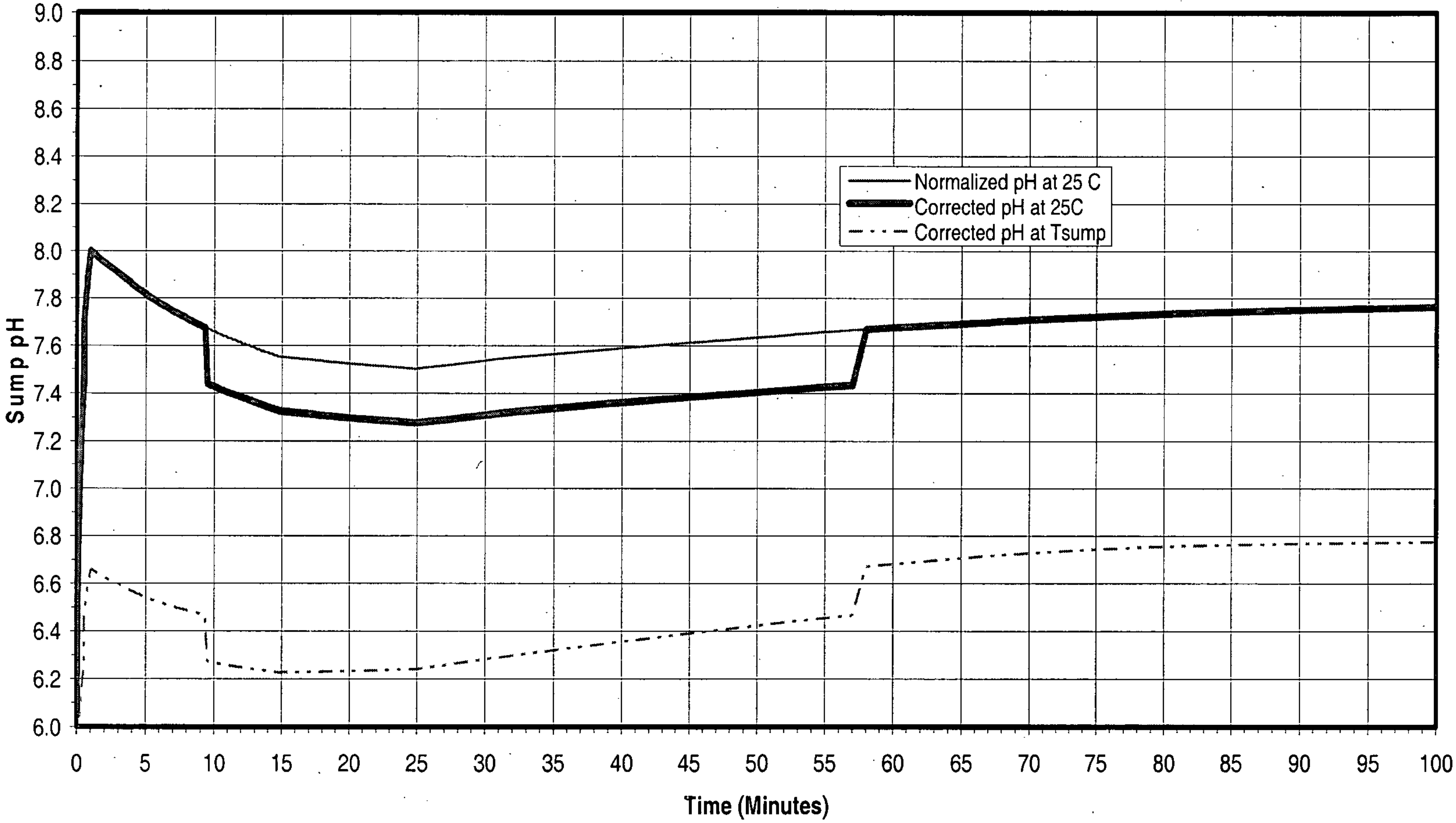
- Borated water from the Refueling Water Storage Tank (via the Containment Spray System and other components of the Emergency Core Cooling System, 2875 ppmB)
- Borated water from the cold leg accumulators (2875 ppmB)
- Borated water from the reactor coolant system (NC) inventory released as a result of the breach in the primary system (2875 ppmB)
- Borated water from the ice melt draining from the ice condensers (1800 ppmB).
- Nitric acid from the irradiation of moisture and air in containment (7.3E-09 mol/g H_2O – MRad)
- Hydrochloric acid generated from the radiolysis of electrical cable insulation (4.6E-04 mol/lbm insulation – MRad)
- Sodium ions released from the disassociation of borax during ice melt which is carried to the sump via drainage from the ice condensers (equivalent NaOH concentration, determined from the boron concentration, 0.33%)
- Lithium hydroxide which is added to the reactor coolant system for pH control during operation and released as a result of the breach in the primary system (Li concentration 7.7 ppm)

Figure 2 shows a minimum sump pH (referenced to the standard temperature of 25 °C) during the transient post LOCA period of 7.3 with an eventual equilibrium pH of 7.8.

Figure 2 also shows the sump pH profile for both sump temperature and normalized conditions. It also shows the application of a 3% correction between 9.5 and 57 minutes. This is a conservative adjustment made to account for differences between the PHSC code and its benchmark. The PHSC program was seen to produce higher pH values than the benchmark titration curves for boron concentrations greater than 3000 ppmB and sodium concentrations less than 578 ppm. The limiting difference was found to be 3%, so the results are conservatively reduced by this amount when these boron or sodium conditions exist (in this case it is due to the sodium condition only). The analysis was performed for 3000 minutes, however, after the first 100 minutes the results are fairly constant, so only the first 100 minutes are shown. Further explanation of the methodology and the plots produced from the analysis can be found in the documentation of phone conversation responses to chemistry related Catawba AST LAR RAIs in the second part of Reference 16.

Thus, the sump pH profile supports the use of the RG1.183 iodine specie model for the Reactor Building.

Figure 2: McGuire LOCA Sump pH Response



4.5 ACTIVITY TRANSPORT AND RELEASE

The activity released to the containment atmosphere is ultimately released to the environment through leakage which is captured by the annulus and processed prior to release, or through leakage that is released directly to the environment from containment, bypassing the annulus. The activity accumulated in the containment sump is released to the environment by leakage from the ECCS system to the Auxiliary Building and the FWST while in sump recirculation.

4.5.1 Containment Bypass Leakage

Leakage from containment that is not captured in the annulus and subjected to processing by the Annulus Ventilation System is referred to as "bypass" leakage. Bypass leakage releases are not mitigated by plant systems or filters prior to release to the environment (although activity released to the environment will be filtered by the VC System prior to entering the control room atmosphere). A vacuum is established in the annulus by the Annulus Ventilation System in its response to the accident. This vacuum creates a pressure differential with adjacent spaces which promotes air flow into the annulus where it can be filtered prior to release to the environment.

In the absence of a vacuum in the annulus, it is conservatively assumed that all containment leakage is bypass leakage. Thus, at the beginning of the accident, before the VE system has established a vacuum throughout the annulus, all containment leakage is assumed to bypass it. Since the LOCA occurs in lower containment, there is no activity in upper containment at the start of the accident. All activity associated with the bypass leakage initially comes from lower containment where it leaks into the Auxiliary Building and is subsequently released from the unit vent.

The total containment leakage rate (L_a) is set by Technical Specification 5.5.2 to be 0.3% of the containment air weight per day at the peak containment internal pressure for a LOCA, P_a (14.8 psig). L_a was calculated for the containment leakage testing program to be 140,000 sccm (standard cubic centimeters per minute) at standard conditions (14.7 psia and 68 °F). The air mass in containment, at P_a, was calculated to be 149,000 lbm, and the air mass in the ice condenser region was calculated to be 30,000 lbm. These regions are at different thermodynamic conditions, but applying the appropriate densities yields a L_a at P_a of 2.50 cfm. Once annulus vacuum is established, 7% of containment leakage is modeled as bypass leakage and the rest of the leakage is processed through the annulus and the Annulus Ventilation System. 7% is the maximum bypass portion of containment leakage permitted by Technical Specification SR 3.6.3.8.

There are several potential bypass leakage release points: the reactor building equipment hatch, the personnel access hatches, through systems which penetrate into containment from the Auxiliary Building and the Ventilation Purge System inlet. Each of these will be evaluated as a potential release point. The release model encompasses a conservative application of the potential release locations.

The equipment hatch separates the internal containment area from the exterior environment and is only opened during outages to move equipment into and out of the reactor building. It is periodically tested with a procedural acceptance limit of 500 sccm (0.009 cfm at P_a). Because the equipment hatch atmospheric dispersion factors are the worst of the potential upper containment release locations, it is assumed that this is the preferential leakage location associated with this compartment.

There are two personnel access hatches: one in lower containment and one in upper containment. Both are assumed to leak to the Auxiliary Building at their Technical Specification 5.5.2 limit of 0.01 L_a at P_a. Since this leakage flows into the Auxiliary Building, it is modeled as a release from the unit vent by the Auxiliary Building Ventilation System (VA). However, no credit is taken for VA filtration of this flow path as it is not in the post LOCA VA filtered flow path alignment.

The atmospheric dispersion factors associated with releases from the unit vent (to the control room) are greater than those associated with any postulated leakage from the VP intakes. Releases from the unit vent bound those from the VP intakes. Thus, it is assumed that the balance of the bypass leakage releases associated with upper and lower containment is made from the unit vent. No leakage is expected or modeled from VP valves. Even if the potential for a release from VP exists, it would be bounded by the unit vent dispersion factors.

To summarize, prior to the establishment of annulus vacuum, lower containment leakage bypasses the annulus and activity is released from the unit vent. Once annulus vacuum is established, 7% of containment leakage becomes bypass leakage which is modeled as upper containment flow from the equipment hatch at the testing acceptance criteria rate (500 sccm) with the balance of the bypass leakage (from both compartments) released from the unit vent. In accordance with RG 1.183, L_a is halved 24 hours after accident initiation.

4.5.2 Containment Leakage to the Annulus

After annulus vacuum has been established by the Annulus Ventilation System (VE), 93% of containment leakage (L_a at P_a) is assumed to flow into the annulus. The VE system filters the annulus atmosphere in both recirculation and exhaust modes. In recirculation mode, the system filters the annulus atmosphere and returns it to the annulus. In exhaust mode the system filters the annulus atmosphere as it discharges it to the environment. Thus, the majority of the leakage, which flows to the annulus after the establishment of vacuum, is filtered by the VE system prior to discharge. This mitigates the activity release to the environment. The VE system discharges from the unit vent.

4.5.3 Distribution of Containment Leakage Between Upper and Lower Containment

The distribution of containment leakage between the upper and lower compartments could be apportioned in a number of ways including use of their volumes or use of the number of penetrations in each compartment. Duke has constructed a conservative, technically based model utilizing the distribution of the areas of the process piping and ducting for those penetrations in the containment leakage monitoring program (option B type C). Numerically, many more penetrations are in lower containment, however, most of the larger ones (including the equipment hatch) are in upper containment. The population of penetrations was reduced to a reasonable and credible size by including only those with the most propensity to leak. This also includes those which are programmatically and periodically tested. The following criteria were applied in modeling the distribution of containment leakage:

- All electrical penetrations were assumed to be sufficiently sealed such that there is a reasonable expectation that they will not leak.
- Spare penetrations or blank flanged penetrations were also assumed to not leak.
- The fuel transfer tubes were not assumed to leak since they are flanged and have the spent fuel pool (water) on the opposite side.
- Instrument lines were assumed not to leak.

After the application of these criteria over 50 penetrations still remain in the population to be used in determining the leakage distribution. This population includes those penetrations which are in the testing program (option B, type C). The cross sectional area of the process lines in these penetrations was calculated for each compartment and the results ratioed to derive a conservative model of 60% of leakage from lower containment and 40% from upper containment. This model is more conservative than the volume based model, since the activity concentration in the lower compartment will be greater for the first portion of the accident, resulting in a greater activity release and larger doses.

The containment leakage model is summarized in Table 5. This table also provides the values for containment volume and the volumes of upper and lower containment. Ice condenser free volumes are included in the compartment volumes by assigning the portion of the ice condensers below the operating deck (inlet plenum) to lower containment the rest of the volume to upper containment. The annulus volume is provided during the discussion of the annulus model later.

Table 5
Containment and Bypass Leakage Model

Compartment and Modeled Volume (ft^3)	Containment and Bypass Leakage Rates from 0 to 24 hours[1]					Containment and Bypass Leakage Rates after 24 hours[1]				
	Total Leak Rate[2] (cfm)	Cont. Leak Rate to Annulus (cfm)	Total Bypass Leak Rate (cfm)	Bypass to Equip Hatch[3] (cfm)	Bypass to Unit Vent (cfm)	Total Leak Rate (cfm)	Cont. Leak Rate to Annulus (cfm)	Total Bypass Leak Rate (cfm)	Bypass to Equip Hatch (cfm)	Bypass to Unit Vent (cfm)
Upper Containment 826,752	1.00	0.930	0.070	0.009	0.061	0.50	0.465	0.035	0.0045	0.031
Lower Containment 370,623	1.50	1.395	0.105	N/A	0.105	0.75	0.698	0.053	N/A	0.053
Total Containment 1,197,375	2.50	2.325	0.175	0.009	0.166	1.25	1.163	0.088	0.0045	0.083

[1] Some minor inconsistencies may result from rounding. These differences are insignificant and in the conservative direction.
[2] All leakage is released from the unit vent as bypass leakage until annulus vacuum is established.
[3] Equipment Hatch leakage from upper containment is constant at 500 standard cc/min (0.009 cfm at P_a).

4.5.4 ECCS Back-leakage to the Auxiliary Building and FWST

The release of sump water activity to the environment is postulated to occur from two pathways: through the Auxiliary Building to the unit vent and from the Refueling Water Storage Tank (FWST). Initially, the spray system uses water from the FWST as a suction source. At 3000 seconds, system realignment to use the sump as a suction source begins. The realignment is projected to be accomplished in four minutes (240 seconds). This alignment takes water and activity from the sump and brings it outside of the Reactor Building so that it can be used as a suction source for the spray system. It is postulated that this water could leak into the Auxiliary Building or the FWST from pump or valve leakage.

One gpm of ECCS leakage to the Auxiliary Building and 20 gpm of leakage to the FWST are modeled. In accordance with Reference 1, these values are twice the amount that will be allowed by plant procedures. The area in the Auxiliary Building which houses components (pumps and valves) which could leak ECCS water to the Auxiliary Building is ventilated by the VA system. Air from this area is drawn into the filtered portion of the VA system which remains aligned to these spaces in a post LOCA situation. The FWST is vented directly to atmosphere through an open vent pipe. The same partitioning models used for these two release paths in the Catawba LOCA submittal (References 6 and 7) are adopted based upon a comparison of inputs between these two plants.

ECCS leakage does not begin until sump recirculation flow starts, in accordance with RG 1.183. The source is released to the sump using the same timing and release model as the containment atmosphere, except that Noble Gases are not included. But, only elemental and organic iodines are released from the ECCS in the specie fractions described in Section 4.4.10. Particulates are assumed to be retained in the sump water. Besides being part of the failed fuel source term, iodines can also be produced as daughter products from tellurium. This production of additional iodine is included in the model for the duration of the 30 day ECCS leakage release period.

The volume of water in the sump is modeled as a time dependent parameter, as shown in Table 6. The time dependent model is more conservative than a model which uses a constant volume based upon the long term equilibrium sump volume because the activity concentration in the sump is greater when the volume of water in the sump is smaller. Thus, there is a greater activity release for a given rate of flow from the sump prior to the establishment of the equilibrium sump volume. The leakage rate, however, is the same regardless of the sump volume or the number of trains of ECCS in service.

ECCS leakage starts at the time that realignment for sump recirculation (3000 seconds) begins and auxiliary spray starts. This leakage is not assumed to cease until the end of the accident. Holdup and dilution are not credited in the Auxiliary Building or the FWST. Leakage into these nodes/volumes is released to the environment as quickly as possible in the model to bound actual conditions. Thus, VA system flow was not explicitly modeled since releases from the ECCS to the Auxiliary Building were assumed to be immediately transported to the unit vent for release (by the VA system). Likewise, the FWST was not modeled with any appreciable volume.

Table 6
Time Dependent Containment Sump Water Volume Model

Time		Sump Volume
(sec)	(hrs)	(ft^3)
0	0	0
45	0.0125	19,000
1560	0.4333	56,240
1800	0.5	59,600
1830	0.5083	60,020
3000	0.8333	72,140
3600	1.0	74,075
4800	1.333	76,900
6000	1.667	77,300
8700	2.417	77,400
10,200	2.833	77,600

While RG 1.183 provides a 10% release fraction model when sump water temperature is held below boiling, it also provides an option for licensees to use a technically justifiable plant specific partitioning model for ECCS leakage paths. Duke submitted partitioning models in the Oconee AST (Reference 17) MHA LAR and the Catawba AST LOCA LAR. The Catawba model calculated reduced partitioning fractions (Reference 6) which were reviewed as part of that submittal (Reference 7). As will be discussed in Sections 4.5.5 and 4.5.6, comparisons of significant McGuire and Catawba parameters and their post LOCA responses show that the Catawba partitioning model is bounding for McGuire and, thus, can be adopted for the McGuire AST LOCA analysis.

In addition, during the review of the Catawba LOCA AST LAR, the ECCS leakage partitioning models were discussed with the NRC Staff via phone. The response to these RAIs is captured in the first half of Reference 20. Since McGuire is employing the same methodology and models, the explanatory information presented in Reference 20 related to the LOCA analysis ECCS partitioning models is applicable to the McGuire LOCA modeling methodology.

4.5.5 Partitioning of ECCS Back-leakage Released to the Auxiliary Building

RG 1.183 Appendix A (Sections 5.4 and 5.5) provides two options for ECCS leakage partitioning factors. The choice between these two options is dependent upon whether the temperature of the sump fluid exceeds 212 °F. The post LOCA sump temperature profile for McGuire is shown in Figure 3. Sump temperature does not exceed 180 °F during the time that the plant is in sump recirculation (after 3000 seconds). Because the temperature of the McGuire leakage is less than 212 °F, RG 1.183 Appendix A Section 5.5 provides the option of a static model that releases 10% of the iodine activity during the accident, or it allows for the potential for the application of smaller partitioning factors based upon a technically justifiable plant specific model. The same partition factors and model used in the Catawba AST analysis were adopted for McGuire based upon the comparison described below. Further details can be found in Section 1.5.2 of Reference 6, as well as the subsequent correspondence related to that model in Reference 20.

The values of the parameters affecting the Catawba partitioning model were compared to the corresponding values for McGuire to determine whether this model could be conservatively applied to McGuire. The iodine partition fraction for post LOCA ECCS System leakage depends on variables such as hydrogen ion concentration (pH), iodine concentration, Auxiliary Building Filtered Ventilation airflow rates and air temperatures over the leakage, characteristics of the rooms in which the leakage was enclosed, and the leak rate. The appropriate values for these variables for Catawba and McGuire were compared. The evaluation included reviews of design documents and plant walkdowns.

From this evaluation, the following comparisons were noted:

1) The minimum transient sump pH (referenced to standard conditions) reported in the Catawba submittal was 7.3 which is the same as is reported for McGuire in this submittal. The lower bound equilibrium sump pH (at standard conditions) reported for Catawba was 7.7 which is slightly lower than the McGuire value of 7.8. Since the McGuire equilibrium pH is slightly higher than Catawba's, the hydrogen ion concentration in McGuire ECCS leakage will be slightly lower than the hydrogen ion concentration in Catawba's ECCS leakage. Lower hydrogen ion concentration yields a lower iodine partition fraction for ECCS leakage which results in less release.

2) Long term sump water volume is slightly lower for McGuire (77,600 ft^3) relative to Catawba (79,000 ft^3). Since both use the same source term isotopics, the iodine concentration would increase in a lower sump water volume (McGuire) which would tend to yield a slightly higher iodine partition fraction for McGuire ECCS leakage.

3) The long-term post accident sump water temperature is slightly lower for McGuire (175 °F) than for Catawba (185 °F), which would also tend to yield a

slightly lower iodine partition fraction for ECCS leakage. This offsets the impact from the long term sump water volume.

4) The total floor area of the rooms which could receive ECCS leakage is slightly less at McGuire. The room heights are the same, so the room volume is slightly less for McGuire (than Catawba).

5) The airflow rate in the affected rooms at McGuire is nearly half the rate at Catawba, resulting in lower air velocities in the affected areas. The lower air velocity at McGuire will result in lower partitioning values relative to Catawba.

6) The post accident air temperature in the affected rooms is slightly lower at McGuire which would tend to result in lower partitioning.

From this review, it was seen that the cross sectional areas for airflow were essentially the same at Catawba and McGuire. Therefore, the lower McGuire airflow rate will result in lower VA filtered airflow velocities for McGuire compared to Catawba. This yields a lower rate of mass transfer across the surface of a pool of ECCS leakage. The floor areas of the affected rooms are slightly lower for McGuire than Catawba. This yields a slightly lower mass transfer rate across a pool of leakage. The bounding post accident air temperatures in the rooms with ECCS equipment are slightly lower for McGuire compared to Catawba. The resulting effect is a lower value for mass transfer of iodine across a pool of ECCS leakage.

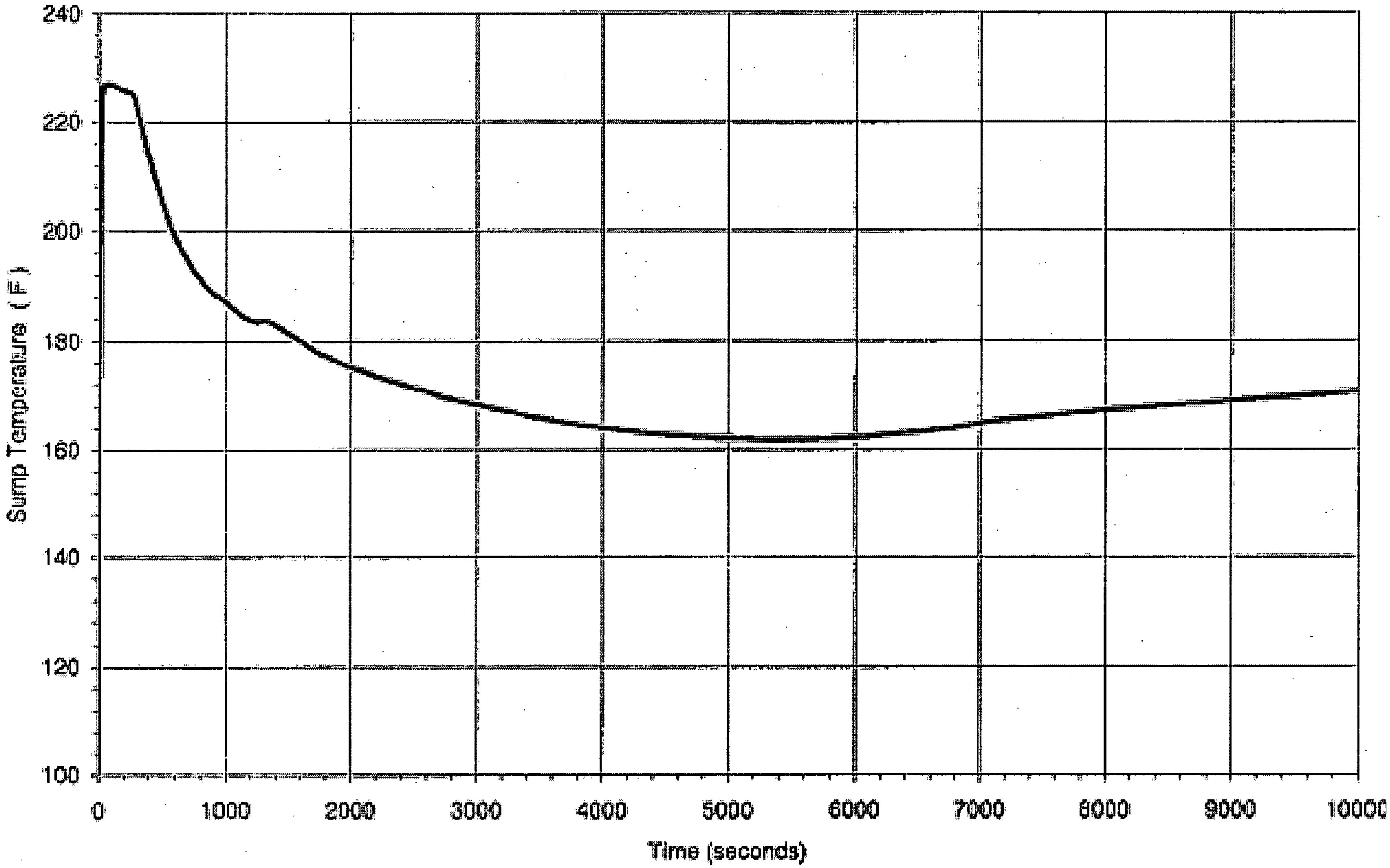
Iodine partition fractions for ECCS leakage in the Auxiliary Building were developed by Duke based on a limiting sump pH history and limiting room ventilation flow rates. These partitioning values originally were calculated for the Catawba AST LOCA LAR (Reference 6). The iodine partition fraction for ECCS leakage in the Auxiliary Building for Catawba is an upper bound for McGuire and adoption of the Catawba model is appropriate and conservative for McGuire.

The time dependent partitioning factors for ECCS back-leakage to the Auxiliary Building are shown in Table 7. This is the same set that was used in the Catawba AST LOCA analysis and submittal. Further detailed information on this model is contained in Reference 20.

Table 7
Design Basis Iodine Partition Fractions and Release Rates for 1 gpm ECCS Leakage into the McGuire Auxiliary Building

Start Time (hr)	**End Time** (hr)	**Iodine Partition Factor**	**Release Rate** (cfm)
0	3	0.100	**0.01337**
3	72	0.028	**0.00374**
72	720	0.010	**0.001337**

Figure 3: McGuire Post LOCA Sump Temperature Response



4.5.6 Partitioning of ECCS Back-leakage to the FWST

The initiation of sump recirculation flow also results in leakage to the Refueling Water Storage Tank. As with Auxiliary Building ECCS leakage partitioning, Duke has created an in-house model for partitioning FWST releases. This phenomenon was also modeled in the Catawba AST LOCA LAR using results from the IODEX code which is based on NUREG/CR-5950 (Reference 5). This method was developed by Duke to calculate the release fraction for iodine transport to the FWST, formation of volatile diatomic iodine, partitioning of iodine to the airspace in the FWST, and release to the environment. It is essentially the same method as that used in the analyses of post MHA ECCS back-leakage to the Borated Water Storage Tank at Oconee Nuclear Station (Reference 17).

The significant features of this method are:

- The containment sump iodine inventory includes stable I^{127} and long-lived I^{129}.
- FWST back-leakage is simulated for the assumed duration of the event (30 days).
- No refill of the FWST is simulated.
- FWST pH and hydrogen ion concentration is calculated at the solution temperature.
- FWST atmosphere air displacement from the back-leakage and from diurnal expansion is simulated.

This method was employed to calculate the portion of iodine entrained in the ECCS leakage to the FWST that is released to the environment following a design basis LOCA at Catawba. Significant input parameters to the model were compared for both plants to determine whether the Catawba release fractions could conservatively be applied to the McGuire LOCA.

1) **Post LOCA containment sump pH and associated parameters:**
As discussed above in Section 4.4.10, the minimum sump pH for McGuire is the same as that reported for Catawba and the equilibrium pH is slightly higher for McGuire. The primary solutes are sodium ions and borate ions. The sump sodium concentration is slightly lower for McGuire than for Catawba. However, the sump boron concentration also is lower for McGuire than for Catawba. This offsets the slight difference in sodium concentration, as indicated by the difference in long term post LOCA sump pH. The Catawba sump temperature profile is more limiting due to the higher long term temperature. The effect from the higher boron concentration at Catawba dominates the effect of the lower ice melt mass (and lower sodium concentration at McGuire) which results in a lower sump pH for Catawba. Thus, the Catawba post LOCA sump chemistry IODEX input is limiting.

2) **Initial FWST water temperature and boron concentration**
The upper bound temperature of the water in the FWST is the same for both McGuire and Catawba (100 °F). The upper bound value of the initial boron

concentration for Catawba (3075 ppm) is higher than the initial boron concentration for McGuire (2875 ppm) and, therefore, limiting.

3) **FWST dimensions**
The dimensions of the FWST are the same for McGuire and Catawba.

4) **Rate of ECCS back-leakage to the FWST**
The value for this parameter is set to 20 gpm for both McGuire and Catawba Nuclear Stations. Holdup of this leakage in the FWST tank is not modeled or credited.

5) **Meteorological inputs to IODEX**
The meteorological parameters germane to iodine releases from the FWST include outside air temperature (lower bound), temperature swing (difference between daily low and high), and amount of daylight (upper bound). The data for these parameters is associated with the greater Charlotte-Mecklenburg area and therefore is common to both McGuire and Catawba Nuclear Stations.

The review of the significant parameters shows that they are the same or nearly the same for McGuire and Catawba. Where a slight difference exists, the effect of the Catawba value was shown to bound McGuire. Thus, the inputs associated with the Catawba ECCS back-leakage iodine FWST release fractions bound those which would be associated with similar releases from the McGuire FWST.

Table 8 summarizes the FWST partitioning model and release rates. Further detailed information on this model is contained in Reference 20.

Table 8
FWST Release Model for 20 gpm ECCS Back-leakage

Start Time (sec)	End Time (sec)	IODEX release fraction	Release Rate (cfm)
0	790	0	**0**
790	810	9.197E-11	**2.459E-10**
810	900	2.894E-09	**7.739E-09**
900	1200	3.443E-08	**9.207E-08**
1200	1400	9.799E-08	**2.620E-07**
1400	1800	1.772E-07	**4.738E-07**
1800	3600	3.486E-07	**9.321E-07**
3600	4800	4.228E-07	**1.131E-06**
4800	6000	4.128E-07	**1.104E-06**
6000	7200	3.916E-07	**1.047E-06**
7200	28800	3.376E-07	**9.027E-07**
28800	36000	3.284E-07	**8.781E-07**
36000	86400	1.873E-07	**5.008E-07**
86400	345600	3.444E-07	**9.209E-07**
345600	2592000	6.388E-06	**1.708E-05**

4.6 SYSTEM RESPONSES TO THE LOCA AND RELEASE MITIGATION

4.6.1 The Annulus Ventilation System

The Annulus Ventilation System (VE) is responsible for establishing a vacuum in the annulus to promote air flow from adjacent higher pressure spaces into the annulus where it can be held and filtered prior to release. Prior to the establishment of vacuum in the annulus, it is assumed that containment leakage is released directly to the environment. After the establishment of annulus vacuum, only 7% of the leakage bypasses the annulus, as discussed in Section 4.3.3.

In response to a LOCA, the McGuire VE system starts in exhaust mode, discharging through the filters, to reduce pressure in the annulus. The VE system is required to maintain the annulus between -0.5 and -3.5 inwg. In order to assure that the model reflects this requirement or is conservative relative to it, the worst case impact from a difference between the thermodynamic conditions in the annulus and those conditions external to it is modeled by including the effect on stagnant head pressure from the elevation difference between the top of the annulus and the pressure detector with an extreme outside air temperature. Instrument uncertainty is also included. Together, these factors could cause the pressure detector to indicate up to 0.7 inwg lower than the conditions at the top of the annulus. So, to ensure that the modeled setpoints were inclusive of the pressure at the most remote location in the annulus (and thereby throughout the annulus), and to ensure that the pressure requirement was satisfied (or exceeded) in the worst case at all locations, -0.7 inwg was applied to the setpoints to arrive at a modeled control band of -1.2 inwg to -4.2 inwg. Thus, at a modeled -4.2 inwg the system switches to recirculation mode. In this mode the annulus environment is recirculated by the VE fans through the VE filters. No release is made to the environment. When modeled annulus vacuum drops to -1.2 inwg (rising pressure in the annulus), the system swaps back to exhaust.

In the LOCA scenario, the VE system starts at 39 seconds (starts 34 seconds after the accident with a 5 second fan startup time) into the accident in response to the generation of the Phase B containment isolation signal at 3 psig containment pressure. It starts in exhaust mode and discharges to the environment via the VE filters and the unit vent stack. When indicated annulus pressure is -0.9 inwg, VE is assumed to have achieved a vacuum (-0.25 inwg) throughout the annulus (at 71 seconds). The modeled value of -0.9 inwg includes indication conservatism. By requiring greater vacuum, the length of time that all containment leakage bypasses the annulus and the duration of the initial exhaust cycle are both lengthened. Both of these features are conservative.

Annulus vacuum is established slightly greater than a minute after accident initiation with only one train available. In exhaust mode, annulus vacuum increases in the model until -4.2 inwg is reached. The system then realigns to recirculation mode and the annulus atmosphere is recirculated. The discharge of the VE system is directed through the VE filters and back into the annulus. During recirculation, the pressure in the annulus will increase as it loses vacuum due to in-leakage from surrounding spaces and the environment to the annulus. VE system in-leakage modeling is based upon an exterior environment temperature of 18 °F which satisfies the 95th percentile temperature data requirement of RG 1.183. Since Catawba and McGuire are both in the Charlotte Mecklenburg area, they share the same weather data; this temperature is the same as was reported for Catawba. When indicated annulus pressure reaches -1.2 inwg, the system realigns to exhaust mode and the atmosphere in the annulus is again discharged (including radionuclides in the annulus) through the VE filters to the unit vent stack. The system continues to change modes between exhaust and recirculation as these set points are met for the duration of the accident.

In the Minimum Safeguards scenario (the limiting scenario), the performance of only one train of VE is modeled due to the postulated single failure which removes the other train from service. The operation of the system leads to a large number of timesteps as the systems swaps between recirculation and exhaust modes. At the beginning of the accident, these time steps show some variation in the duration of the exhaust and recirculation modes. As the accident progresses, the VE system performance stabilizes into a more predictable (and eventually a perpetually modeled) pattern.

Besides the removal of one train of the system by single failure, VE response is further degraded by assuming the lowest acceptable fan performance for the duration of the accident. The VE fan is assumed to operate at 7200 cfm, which is the lowest flow (8000±10%) performance that the fan is permitted. This reduction in flow bounds a diesel generator under-frequency and under-voltage scenario in which a proportional fan flow reduction would be expected from the diesel generator operating at its lowest Technical Specification SR 3.8.1.2 permitted frequency (2% below nominal), and voltage (10% below nominal). Lower VE fan flow is conservative as it requires greater amounts of discharge (and longer times in discharge model) to establish the initial annulus vacuum and maintain the vacuum during subsequent exhaust modes. It also provides for less recirculation filtration during operation in that mode.

Table 9 provides a summary of the timesteps and the nodal flow model for the response of one train of VE to a LOCA at McGuire. After the time associated with the last listed sequence step, the exhaust and recirculation durations provided at the bottom of the table are used until the end of the problem.

The computation of VE filter efficiencies is described in Section 4.6.11.

4.6.2 Effect of High Wind Speed on the Annulus Ventilation System Performance

RG 1.183 requires that the effect of high wind speed on the ability of the secondary containment to maintain a vacuum be modeled. The assumption of high wind speed would add a slight amount of conservatism with respect to local effects on the stagnant pressure at the Reactor Building. But, high wind speed would greatly reduce off-site and control room atmospheric dispersion factors. Using low wind speeds to calculate χ/Qs while not including high wind speeds in post LOCA operation of the VE System results in higher (more conservative) calculated radiation doses than does modeling high wind speed in the calculation of χ/Qs and post LOCA VE operation.

The only credible leak path from the environment to containment (and the annulus) is through the equipment hatch. The equipment hatches of both units are located very close to a corner defined by the Seismic Category I Reactor Building and Seismic Category I Fuel Building. Furthermore, they lie behind the Seismic Category I equipment hatch missile shields. These shields are locked closed during power operations. The effect of wind speed on VE System response is made insignificant by these features and the assumption of low wind speeds is the more conservative option if this parameter is to be consistently applied and its effect consistently modeled.

This position is identical to that reviewed and approved in the Catawba AST LOCA submittal and SER (References 6 and 7).

4.6.3 Annulus Volume and Annulus Mixing Credit

RG 1.183 allows credit for mixing in the annulus (secondary containment) when adequate mixing exists. Since the VE systems on the two units are similar, Unit 1 can be arbitrarily chosen as the basis for evaluation. The return headers (suction supply to the VE fans) are at elevations 844+0 and 853+0. These return headers are arranged in two ring headers at each elevation and run almost around the entire circumference of the annulus at these levels except for about a 20 degree gap. Thus, there is a fairly even distribution of return intakes around these headers. The discharges are located in a less uniform pattern. The discharges are located around the annulus in more discreet locations at elevations between about 730+0 and 770+0, with most located below 750+0.

Other than the airlocks, the containment penetrations are relatively small and distributed around the annulus. Most penetrations are below the 767+5 level, except for the equipment hatch and the upper personnel airlock as well as a few piping penetrations. The equipment hatch is a major bypass leakage release point. Therefore, the majority of the leakage to the annulus would be expected to occur via the penetrations at the lower elevations. This leakage would occur mainly at elevations near the locations of VE discharges. The leakage flow would then be driven by the differential pressure between the discharges at the lower elevations to the upper elevation suction header. At a minimum, this would be a change in elevation of approximately 70 feet, with a more nominal elevation change of 100 feet or more.

The supposition of natural circulation airflow in the annulus is reasonable. Lower containment will contain a large heat source as well as a large gamma flux resulting in heat transfer to, and gamma heating of, the lower annulus atmosphere. The upper portion of the annulus will be cooler than the lower portion just as upper containment is cooler than lower containment due to sprays. This will promote a natural exchange of atmospheres between the lower and upper portions of the annulus.

The discussion above, including the distribution of the penetrations around the annulus, the relatively small width of the annulus (only a couple of feet), and the large elevation difference between the discharges and main in-leakage locations and the return header, supports a conclusion of nearly complete mixing of leakage into the annulus prior to discharge from the VE system.

Therefore, mixing in the annulus is credited. A 50% credit for annulus mixing is permitted by Reference 1. This results in a volume of approximately 213,000 cubic feet.

Table 9
Annulus Ventilation System Performance Model
in a Minimum Safeguards (Single Train) Scenario

Sequence Number	Initiation of Discharge (sec)	Duration of Exhaust (sec)	Initiation of Recirc (sec)	Duration of Recirc (sec)	Volumetric Flow Rate (cfm)	Quantity Discharged (ft^3)
1	39	168	207	22	7200	20160
2	229	105	334	27	7200	12540
3	361	57	418	33	7200	6900
4	451	43	494	40	7200	5160
5	534	35	569	49	7200	4260
6	618	31	649	61	7200	3720
7	710	28	738	73	7200	3360
8	811	26	837	86	7200	3120
9	923	25	948	95	7200	3000
10	1043	24	1067	101	7200	2940
11	1168	24	1192	107	7200	2940
12	1299	24	1323	107	7200	2880
13	1430	24	1454	100	7200	2880
14	1554	24	1578	98	7200	2940
15	1676	24	1700	98	7200	2940
16	1798	24	1822	102	7200	2940
17	1924	24	1948	108	7200	2880
18	2056	24	2080	117	7200	2820
19	2197	24	2221	126	7200	2820
20	2347	23	2370	131	7200	2760
21	2501	23	2524	134	7200	2760
22	2658	23	2681	133	7200	2760
23	2814	23	2837	126	7200	2760
24	2963	23	2986	120	7200	2820
25	3106	24	3130	116	7200	2820
26	3246	24	3270	121	7200	2820
27	3391	24	3415	134	7200	2820
28	3549	23	3572	145	7200	2760
29	3717	23	3740	148	7200	2700
30	3888	22	3910	140	7200	2700
31	4050	23	4073	126	7200	2760
32	4199	23	4222	118	7200	2820
33	4340	23	4363	111	7200	2820
34	4474	24	4498	109	7200	2880
35	4607	24	4631	107	7200	2880
36	4738	24	4762	105	7200	2880

Sequence Number	Initiation of Discharge (sec)	Duration of Exhaust (sec)	Initiation of Recirc (sec)	Duration of Recirc (sec)	Volumetric Flow Rate (cfm)	Quantity Discharged (ft^3)
37	4867	24	4891	103	7200	2880
38	4994	24	5018	101	7200	2940
39	5119	25	5144	98	7200	2940
40	5242	25	5267	97	7200	2940
41	5364	24	5388	96	7200	2940
42	5484	24	5508	96	7200	2940
43	5604	24	5628	98	7200	2940
44	5726	24	5750	101	7200	2940
45	5851	24	5875	102	7200	2880
46	5977	24	6001	103	7200	2880
47	6104	24	6128	105	7200	2880
48	6233	24	6257	106	7200	2880
49	6363	24	6387	106	7200	2880
50	6493	24	6517	110	7200	2880
51	6627	23	6650	114	7200	2820
52	6764	24	6788	121	7200	2820
53	6909	23	6932	126	7200	2820
54	7058	23	7081	128	7200	2760
55	7209	23	7232	129	7200	2760
56	7361	23	7384	131	7200	2760
57	7515	23	7538	131	7200	2760
58	7669	23	7692	132	7200	2760
59	7824	23	7847	132	7200	2760
60	7979	23	8002	132	7200	2760
61	8134	23	8157	132	7200	2760
62	8289	23	8312	132	7200	2760
63	8444	23	8467	133	7200	2760
64	8600	23	8623	132	7200	2760
65	8755	23	8778	133	7200	2760
66	8911	23	8934	132	7200	2760
67	9066	23	9089	133	7200	2760
68	9222	23	9245	132	7200	2760
69	9377	23	9400	133	7200	2760
70	9533	23	9556	133	7200	2760
71	9689	23	9712	133	7200	2760
72	9845	23	9868	133	7200	2760
duration of accident		23		133	7200	2760

4.6.4 Containment Air Return System

The Containment Air Return (VX) System works with the Containment Spray (NS) System to remove activity from the containment atmosphere, although the VX System does so indirectly as it has no filtration or mitigation features of its own. The purpose of this system is simply to promote the exchange the atmospheres of the upper and lower containment compartments. In doing this, the system returns air from upper containment to lower containment. The air in lower containment is forced through the ice condenser and into upper containment where the spray system can remove the activity deposited in its atmosphere. While it is expected that activity will be removed by the ice condensers, as will be discussed in Section 4.6.10, this removal mechanism is not credited. Thus, as modeled, the VX system works with the NS system to remove activity from the (upper) containment atmosphere.

The VX system does not possess any filters, so no filtration or any other mitigation action is directly credited to this system. It is modeled to start 10 minutes after the accident (9±1 minute) which is the latest possible time based upon worst case diesel generator loading. Since the spray system is not credited until the VX system moves activity into upper containment, and because the VX system starts much later than the spray system, this model delays spray credit by about 8 minutes. The single failure removes one VX fan leaving only one in operation. This fan nominally produces 30,000 cfm, but in order to bound any fan performance reduction due to a diesel generator under-frequency and under-voltage conditions, its flow is modeled at 29,000 cfm, which is conservative relative to the reduction expected to result from these conditions.

4.6.5 Containment Spray System

The Containment Spray System (NS) is the primary activity mitigation system inside of containment. NS is used to remove heat from containment and to remove iodine from the containment atmosphere (radiological consequences purpose). Iodine is removed by the NS system by "washing" it from the atmosphere through interactions between the airborne iodine and the sprayed water droplets. The spray headers are located high in upper containment. Because the operating deck separates upper and lower containment, spray is only effective at removing iodine from the upper containment atmosphere.

The operating deck effectively prevents direct communication between upper and lower containment, so the release from the breech in the Reactor Coolant (NC) System is made to lower containment only. The source term remains in lower containment until the Containment Air Return System (VX) system starts at 600 seconds, as discussed in Section 4.6.4. VX fans force air flow from upper containment to lower containment which promotes the flow of the lower containment atmosphere through the ice condensers to upper containment. Once the activity reaches upper containment it can be removed by the spray system which provides full spray flow coverage to upper containment by 120 seconds after accident initiation. Initially (injection phase), the spray system uses Refueling Water Storage Tank (FWST) water to provide suction to the spray pumps. When that supply is exhausted, the system is realigned to recirculation mode to take suction from the containment sump.

Activity removal by the spray system is modeled using "spray lambdas". The lambdas model the ability of the NS system to remove elemental and particulate iodines from the upper containment atmosphere in a post LOCA environment. They are derived using inputs related to the characteristics of containment, the flow characteristics of the NS system, and, during recirculation, the chemistry (pH) and temperature profiles of the containment sump fluid.

Once the decontamination factor (DF) exceeds its limit, the NS System effectiveness in washing the iodine from the containment atmosphere is reduced. The effectiveness of spray for particulates is reduced by a factor of ten when the DF for particulates reaches 50 (7100 seconds) as required by References 1 and 19. Elemental iodine spray credit ceases when its DF reaches 200 (at 46,000 seconds), again as required by References 1 and 19. Organic Iodines are not removed by sprays in the model.

In the Minimum Safeguards scenario, only one train of spray responds. Full spray flow (injection flow from the FWST) is provided to upper containment by 120 seconds post accident initiation. At 3000 seconds, sump recirculation begins. While the spray system is realigned for sump recirculation, spray flow is supplied from the auxiliary spray header via the Residual Heat Removal (ND) System pumps. At 3240 seconds, the NS system begins supplying recirculated spray flow. This continues until spray flow is assumed to be secured 24 hours into the accident. Full spray coverage is achieved under minimum safeguards NS operation. The credited spray flow reflects reductions for water flow which could impinge upon the containment walls, rather than falling through upper containment to the operating deck. The spray pump flowrate modeled has been reduced to bound the effect of reduced diesel generator frequency and voltage.

The spray cutoff times for elemental and particulate iodine and their spray lambdas are reflected in Table 10. These values were generated with the WASHOUT code whose methodology is based on Reference 18 and Section 6.5.2 of Reference 19.

Table 10
Spray Lambdas for One Train of VX, NS, and ND (Minimum Safeguards)

Start Time (sec)	**End Time** (sec)	**Elemental Spray Lambda** (hr^{-1})	**Particulate Spray Lambda** (hr^{-1})
0	120 (600)[1]	**0**	**0**
120 (600)[1]	3000	**20**	**9.36**
3000	3240	**0.22**	**7.19**
3240	3500	**0.50**	**16.5**
3500	4000	**0.53**	**16.5**
4000	4500	**0.56**	**16.5**
4500	5000	**0.58**	**16.5**
5000	7100	**0.59**	**16.5**
7100	24,600	**0.59**	**1.65**[2]
24,600	30,000	**0.58**	**1.65**[2]
30,000	40,000	**0.56**	**1.65**[2]
40,000	46,000	**0.53**	**1.65**[2]
46,000	86,400	**0 (No credit)**[3]	**1.65**[2]
86,400	end	**0 (No credit)**[4]	**0 (No credit)**[4]

[1] Spray is not credited with iodine removal in radiological modeling until VX starts at 600 seconds. McGuire spray flow starts by 120 seconds.
[2] After 7100 seconds, the particulate spray lambdas are reduced by a factor of 10 for reduce spray effectiveness due to particulate spray washout at DF of 50.
[3] After 46,000 seconds, spray washout occurs for elemental iodines (DF reaches 200) and credit ceases for elemental spray removal.
[4] Spray is not credited for iodine removal after 24 hours.

4.6.6 Auxiliary Building Ventilation System Filtered Exhaust

The filtered exhaust portion of the Auxiliary Building Ventilation (VA) System establishes and maintains a negative pressure in the ECCS pump rooms. It is also credited with mitigating activity released into these rooms from ECCS back-leakage. Air drawn from these rooms into the filtered portion of the VA System is released from the unit vent making this the only accident scenario release path for which VA mitigation is credited. The system filter units include two inch thick nuclear grade activated carbon beds.

The VA filtered exhaust system was not initially designed as a safety related system nor credited to mitigate a design basis accident. However, during initial plant licensing the system was re-classified as an Engineered Safety Feature (ESF) atmosphere cleanup system and was included in Technical Specifications. In August 1975, the NRC stated

(Reference 21) that McGuire's VA filtered exhaust system is needed to mitigate the dose consequences of a postulated ECCS pump seal failure during a LOCA, and that it met RG 1.52. To meet this new requirement, Duke and the NRC agreed upon a minimum level of upgrades to the system, including:

- upgrade the filter units to safety grade
- provide class 1E safety related power to the filtered exhaust fan motors and controls
- place the fan motors on the emergency diesel generator sequencers

The existing system meets the criteria of RG 1.52 with the exceptions (Reference 22) documented in McGuire UFSAR Table 9-38. Technical Specification Surveillance Requirement 3.7.11.4 demonstrates that a VA train maintains a negative pressure in the ECCS pump rooms with the system in the post LOCA mode of operation.

Hold up in the Auxiliary Building is not credited. The model assumes instantaneous transport of activity through the Auxiliary Building, so it bounds any realistic fan flowrate.

The computation of filter efficiencies is described in Section 4.6.11 of this LAR.

4.6.7 Control Room Unfiltered Air In-leakage

The impact of the accident on the control room operators is calculated for 30 days after accident initiation. Activity enters the control room through the VC outside air intakes. The LOCA results in a large, swift energy release from the NC System. The pressure increase in containment is very rapid and it is assumed that the Engineered Safeguards Actuation System actuates nearly instantaneously with the initiation of the accident. The control room is automatically pressurized by the VC system in response to the Engineered Safeguards Actuation System actuation. In the Minimum Safeguards scenario, only a single train (fan and filter) of filtered, outside air, pressurization responds. The fan provides airflow at the lower limit of the test acceptance band starting 11 seconds after the accident and pressurization is assumed complete 30 seconds post accident initiation. The control room remains pressurized for the duration of the accident.

Pressurization of the control room also affects the modeled rates of unfiltered in-leakage which are based upon the results of control room tracer gas testing conducted in October 2003. The nominal results of those tests were adjusted to include the maximum quantified test error band (in accordance with Reference 23), and an allowance of 10 cfm for control room ingress and egress. This result was then increased for additional conservatism. The value of 625 cfm for in-leakage prior to pressurization is selected as a bounding value based upon the nominal testing value of 505±15 cfm. The unfiltered in-leakage value after pressurization is computed similarly, but it is also based upon the most conservative testing result. The test was performed in two train and single train configurations. The most conservative nominal value (plus its associated error band) was used as the basis for the computation of the post pressurization value. The conservatisms described above were also included in the computation of the modeled pressurization unfiltered in-leakage value. The value modeled, 210 cfm, was selected as a bounding value based upon the limiting case test result of 131±36 cfm. The in-leakage values used in the models are summarized in Table 11. They are consistent with those used in McGuire AST Fuel Handling Accident submittal (Reference 10). These values bound the current testing results with allowances for uncertainties and control room ingress and egress. The results of future tracer gas tests could change the values for control room unfiltered in-leakage shown in Table 11.

Table 11
Summary of Control Room Unfiltered In-leakage Rates and VC System Model Parameters for the Minimum Safeguards Scenario

Status of Control Room	**Filtered Airflow Supplied to the Control Room** (cfm)	**Unfiltered In-leakage** (cfm)
Prior to Pressurization (t < 30 seconds)	0	625
One Train after Pressurization (t ≥ 30 seconds)	1800	210

The air flow rate supplied from the VC system to the control room for each scenario is based upon the lowest permitted flow from the fan. Each fan is required to supply 2000 ± 10% cfm to the control room. For these calculations, 1800 cfm is modeled for a single VC fan and train. Low air flow to the control room is also more conservative than higher air flow to the control room for this scenario due to a longer activity residency time in the control room. The control room volume is constant at 1.07E+05 cubic feet. This model is consistent with the McGuire AST FHA submittal (Reference 10).

4.6.8 Control Room Receptor Modeling

The control room model also includes several factors to aid in quantifying the impact to the operator. In particular, these are the amount of time that the operator is assumed to spend in the control room over this 30 day period and his rate of inhalation of airborne radioactivity. These values are taken from RG 1.183 and are summarized in Table 12.

Table 12
Control Room Operator Receptor Constants

Time	Control Room Occupation Factor	Control Room Operator Breathing Rate
Initiation to 1 day	100%	3.5E-04 m^3/sec
1 day to 4 days	60%	3.5E-04 m^3/sec
4 days to 30 days	40%	3.5E-04 m^3/sec

4.6.9 Control Room Area Ventilation System Modeling

The McGuire VC system originally included a means to recirculate the control room atmosphere. The filtered recirculation mode of operation is not used, and thus, is not modeled or credited. The McGuire control room is operated and modeled as a "once-through" ventilation system.

McGuire was categorized as a dual intake plant during initial licensing (Reference 24). It has no manual or automatic selection controls. The VC system has two trains, each train has one intake location with two separate inlets for a total of four inlets. The intake locations (see Appendix A) are sufficiently separated to eliminate the possibility of both locations being exposed to a concentrated plume of activity (Reference 24). The VC outside air inlets 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 outside air inlets (in conjunction with the remainder of the VC system) to achieve and maintain a positive pressure in the control room is not degraded by tornado missiles. These system features were also discussed in Reference 10.

The isolation valves for each of the inlet paths fails "as is" and receive no automated closure demands. No failure mode was identified that will cause an outside air inlet or one of its isolation valves to close. The minimum normal plant operation configuration of this system provides for one of the four inlets to be out of service or closed. In this alignment there are two inlets open at one intake location and one inlet open at the other (commonly written "2/1"). This is the assumed configuration of the VC system in Mode 1 at the beginning of the postulated accident.

Duke has recognized that the control room intake flow distribution between the two intake locations changes with the system alignment and that the system may not be perfectly balanced at all particular points in time. McGuire recently tested the "flow split" of its system in nine potential alignments which encompass combinations of either fan running alone or both fans running, with a single inlet isolated at either location or all inlets open. A flow split of 65%/35% was established to bound the test results from all system alignments. Thus, 65% of the VC intake flow is assumed to be from the contaminated stream and 35% is assumed to be from the non-contaminated stream for all normal VC alignments. This flow split is applied to the control room atmospheric dispersion factors prior to their entry into the radiological consequences model.

When the computation of filter efficiencies is described in Section 4.6.11, a safety factor of 2 will be included as margin against the degradation of filter performance between filter tests. McGuire also has a monitoring/maintenance program in place for ensuring VC system in-leakage performance does not significantly degrade relative to the control room in-leakage test results. This program includes intrusion from toxic gas, unfiltered in-leakage, and smoke.

Toxic Gas

- For off-site chemicals, the preventive maintenance program initiates a periodic review for new toxic/hazardous chemicals within McGuire's 5 mile radius.
- For on site chemicals, initial chemical purchases greater than 100 pounds are reviewed by Systems Engineering.

Unfiltered In-leakage

- The preventive maintenance program requires the inspection of portions of the control room boundary (duct, flex joints, air handling units) that are outside the control room. This inspection focuses on areas of the ductwork that are at a negative pressure as compared to the ambient pressure adjacent to these portions.
- Flexible joints associated with certain air handling units in the control room envelope are inspected periodically.
- Control room pressurization is performed periodically to assess the condition of the control room. During testing, various door seals in both frequently used and infrequently used doors are inspected.

Smoke

- In the case of smoke intrusion into the control room, certain critical functions of the control room are performed by the Standby Shutdown System (SSS). Depending upon the presence or absence of control room control functions, plant control is coordinated either from the Auxiliary Shutdown Panel (in the Auxiliary Building) or from the Standby Shutdown Facility (SSF). The SSF is a plant structure separate from the Reactor Building/Auxiliary Building/Turbine Building complex.

4.6.10 Mitigation Not Credited-- Ice Condensers and Natural Processes

If a LOCA were to occur at McGuire, it is expected that iodine activity releases would experience some mitigation from the ice condensers and from natural processes (plateout on Reactor Building surfaces). Neither of these phenomena is credited in the analysis of radiological consequences of a LOCA at McGuire. This serves as additional conservatism in the McGuire analysis. The primary activity mitigation process credited in containment is the containment spray system.

The resulting post LOCA pressure increase in lower containment will cause the ice condenser lower inlet doors to open to promote the flow of lower containment atmosphere into the ice condensers at accident initiation (prior to VX start). While this interaction in the ice condenser could provide some mitigation prior to VX start, ice condenser mitigation is not credited before or after VX start.

4.6.11 Filtration Modeling

During the review of the Catawba AST LAR, the NRC Staff communicated to Duke their expectations for the computation of filter efficiencies. Furthermore, the use of a design safety factor of 2 in this methodology as a hedge against degradation between filter tests was also communicated. McGuire has employed the guidance of RG 1.52, Revision 2 (Reference 26) and Generic Letter 99-02 (GL 99-02, Reference 25) to meet this expectation. These documents are included in McGuire's licensing basis and their application meets the intent of the Staff's direction. Additionally, and also in accordance with expectations communicated during the Catawba review, filter bypass flow is explicitly included in filtration modeling.

Attachment 2 to GL 99-02 contains a filter efficiency model which relates allowable filter penetration and design margin ("safety factor") to the credited/modeled filter efficiency in the consequence analysis. Table 2 of RG 1.52 (Reference 26) prescribes maximum elemental and organic filtration efficiencies. Specific guidance applicable to the Control Room Ventilation System particulate filter efficiency modeling is contained in Section C.5.c of RG 1.52 (Reference 26). The concept of a relationship between filter penetration and creditable efficiency with the application of a safety factor of 2 is applied to the VA and VE particulate filtration models.

<u>Organic and Elemental Filtration Modeling</u>

VC, VE, and VA filters are modeled using the following equation from GL 99-02 Attachment 2 with the application of a safety factor of 2:

$$\text{Allowable Penetration} = \frac{(100\% - \text{Methyl Iodide Efficiency for Charcoal Credited in Licensee's Accident Analysis})}{\text{Safety Factor}}$$

Carbon bed thicknesses and Technical Specification required filter characteristics are summarized in Table 13 along with the calculated filter efficiencies.

<u>Control Room Ventilation Particulate Filtration</u>

Section C.5.c of RG 1.52 (Reference 26) states that for a HEPA filter bank with a tested penetration of <0.05%, a removal efficiency of 99% can be credited. The MNS VC filters are tested at rated flow ± 10%. The VC filter testing conditions stated in the Ventilation Filter Test Program in Technical Specification 5.5.11 reflect these requirements. Thus, the VC particulate filter efficiency modeled is 99%.

VE and VA Particulate Filtration Modeling

These filter efficiencies are computed using a similar methodology and equation as the elemental and organic filtration models above. A safety factor of two is also employed in this model.

Technical Specification required filter characteristics are summarized in Table 13 below along with the calculated and modeled filter efficiencies.

Table 13
Control Room Ventilation, Auxiliary Building Ventilation, and Annulus Ventilation Requirements and Filter Efficiencies

Filter	Allowable Methyl Iodide Filter Penetration[1]	Allowable Charcoal Filter Bypass[2]	Credited Elemental and Organic Iodine Filter Efficiency	Allowable Particulate Filter Penetration and Bypass[3]	Credited Particulate Iodine Filter Efficiency
Control Room Vent. (VC) (4 inch carbon bed)	<0.95%	<0.05%	98.1%	<0.05%	99%
Auxiliary Bldg. Vent. (VA) (2 inch carbon bed)	<4%	<1%	92%	<1%	98%
Annulus Ventilation (VE) (2 inch carbon bed)	<4%	<1%	92%	<1%	98%

[1] McGuire Technical Specification 5.5.11.c
[2] McGuire Technical Specification 5.5.11.b
[3] McGuire Technical Specification 5.5.11.a

Only the VC filters are common to the McGuire AST LOCA analysis and the McGuire AST FHA analysis. The filtration efficiency modeling for the VC System is consistent for these two analyses.

4.7 ATMOSPERIC DISPERSION AND IN-LEAKAGE

Air dispersion factors are determined at three receptors: Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room (CR). The off-site receptor locations (EAB and LPZ) use the existing licensing basis dispersion factors. In response to control room dispersion modeling comments in the McGuire AST FHA SER Section 3.3.1.2 (Reference 2), Duke has updated the meteorological inputs to use only data from the current meteorological tower and has recomputed the control room dispersion factors. The recomputed values are discussed below.

4.7.1 Off-site Receptors Modeling and Atmospheric Dispersion

The existing licensing basis atmospheric dispersion factors are used in the LOCA analysis. No change has been made to these values. These are the same values used in the McGuire AST FHA submittal (References 2 and 10). The breathing rates from RG 1.183 are employed. The off-site χ/Q values are summarized in Table 14. The LOCADOSE code determines the 2 hour period of maximum releases and computes the EAB dose over than period using the 0-2 hr EAB χ/Q.

Table 14
Off-site Atmospheric Dispersion Factors and Receptor Data

Time	**EAB χ/Q** (sec/m^3)	**LPZ χ/Q** (sec/m^3)	**Off-site Breathing Rate** (m^3/sec)
0 to 2 hours	9.0E-04		3.5E-04
0 to 8 hours		8.0E-05	3.5E-04
8 hours to 1 day		5.2E-06	1.8E-04
1 day to 4 days		1.7E-06	2.3E-04
4 days to 30 days		3.7E-07	2.3E-04

4.7.2 Control Room Receptor Modeling and Atmospheric Dispersion for LOCA

The discussion of the VC System described some of the control room receptor modeling features, including the control room volume, occupancy factors and breathing rates (see Table 12). Potential leakage release points were described during the integrated plant and system response in Section 4.3.3. Table 15 lists the point to point dispersion factors for these identified potential release locations. Appendix A contains a sketch of the McGuire site annotated with potential release locations and the control room intake locations. A full listing and discussion of McGuire control room dispersion factors, and their related inputs, is presented in Appendix B.

The following release points are postulated and modeled:

- Unit Vent Stack

  The unit vent stack provides a release point for ventilation related systems including the Containment Purge System (VP), the Containment Air Release and Addition System (VQ), the Annulus Ventilation System (VE), and the Auxiliary Building Ventilation System (VA). Bypass leakage to the Auxiliary Building could also be exhausted to the unit vent by VA. The unit vent is also the ultimate discharge point for ECCS back-leakage releases to the Auxiliary Building, which are made via VA. The unit vents are modeled as point sources.

- Equipment Hatch

  The equipment hatch penetrates the reactor building in upper containment and is assumed to leak at 500 sccm. It leaks (preferentially) at this rate until it is halved with the rest of containment leakage 24 hours after initiation. The remainder of the leakage is available to other release points. The equipment hatch is modeled as a horizontal point source for all receptors, and also as a vertical area source representing an open door (or multiple holes) when the flowpath intake is on the same unit.

- Refueling Water Storage Tank (FWST)

  Back-leakage to the FWST is released from the tank vent to the atmosphere. Since the tank is continuously vented to atmosphere through an open pipe, this release path is always available. Releases are postulated to occur resulting from changes in plant conditions or changes in atmospheric conditions. The FWST vent is hooded so this source is modeled as a horizontal point source.

Table 15 lists the maximum source to receptor atmospheric dispersion factors calculated. These values have been calculated in accordance with RG 1.194 (Reference 27). They represent the worst case results from both inter-unit and intra-unit releases. For those release points which were modeled as several different source configurations (horizontal, vertical, etc.), the values shown in Table 15 represent the largest and most conservative (maximum) value for all of the source models and source/receptor pairs associated with this release point. The values in Table 15 have not been adjusted for control room intake or VC System configurations (inlet flow distribution flow split).

Table 15
Maximum Control Room Atmospheric Dispersion Factors (sec/m^3)

Time	**Unit Vent** M1UV1	**Equipment Hatch** M1EQ1PTM (arc)	**Refueling Water Storage Tank** M2FWST2
0 – 2 hours	1.66E-03	4.01E-03	1.83E-03
0 – 4 hours	1.47E-03	3.73E-03	1.74E-03
0 – 8 hours	1.41E-03	3.48E-03	1.62E-03
2 – 8 hours	1.32E-03	3.30E-03	1.55E-03
4 – 8 hours	1.35E-03	3.23E-03	1.50E-03
8 – 24 hours	6.75E-04	1.58E-03	7.60E-04
1 – 4 days	5.35E-04	1.23E-03	5.86E-04
4 – 30 days	4.05E-04	9.63E-04	4.36E-04

Not adjusted for dual intakes or VC System flow split.
See footnote in Table B-3 of Appendix B for explanation of source receptor nomenclature.

Lower containment bypass leakage is modeled using the unit vent atmospheric dispersion factors both before and after the establishment of annulus vacuum. Upper containment leakage is more complicated. Bypass flow could be released from the unit vent (personnel airlocks and penetrations), or the equipment hatch. Review of the Table 15 dispersion factors shows that the most conservative set of values belongs to the equipment hatch release point. The equipment hatch atmospheric dispersion factors will be used for the first 500 sccm of leakage, based upon the hatch leakage test acceptance criteria. The remainder of the upper containment leakage is modeled to be released from the unit vent.

The most conservative control room atmospheric dispersion factors (the 0-2 hour factors) are applied during the 2 hour period of the greatest amount of radioactivity release. Thus, with 0-2 and 2-8 hour factors available, the 0-2 hour factors are applied over the 2 hours of greatest release and the 2-8 hour factors over the remaining portion of the first 8 hours.

The data in Table 15 is consolidated in Table 16 to show only the dispersion value periods used in the radiological models. However, the Table 16 values have not been corrected for McGuire dual intakes and VC flow split.

Table 16
McGuire LOCA Control Room Atmospheric Dispersion Factors
(Unadjusted for Control Room Area Ventilation System Intake Flow Split, sec/m^3**)**

Time	Unit Vent	Equipment Hatch	Refueling Water Storage Tank
0 – 2 hours	1.66E-03	4.01E-03	1.83E-03
2 – 8 hours	1.32E-03	3.30E-03	1.55E-03
8 – 24 hours	6.75E-04	1.58E-03	7.60E-04
1 – 4 days	5.35E-04	1.23E-03	5.86E-04
4 – 30 days	4.05E-04	9.63E-04	4.36E-04

The bounding VC system airflow distribution assumed is 65% from the contaminated stream and 35% from the non-contaminated intake location for all normal VC configurations and alignments. Therefore, the control room atmospheric dispersion factors in Table 16 are multiplied by 65% to reflect McGuire's dual intake classification. The resulting atmospheric dispersion factors shown in Table 17 are applied in the radiological consequences model. The failure of a train of mitigation equipment (including a VC fan train) has no effect on the control room intakes. As previously discussed in Section 4.6.9, these dispersion factors are applicable to the Minimum Safeguards scenario and bound normal VC system alignments and configurations.

Table 17
Control Room Atmospheric Dispersion Model for VC Alignments
(sec/m^3, **65/35 flow split)**

Time	Unit Vent[1]	Equipment Hatch[2]	Refueling Water Storage Tank
0 – 2 hours[3]	1.08E-03	2.61E-03	1.19E-03
2 – 8 hours	8.58E-04	2.15E-03	1.01E-03
8 – 24 hours	4.39E-04	1.03E-03	4.94E-04
1 – 4 days	3.48E-04	8.00E-04	3.81E-04
4 – 30 days	2.63E-04	6.26E-04	2.83E-04

[1] Upper and lower containment bypass leakage flow path. Release point for other discharges.
[2] Upper containment bypass leakage flow path.
[3] Values to be used during 2 hour period of maximum activity release

4.7.3 Control Room In-Leakage Transport Assessment

The subject of the applicability of control room atmospheric dispersion factors to unfiltered in-leakage was raised in Reference 28 and discussed in response to RAIs for the AST Fuel Handling Accident LAR (Reference 29). The McGuire AST FHA submittal utilizes a similar approach, applying the control room dispersion factors from Table 17 to model unfiltered in-leakage.

Tracer gas testing conducted at McGuire was performed with the intent of quantifying the amount of unfiltered in-leakage to the control room. The testing was successful in achieving this objective, but it was not intended to provide information about potential flow paths associated with this in-leakage. Thus, information on potential in-leakage flow paths was not produced by the testing.

However, it is recognized that radioactive material can be introduced into the control room from intentional and unintentional pathways. Potential unfiltered in-leakage flow paths were reviewed and assessed. This review examined the assumption that the use of the dispersion factors associated with the control room intakes was limiting for McGuire control room dose calculations. The review included plant walkdowns, the use of drawings, and reviews of regulatory guidance. Alternate pathways for the flow of unfiltered in-leakage into the control room evaluated include:

- Control room access doors and the doors' seals.
- The boundary separating the outside environment from inside plant spaces.
- Control room outside air intake penetrations.
- Control Room air intake dampers.
- Intra-plant transport.

A review of the postulated flow paths associated with the accident showed that the distance from the projected main release points (equipment hatch, unit vent stack) to the control room intakes is short in comparison with other flow paths inside the Auxiliary Building. The alternate flow paths reviewed which travel inside of the QA-1, seismic Auxiliary Building on the way to the control room or the VC System would encounter additional distance as well as flow restrictions not found in the paths associated with the dispersion factors used. The alternate postulated entrances of radiation into plant structures (such as the doors from the Turbine Building to the exterior environment) are farther from the potential release points and would involve further travel along a torturous path through the buildings. These paths would be more restrictive to the transport of air than the external paths modeled in the control room dispersion factors used.

In response to the accident, the control room is automatically and quickly pressurized. This increases the pressure in the control room which will then be above the pressure in the adjacent spaces. The ability to pressurize the control room is periodically tested in accordance with Technical Specification SR 3.7.9.4. Any postulated flow into the control room would be resisted by the pressurization of the control room. Thus, in this mode of operation, the most credible flow paths into the control room would be via the

Control Room Area Ventilation System and by ingress and egress into the control room. This system has been tracer gas tested to quantify the amount of in-leakage, and an additional allowance has been made in the radiological calculations for control room ingress and egress.

In conclusion, the Control Room Area Ventilation System and its inlets provide the shortest and easiest flow path for postulated unfiltered radioactivity to enter the control room. Other potential release points were assessed; they were found to be bounded by those used in the accident model. Other flow paths through the plant were evaluated but found to be more restrictive to flow and to require a longer distance which would result in less conservative dispersion factors. Therefore, the atmospheric dispersion factors based upon the releases to the control room intakes bound those which would result from the other potential paths investigated. Thus, it can be concluded that the atmospheric dispersion factors modeled are conservative relative to those associated with the alternate paths investigated.

4.8 DOSE CONVERSION FACTORS AND PHYSICAL CONSTANTS

This submittal reports the calculation of TEDE values at the EAB, LPZ, and in the control room following the design basis LOCA accident. Physical constants such as half-lifes, branching fractions, and decay chains were taken from the libraries internal to LOCADOSE. The dose coefficients used in the analyses conform to RG 1.183. 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 30). The coefficients for Deep Dose Equivalents (DDE values) were taken from Federal Guidance Report 12 (Reference 31). These same references were employed for the approved McGuire AST FHA LAR (References 2 and 10).

4.9 RADIOLOGICAL CONSEQUENCES RESULTS

Off-site and control room radiation doses for a design basis Loss of Coolant Accident at McGuire Nuclear Station have been calculated using the Alternative Source Term and the guidance of RG 1.183. Accident progression and plant response models were discussed in detail in the preceding portions of this submittal. This analysis models releases to the annulus (containment leakage), leakage from containment directly to the environment (bypass leakage), and leakage from the ECCS system to the Auxiliary Building and to the FWST when in sump recirculation. Plant response and activity transport mitigation was provided by the Containment Spray System, the Annulus Ventilation System, the Auxiliary Building Ventilation System, and the Control Room Ventilation System. Doses were computed for receptors at the Exclusion Area Boundary, Low Population Zone, and to the control room operators.

RG 1.183 also requires that all sources of control room operator dose be included in the computation of dose to control room personnel, and lists several sources for consideration including the impact from:

- infiltration of released activity into the control room
- the infiltration of releases from adjacent structures or areas
- radiation shine from radioactive material in containment
- radiation shine from the plume
- radiation shine from activity built up on systems and components

The infiltration of releases is included in the effluent transport model as described in this submittal. Direct shine impacts were computed separately. This analysis examined and evaluated the impact of these potential sources. The total impact from direct radiation sources is shown in Table 18.

Table 18
Off-site and Control Room Doses for the McGuire Design Basis LOCA
(Rem TEDE)

	Minimum Safeguards		
	EAB	**LPZ**	**Control Room**
Containment Effluent	8.15	1.67	2.05
ECCS Effluent	1.31	0.23	0.76
Total Effluent Dose	9.46	1.90	2.81
External Operator Shine Dose			1.26
Total Dose	**9.46**	**1.90**	**4.07**
Acceptance Criteria	25	25	5

The time period of maximum release associated with the containment effluents was 0.6 - 2.6 hours.

The time period of maximum release associated with the ECCS effluents is 1.0 - 3.0 hours.

Although the periods of maximum release are not concurrent, the effluent doses in Table 18 reflect the combination of maximum release for each individual effluent release model (containment and ECCS). These are conservatively added together to produce the total effluent dose.

4.10 CONCLUSION

Radiological consequences to a design basis Loss of Coolant Accident at McGuire Nuclear Station have been computed utilizing the guidance of RG 1.183 in a manner very similar to that submitted and reviewed for Catawba Nuclear Station. The radiological consequences are:

- The Exclusion Area Boundary dose was computed to be 9.46 Rem TEDE.
- The Low Population Zone dose was computed to be 1.90 Rem TEDE.
- The control room dose was computed to be 4.07 Rem TEDE.

The computed doses for this accident at McGuire are within the acceptance criteria of 10 CFR 50.67(b)(2).

4.11 REFERENCES

1. U.S. Nuclear Regulatory Commission, RG 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, Original Issue, July 2000.
2. Letter from J. Stang (USNRC) to G. R. Peterson (Duke), *McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Implementation of Alternative Source Term Methodology (TAC Nos. MC9746 and MC9747)*, December 22, 2006 and enclosed Safety Evaluation Report.
3. Letter from G. R. Peterson (Duke) to USNRC, *Duke Energy Corporation McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 Response to NRC Generic Letter 2003-01: Control Room Habitability*, February 19, 2004.
4. Letter from E. J. Leeds (NRC) to J. W. Davis (NEI), *NEI Draft White Paper, Use of the Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability*, January 30, 2004
5. Beahm, Lorenz, Weber, NUREG/CR-5950, *Iodine Evolution and pH Control*, ORNL/TM-12242 R3, December 1992.
6. Letter from G. R. Peterson (Duke) to USNRC, *Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Bases 3.6.16 Reactor Building Technical Specification and Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specifications 5.5.11 Ventilation Filter Testing Program (VFTP)*, November 25 2002.
7. Letter from S. E. Peters (USNRC) to D. M. Jamil (Duke), *Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB7014 and MB7015)*, September 30, 2005, and attached Safety Evaluation Report.
8. Letter from L. N. Olshan (USNRC) to R. A Jones (Duke), *Oconee Nuclear Station, Units 1, 2 and 3 Re: Issuance of Amendments (TAC Nos. MB3537, MB 3538, and MB 3539)*, June 1, 2004 and attached Safety Evaluation Report.
9. Bechtel Corporation, *LOCADOSE NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation*, Version 6.0, Revision 8, February 2001.
10. Letter from G.R. Peterson (Duke) to USNRC, *License Amendment Request for Selective Implementation of Alternative Source Term and Revision to Technical Specification 3.9.4. Containment Penetrations*, December 20, 2005.
11. G.R. Peterson (Duke) to USNRC, *Duke Energy Corporation Catawba Nuclear Station Unit(s) 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Amendment for Partial Scope Implementation of the Alternative Source Term and*

Proposed Amendment to Technical Specifications (TS) 3.7.10, Control Room Area Ventilation System, TS 3.7.13, Fuel Handling Ventilation Exhaust System, and TS 3.9.3 Containment Penetrations, December 20, 2001.

12. Oak Ridge National Laboratory, ORNL/ NUREG/CSD-2/R6, *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Revision 6, NUREG/CR-0200, September 1998.

13. Pacific Northwest National Laboratory, PNNL-10521, *Atmospheric Relative Concentration in Building Wakes*, NUREG/CR-6331, Revision 1, May 1997.

14. EPRI Report NP-5561-CCML, *MULTEQ: Equilibrium of An Electrolytic Solution with Vapor-Liquid Partitioning and Precipitation, Volume 2: The Database*, Revision 3.

15. EPRI Report TR-105714, *PWR Primary Water Chemistry Guidelines*, Revision 3.

16. Letter from D. M. Jamil (Duke) to USNRC, *Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Bases 3.6.16 Reactor Building Technical Specification and Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specifications 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB 7015*, September 22, 2004.

17. Letter from W. R. McCollum (Duke) to USNRC, *Duke Energy Corporation Oconee Nuclear Station Units 1, 2, and 3 Docket Nos. 50-269, 50-270, 50-287 License NOS. NPF-38, NPF-47, and NPF-55 Licensed Amendment Request for Full-Scope Implementation of the Alternative Source Term and Technical Specifications 3.3.5, Engineered Safeguards Protective System (ESPS) Analog Instrumentation; 3.3.6, Engineered Safeguards Protective System (ESPS) Manual Initiation; 3.3.7, Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels; 3.7.10, Penetration Room Ventilation System; 3.7.17 Spent Fuel Pool Ventilation System; 3.9.3.Containment Penetrations; 5.5.2,Containment Leakage Rate Testing Program; and 5.5.12, Ventilation Filter Testing Program Technical Specification Change Number- 01-07*, October 16, 2001.

18. Postma, A. K., Sherry, R. R., and Tam, P. S., *Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels*, NUREG/CR-0009, Benton City Technology, Washington, 1978.

19. U.S. Nuclear Regulatory Commission, NUREG-0800, *Standard Review Plan*, Revision 2.

20. Letter from J.R. Morris (Duke) to USNRC, *Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and*

Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Bases 3.6.16 Reactor Building Technical Specification and Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specifications 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB 7015, July 8, 2005.

21. Letter from D. B. Vassallo (USNRC) to W. O. Parker, Jr. (Duke), *Additional Information Required During FSAR Review*, August 28, 1975.

22. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (USNRC), *Degree of Conformance to Regulatory Guides*, April 6, 1979.

23. U.S. Nuclear Regulatory Commission, RG 1.197, *Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors*, Original Issue, May 2003.

24. U.S. Nuclear Regulatory Commission, NUREG-0422, *Safety Evaluation Report by the Office of Nuclear Reactor Regulation US Nuclear Regulatory Commission in the Matter of Duke Power Company McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370,* March 1978.

25. U.S. Nuclear Regulatory Commission, Generic Letter 99-02, *Laboratory Testing of Nuclear-Grade Activated Charcoal*, June 3, 1999.

26. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.52, *Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants*, Revision 2, March 1978.

27. US Nuclear Regulatory Commission Regulatory Guide 1.194, *Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants*, Original Issue, June 2003.

28. NRC Regulatory Issue Summary 2006-04, *Experience with Implementation of Alternative Source Terms*, March 2006.

29. Letter from G. R. Peterson (Duke) to USNRC, *License Amendment Request for Selective Implementation of the Alternative Source Term and Revision to Technical Specification 3.9.4, Containment Penetrations Response to Request for Additional Information*, May 2006.

30. United States Environmental Protection Agency, Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, EPA-520/1-88-020, September 1988.

31. United States Environmental Protection Agency, Federal Guidance Report No. 12, *External Exposure to Radionuclides in Air, Water, and Soil*, EPA-402-R-93-081, September 1993.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that 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 Loss of Coolant Accident. This LAR represents full scope implementation of the AST as described in NRC Regulatory Guide 1.183. The use of AST methodology for the McGuire Fuel Handling Accident radiological analyses was approved by the NRC on December 22, 2006.

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 Loss of Coolant Accident (LOCA). This type of change is analytical, thus, does not increase the probability of an accident previously evaluated. It has been demonstrated that the dose consequences of the re-analyzed accident remain within the dose limits of 10 CFR 50.67 and Regulatory Guide 1.183.

This proposed change assumes an increase in the amount of unfiltered air in-leakage into the control room. The current Technical Information Document (TID) based McGuire dose consequence analysis for the LOCA assumed control room unfiltered in-leakage of 10 scfm. Tracer gas testing performed at McGuire revealed that unfiltered in-leakage into the control room exceeded this amount by as much as 167 scfm as discussed in McGuire's response to NRC GL 2003-01 dated February 19, 2004. Use of the AST methodology can accommodate a larger control room pressurization unfiltered in-leakage rate without exceeding any regulatory dose limits.

A comparison of the AST analysis results and the TID values (UFSAR Table 15-12) shows that the EAB and LPZ (off-site) doses decrease while the control room dose increases. The new AST based analysis not only implements changes which affect both off-site and control room doses, such as the change in source term methodology, it also includes changes to the LOCA model which only impact the control room dose, and are responsible for the increased result. These new attributes include a control room in-leakage model that reflects the control room tracer gas testing results and a recomputed control room shine component of the post LOCA control room dose. The dose consequences of the revised analysis, however, are below the 10 CFR 50.67 acceptance criteria for both off-site and control room doses and are not considered a significant increase.

AST radiological methodology does not adversely affect accident initiators or precursors. Nor will it alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an accident.

Therefore, this LAR 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. AST is an updated methodology that was used to re-evaluate the dose consequences of the McGuire UFSAR previously analyzed accidents. This new analysis does not cause any change in the post accident operation of any plant system, structure, or component.

This LAR does not involve an addition or modification to any plant system, structure, or component. This change does not affect the post accident operation of any plant system, structure, or component as directed in plant procedures. New or modified equipment or personnel failure modes that might initiate a new or different type accident are not created as a result of the proposed change.

Therefore, no new or different accident is created by changing to the AST methodology prescribed in Regulatory Guide 1.183.

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 LOCA dose consequences using AST will have no affect on the performance of these barriers. This LAR does not involve an addition or modification to any plant system, structure, or component. This change will not affect the post accident operation of any plant system, structure, or component as directed in plant procedures.

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.

5.2 APPLICABLE REGULATORY REQUIREMENTS / CRITERIA

General Design Criterion (GDC) 19, Control Room

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

GDC-19 is the current licensing basis for the McGuire control room as discussed in UFSAR Sections 3.1 and 6.4. Radiological consequences for the LOCA are currently shown to be less than the 5 rem whole body criteria, or its equivalent to any part of the body. Following approval of this LAR, the provisions of GDC-19 will continue to apply to McGuire except that control room personnel receiving radiation exposures will be limited to 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident in accordance with 10 CFR 50.67.

10 CFR 100.11, Reactor Site Criteria

> (a) As an aid in evaluating a proposed site, an applicant should assume a fission product release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:
>
> (1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
>
> (2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

Paragraphs (1) and (2) of 10 CFR 100.11(a) describe the current accident analysis dose acceptance criteria for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) for McGuire. Following approval of this LAR, the dose acceptance criteria for the EAB and LPZ will be the 25 rem TEDE criteria specified by 10 CFR 50.67.

10 CFR 50.67, Accident Source Term

A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.

The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance to licensees on acceptable applications of alternative source terms. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," describes methods acceptable to the NRC for determining atmospheric relative concentration (X/Q) values that will be used in control room radiological habitability assessments performed in support of LARs.

The McGuire LOCA described in the UFSAR was re-analyzed consistent with the guidance of Regulatory Guides 1.183 and 1.194. Using these methods, the results of the revised LOCA analyzes meet the criteria of 10 CFR 50.67 as shown in Table 5-1. These results demonstrate that the 10 CFR 50.67 dose acceptance criteria are met for the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and the Control Room. In addition, the analysis results described in Section 4, Technical Analysis, also show that the EAB and LPZ dose acceptance criteria from Regulatory Guide 1.183 Table 6 are met.

TABLE 5-1
COMPARISON OF RESULTING AST DOSES TO ACCEPTANCE CRITERIA
(Rem, TEDE)

RECEPTOR	MCGUIRE DOSE	ACCEPTANCE CRITERIA
Control Room	4.07	5
EAB	9.46	25
LPZ	1.90	25

In conclusion, based on the considerations discussed above, there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.3 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 off site, 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.

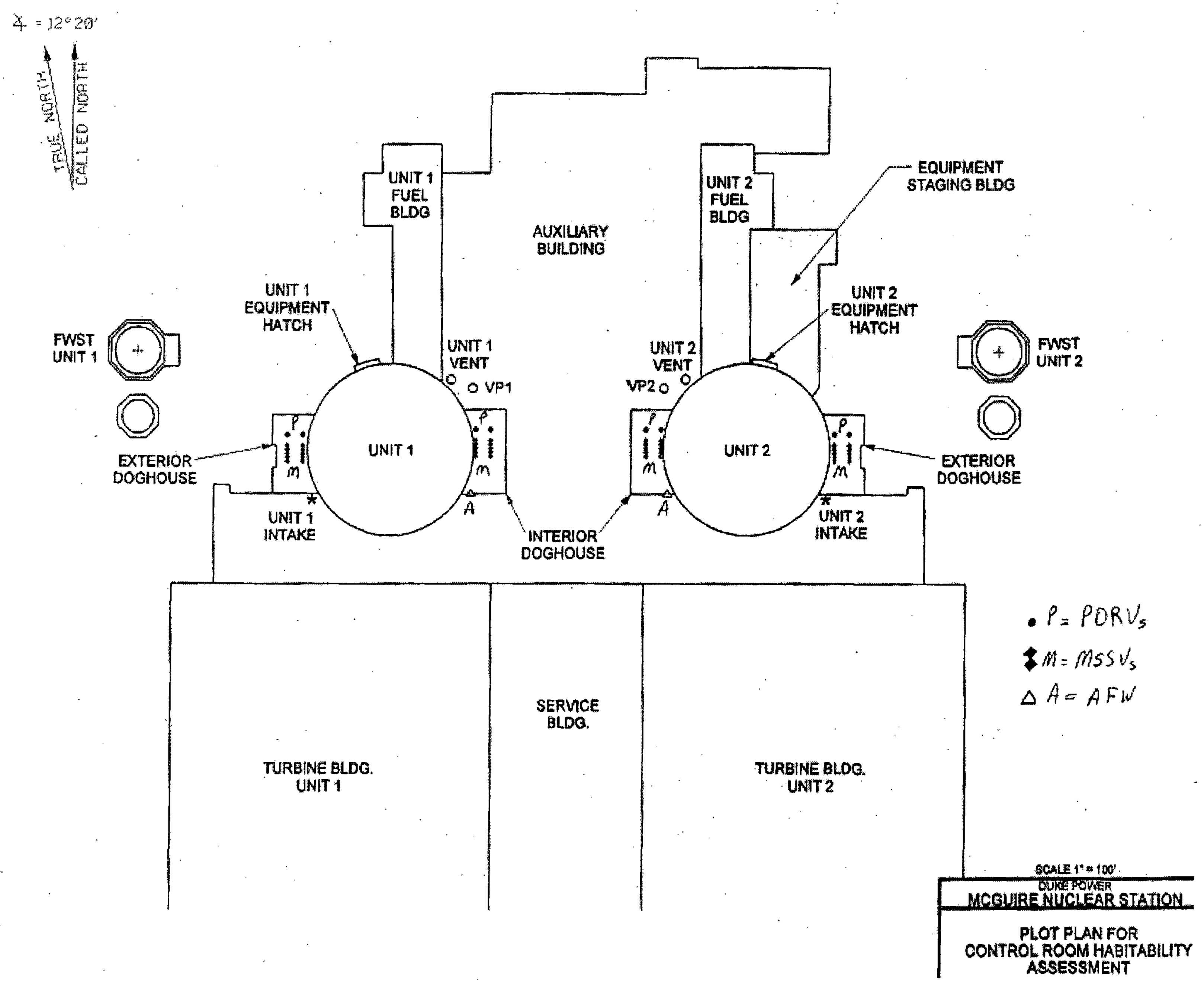
Therefore, this McGuire License Amendment Request meets the criteria provided by 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.

Section 4.0 Appendices

Appendix A:	Sketch of McGuire Nuclear Site and Potential Effluent Release Points
Appendix B:	McGuire Nuclear Station Control Room Atmospheric Dispersion Model
Appendix C:	Summary of McGuire LOCA Model Conformance to Regulatory Guide 1.183
Appendix D:	McGuire Meteorological Data Used in the Calculation of Atmospheric Dispersion Factors from the Current Meteorological Tower (2001 - 2005)

Appendix A

Sketch of McGuire Nuclear Site and Potential Effluent Release Points


∡ = 12° 20'
TRUE NORTH
CALLED NORTH
UNIT 1 FUEL BLDG
AUXILIARY BUILDING
UNIT 2 FUEL BLDG
EQUIPMENT STAGING BLDG
UNIT 1 EQUIPMENT HATCH
UNIT 2 EQUIPMENT HATCH
FWST UNIT 1
FWST UNIT 2
UNIT 1 VENT
VP1
UNIT 2 VENT
VP2
UNIT 1
UNIT 2
EXTERIOR DOGHOUSE
EXTERIOR DOGHOUSE
P
M
A
UNIT 1 INTAKE
UNIT 2 INTAKE
INTERIOR DOGHOUSE
TURBINE BLDG. UNIT 1
SERVICE BLDG.
TURBINE BLDG. UNIT 2
P = PORVs
M = MSSVs
A = AFW
SCALE 1" = 100'
DUKE POWER
MCGUIRE NUCLEAR STATION
PLOT PLAN FOR CONTROL ROOM HABITABILITY ASSESSMENT

Appendix B

McGuire Nuclear Station Control Room Atmospheric Dispersion Model

BACKGROUND AND HISTORICAL INFORMATION

In the SER for the McGuire AST FHA LAR (Reference 2), the NRC Staff expressed concern with the control room χ/Q values submitted, particularly those associated with later time frames. Since the FHA is a short duration release, the short term dispersion factors used in that analysis were accepted, as the values supplied were "similar" to the staff calculated values. However, the staff had concerns about the conservatism of the values beyond the two hour FHA release. These concerns were related to the use of data from both the current McGuire meteorological tower and the previous meteorological tower. These concerns were discussed in Section 3.3.1 of Reference 2 and summarized in Section 4.0 which stated the following:

> *The NRC staff accepts the use of the licensee's 0-2 hour control room x/Q values for both the unit vent and equipment hatch release pathways for the design-basis accidents analyzed in this amendment application. However, the 0-2 hour x/Q values, along with the 0-8 hour, 8-24 hour, 1-4 day, and 4-30 day control room atmospheric dispersion factors presented in Table 5.1 of the licensee's December 20, 2005, submittal, are not acceptable for use in any other DBAs control room dose analyses. These x/Q values should be regenerated using a minimum of 3 years of data collected by the new meteorological monitoring program before being used in any other design-basis control room dose analysis.*

Duke has since regenerated the McGuire control room atmospheric dispersion modeling using five years of data from the current tower location, resulting in revised dispersion factors which are used in this submittal. The entire population of resultant control room design basis accident analysis dispersion factors (and their associated input data) is included in this appendix. The associated meteorological data is included in Appendix D. It is Duke's intention to utilize and apply these factors in the future and as AST is fully implemented.

METHODOLOGY AND DATA

The McGuire control room atmospheric dispersion analysis calculated the χ/Q factors associated with flow from the various release points to each set of control room intakes. McGuire contains a single control room which houses the controls for both units, but the ventilation system has an inlet location associated with Unit 1 and an inlet location associated with Unit 2. Both inlet locations have two intakes. Thus, source/receptor combinations are examined between release points and intake locations associated with the same unit (intra-unit) and flow paths associated with release locations and control room intakes associated with different units (inter-unit). The values calculated are source to inlet point values, so any further adjustments for Control Room Ventilation System alignments or configurations are not included. Those adjustments are applied as part of the radiological consequences analysis. Similarly, no adjustments for McGuire's status as a dual intake plant have been made in the derivation of these factors. The source locations modeled (and their modeling abbreviations) are:

1. Equipment Hatch (EQ)
2. Fuel Building (FUEL)
3. Inboard Doghouse (NDOG)
4. Inboard Doghouse plus Steam & Feedwater Line Penetrations (VNDOG)
5. Outboard Doghouse (ODOG)
6. Outboard Doghouse plus Steam & Feedwater Line Penetrations (VODOG)
7. Reactor Building Surface (RX)
8. Refueling Water Storage Tank (FWST)
9. Unit Vent - with VA flow rate (UV)
10. Containment Purge Supply intake vents (VP)
11. Steam & Feedwater Line Penetrations of Inboard Doghouse — horizontal point source (AGIN)
12. Steam & Feedwater Line Penetrations of Outboard Doghouse — horizontal point source (AGOUT)
13. Steam Generator Power Operated Relief Valves on Inboard Doghouse (PORVin)
14. Steam Generator Power Operated Relief Valves on Outboard Doghouse (PORVout)
15. Main Steam Safety Valves on Inboard Doghouse (MSSVin)
16. Main Steam Safety Valves on Outboard Doghouse (MSSVout)
17. Turbine Driven Auxiliary Feedwater Pump Exhaust Vents (AFW)

The dispersion model was performed in accordance with RG 1.194 (Reference 27). ARCON96 is used for all combinations of sources and receptors except when the distance separating them is less than 10m. In those situations, χ/Qs are calculated using the Murphy-Campe methodology. All of the dispersion factors used in the McGuire AST LOCA analysis are derived using ARCON96. All sources are treated as "ground-level" releases (vertical velocity is 0 m/sec and stack radius equals zero in ARCON96). A sketch of the McGuire site with release locations is shown in Appendix A.

The values used in the LOCA modeling represent the most conservative results of any and all of the cases (inter-unit and intra-unit) associated with source. Table B-1 provides a summary of the common parameters for the ARCON96 models for the various release paths. Table B-2 provides source specific parameters.

RESULTS

Table B-3 presents the maximum χ/Q for each source location for several time periods during a 30 day event duration. Each χ/Q represents the largest value from all of the cases computed for both inter-unit and intra-unit releases. These values are also maximized over all cases where the source was modeled as more than one source type. Each represents the largest dispersion factor (least amount of dispersion) from all of those associated cases. This table forms the basis for the control room atmospheric dispersion modeling for the release points used in the McGuire AST LOCA analysis as discussed in Section 4.7.2 in the body of this LAR.

Table B-1
Summary of ARCON96 Default Inputs for McGuire

Parameter	Default Values Used
Surface Roughness Length	0.2 m
Wind Direction Window (degrees)	90 degrees
Minimum Wind Speed (m/s)	0.5 m/s
Averaging Sector Width Constant	4.3
Initial Diffusion Coefficients	Source-specific. See Table B-2
Hours in Averages	1, 2, 4, 8, 12, 24, 96, 168, 360, 720
Minimum Number of Hours	1, 2, 4, 8, 11, 22, 87, 152, 324, 648

Table B-2 McGuire Source Parameters for ARCON96 Modeling (sheet 1of 3)

Source Type:	**EQ[(a)]**	**FUEL**	**NDOG**	**Vndog**	**ODOG**	**VOdog**	**RX[c]**	**FWST**	**UV**	**VP**
Vertical Point Source									X	
Horizontal or Capped Point	1-CR1; 2-CR2 1-CR2; 2-CR1					1-CR2 2-CR1		X		X
Horizontal Area Source		X	X		X					
Vertical Area Source	1-CR1 2-CR2			X		1-CR1 2-CR2	X			
Release Height	8.3 m; 0m PTM 0 m VAS	19.8 m	18.6 m	0 m	18.6 m	18.6 m PT 0 m VAS	0 m	14.6 m	40.2 m	9.5 m
Flow Rate (m^3/s)	0	0	0	0	0	0	0	0	8.60 m^3/s	0
Sigma-Y	Pt. Src. 0 m Area Src. 1 m	3.2 m	1.9 m	1.9 m	1.6 m	0 m 1.6 m	3.3 m	0 m	0 m	0 m
Sigma-Z	Pt. Src. 0 m Area Src. 1 m	3.2 m	3.1 m	1 m	3.1 m	0 m 1 m	6.8 m	0 m	0 m	0 m
Bldg Cross-sectional Area *(RX or ODOG)*	1588 m^2	1588 m^2	1588 m^2	1588 m^2	opposite unit: 1588 m^2 same unit: 188.1 m^2	opposite unit: 1588 m^2 same unit: 188.1 m^2	1588 m^2	opposite unit: 1588 m^2 same unit: 188.1 m^2	1588 m^2	1588 m^2
Source/Stack Radius (m) [(b)]	0 m	0 m	0 m	0 m	0 m	0 m	0 m	0 m	0 m	0 m
Vertical Velocity[(b)]	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s
1-CR1 Distance_WD	36 m_{arc} 32° arc	65 m 24°	54 m_{arc} 102° arc	54 m_{arc} 102° arc	11 m 349°	11 m 349°	2.1 m 71°	54 m 321°	43 m 62°	59 m_{arc} 108° arc
1-CR2 Distance_WD	116 m 298°	112 m 318°	82 m 290°	82 m 290°	128 m 287°	128 m 287°	85 m 289°	169 m 294°	94 m 299°	88 m 298°
2-CR1 Distance_WD	137 m 76°	112 m 68°	82 m 94°	82 m 94°	128 m 102°	128 m 102°	85 m 95°	169 m 90°	94 m 83°	88 m 83°
2-CR2 Distance_WD	61 m_{arc} 10° arc	65 m 1°	54 m_{arc} 281° arc	54 m arc 281° arc	11 m 32°	11 m 32°	2.1 m 313°	54 m 62°	43 m 323°	59 m_{arc} 265° arc

Table B-2 McGuire Source Parameters for ARCON96 Modeling (sheet 2 of 3)

Source Type:	AGin	AGout	PORVin	PORVout	MSSVin	MSSVout[c]	AFW[d]
Vertical Point Source			X	X	X	X	
Horizontal or Capped Point	X	X					X
Horizontal Area Source							
Vertical Area Source							
Release Height	8.5 m	8.53 m	18.9 m	18.9 m	18.8 m	18.8 m	15.5 m
Flow Rate (m^3/s)	0	0	0 m^3/s	0 m^3/s	0 m^3/s	0 m^3/s	0 m^3/s
Sigma-Y	0 m	0 m	0 m	0 m	0 m	0 m	0 m
Sigma-Z	0 m	0 m	0 m	0 m	0 m	0 m	0 m
Bldg Cross-sectional Area *(RX or ODOG)*	1588 m^2	opposite unit (Rx): 1588 m^2 same unit (Odog): 188.1 m^2	1588 m^2	Opposite unit: 1588 m^2 Same unit: 188.1 m^2	1588 m^2	Opposite unit: 1588 m^2 Same unit: 188.1 m^2	1588 m^2
Source/Stack Radius [(b)] (m)	0 m	0 m	0 m	0 m	0 m	0 m	0 m
Vertical Velocity [(b)]	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s	0 m/s
1-CR1 Distance_WD	42 m_{arc} 80 ° arc	16 m 312 °	43 m 79°	15 m 4°	40 m 88°	9 m [c] 357°	44 m_{arc}, 65° arc[d] or (38m, 100°) straightline
1-CR2 Distance_WD	81 m 284 °	127 m 284°	82 m 293°	126 m 289°	81 m 289°	125 m 286°	85 m 283°
2-CR1 Distance_WD	81 m 102 °	127 m 102 °	82 m 91°	126 m 95°	81 m 95°	125 m 97°	85 m 101°
2-CR2 Distance_WD	42 m_{arc} 300 °arc	16 m 85 °	43 m 304°	15 m 20°	40 m 291°	9 m [c] 21°	44 m_{arc}, 317° arc [d] or (38m, 284°) straightline

Table B-2 McGuire Source Parameters for ARCON96 Modeling (sheet 3 of 3)

Notes:

(a) Three source type runs were made for the equipment hatch: point source and receptor on the same unit; point source and receptor on opposite units; and vertical area source and receptor on the same unit. The limiting case was with a point source and receptor on the same unit, representing a single hole in the hatch. The vertical area source run represents either having the hatch doorway open, or having multiple holes in the hatch.

(b) Values of zero are assumed for the vertical velocity and the stack radius parameters, to treat the release as a ground-level release in ARCON96. Plume rise of the PORV and MSSV sources is NOT accounted for.

(c) Used Murphy-Campe hand-calculation, without wind speed/direction factors, for RX and MSSVout sources on same unit as receptor intake because the separation distance is less than 10m.

(d) Selected maximum χ/Q for each time period for the AFW sources, comparing ARCON96 runs for straight line distances and arclength distances (i.e. around the reactor building).

Table B-3 McGuire Maximum (χ/Q)s per Source (McGuire 2001-2005 Meteorology)

	0-2 Hr	0-4 Hr	0-8 Hr	2-8 Hr	4-8 Hr	8-24 Hr	1-4 day	4-30 day
EQ M1EQ1PTM (arc)	4.01E-03	3.73E-03	3.48E-03	3.30E-03	3.23E-03	1.58E-03	1.23E-03	9.63E-04
FUEL M1FUEL1	7.88E-04	7.08E-04	6.55E-04	6.11E-04	6.02E-04	2.83E-04	2.36E-04	1.88E-04
NDOG	1.13E-03 M2NDOG2 (arc)	9.13E-04 M2NDOG2 (arc)	8.08E-04 M1NDOG1 (arc)	7.23E-04 M1NDOG1 (arc)	7.06E-04 M1NDOG1 (arc)	2.98E-04 M1NDOG1 (arc)	2.42E-04 M2NDOG2 (arc)	1.93E-04 M2NDOG2 (arc)
VNDOG	1.52E-03 M2VNDOG2 (arc)	1.46E-03 M1VNDOG1 (arc)	1.27E-03 M1VNDOG1 (arc)	1.14E-03 M1VNDOG1 (arc)	9.06E-04 M1VNDOG1 (arc)	6.61E-04 M1VNDOG1 (arc)	3.22E-04 M2VNDOG2 (arc)	2.63E-04 M2VNDOG2 (arc)
ODOG M2ODOG2	4.88E-03	4.30E-03	3.97E-03	3.67E-03	3.64E-03	1.92E-03	1.50E-03	1.19E-03
VODOG M2VODOG2	1.70E-02	1.49E-02	1.39E-02	1.28E-02	1.29E-02	6.16E-03	5.18E-03	4.00E-03
RX- *Murphy-Campe* M1RX1; M2RX2	1.26E-3	1.26E-3	1.26E-3	1.26E-3	1.26E-3	1.26E-3	1.26E-3	1.26E-3
FWST M2FWST2	1.83E-03	1.74E-03	1.62E-03	1.55E-03	1.50E-03	7.60E-04	5.86E-04	4.36E-04
UV M1UV1	1.66E-03	1.47E-03	1.41E-03	1.32E-03	1.35E-03	6.75E-04	5.35E-04	4.05E-04
VP M2VP2 (arc)	1.56E-03	1.40E-03	1.29E-03	1.20E-03	1.18E-03	5.15E-04	4.11E-04	3.36E-04
AGIN M1AGIN1 (arc)	2.90E-03	2.78E-03	2.58E-03	2.47E-03	2.38E-03	1.15E-03	8.86E-04	6.53E-04
AGOUT M2AGOUT2	1.73E-02	1.61E-02	1.48E-02	1.40E-02	1.35E-02	6.33E-03	4.92E-03	3.54E-03
PORVin M1PORVn1	2.57E-03	2.35E-03	2.18E-03	2.05E-03	2.01E-03	9.61E-04	7.54E-04	5.54E-04
PORVout M2PORVo2	1.06E-02	9.69E-03	8.90E-03	8.33E-03	8.11E-03	3.76E-03	3.02E-03	2.42E-03
MSSVin M1MSSVn1	2.87E-03	2.59E-03	2.40E-03	2.24E-03	2.21E-03	1.02E-03	7.75E-04	5.62E-04
MSSVout *Murphy-Campe* M1MSSVo1;M2MSSVo2	1.26E-03	1.26E-03	1.26E-03	1.26E-03	1.26E-03	1.26E-03	1.26E-03	1.26E-03
AFW M1AFW1a (arc)	2.59E-03	2.41E-03	2.25E-03	2.14E-03	2.09E-03	1.06E-03	8.13E-04	6.09E-04

Receptor/source nomenclature lists the receptor first, followed by source. For example, "M1UV1" represents a release from the Unit 1 unit vent ("UV1") to the McGuire unit one control room area ventilation system inlet location ("M1").

Appendix C

Summary of McGuire LOCA Model Conformance to RG 1.183

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
3.1	Inventory of fission products based upon: • full power operation including current licensed values for fuel enrichment, burn-up, power uncertainty • irradiation period should allow for maximum activities (or end of life) • Depletion code such as ORIGEN-ARP • TID-14844 (and other source) core inventory factors not recommended.	**Compliance.** The source term is based on full power plus a 2% thermal power uncertainty. It bounds the currently licensed limit of for burn-up and enrichment and is based on end of cycle isotopics. SCALE/SAS2H (ORIGEN-S) were used for the depletion models. ORIGEN-ARP is a derivative SCALE/SAS2H/ORIGEN-S. No core inventory factors were used.
3.1	For the DBA LOCA, all fuel assemblies are affected and the core averaged inventory is used.	**Compliance.** The source term is based on a full core inventory.
3.2	Release fractions	**Compliance.** The release fractions in RG Table 2 are used without modification. The release phases are adopted. These fractions are applied to fuel burned up to 62 GWd/MTU.
3.3	Timing of releases. Activity can be modeled as a ramp over the release time.	**Compliance.** RG Table 4 timing is adopted without modification. Literal adoption of this table results in a 30 second overlap period between gap release and early in-vessel releases which is explicitly modeled. Activity is released linearly over the release time durations.
3.4	Radionuclide composition	**Compliance.** Table 5 of the RG lists the groups by elements. A list of all of the isotopes associated with these elements numbers in the hundreds. Isotopes that are not dose significant are not included in the source term to make the list more manageable and to align with the LOCADOSE library. Evaluation of this list confirms that these are not significant due to the dose conversion factor and/or the half life. Final inventory includes almost 100 isotopes.
3.5	Chemical form of radioiodine released from RCS to containment is 95% particulates, 4.85% elemental, and 0.15% organic.	**Compliance.** These fractions are used for the containment release model. Further discussion is provided with the RG 1.183 Appendix A.2 requirement below.

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
4.1.1	Dose calculations should be in TEDE. All decay progeny which are dose significant should be included.	**Compliance.** Off-site and control room dose results are computed in TEDE. Daughter products are included in the computations for the containment and ECCS transport models and releases.
4.1.2	Inhalation dose conversion factors should be ICRP-30 based. FGR 11 is acceptable to use.	**Compliance.** FGR 11 is used for inhalation dose conversion factors.
4.1.3	Off-site receptor breathing rate model.	**Compliance.** This breathing rate model is adopted as specified.
4.1.4	Off-site DDE should be calculated assuming submergence in a semi-infinite cloud. Dose conversion factors from FGR 12 are acceptable.	**Compliance.** LOCADOSE calculates doses based on the semi-infinite cloud model. It does have a feature by which a finite cloud correction factor can be entered. This feature was not used. DCFs were specified from FGR 12.
4.1.5	EAB TEDE should be computed over the largest two hour release period to produce the largest two hour dose.	**Compliance.** Both the containment release model and the ECCS release model were computed to determine the maximum two hour EAB dose period for each release path. Even though these paths are not concurrent, these maximum two hour doses were used in the computation of the final result. This methodology is conservative compared to a determination of what time frame would produce the greatest sum from the two release paths.
4.1.6	Compute TEDE dose for limiting receptor at the LPZ.	**Compliance.** LPZ doses to the limiting receptor were computed in TEDE.
4.1.7	No credit should be taken for plume deposition in transit.	**Compliance.** No credit was taken for plume deposition during transit.

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
4.2.1	Control room TEDE dose should include all sources of exposure including: • Contamination of the control room atmosphere from contaminated air intake • Contamination of the control room atmosphere from unfiltered in-leakage • Radiation shine from the external plume • Radiation shine from radioactive material from containment • Radiation shine from systems/components/filters external to the control room	**Compliance.** The impact from each of these possibilities is addressed in either the LOCA effluent analysis (first two items) or the control room shine analysis (last three items). These are included in the computation of the control room operator TEDE dose.
4.2.2	Use same source term model, transport models, and release assumptions for control room and off-site dose computations.	**Compliance.** The control room and off-site effluent doses were computed using the same computer model. The LOCADOSE transport and dose input decks are set up to produce dose results from EAB, LPZ and control room from a single model (for each release path).
4.2.3	The effluent and shine models should be made suitably conservative.	**Compliance.** As discussed in the body of the LAR, many instances and levels of conservatism were included in the effluents models. The activity transport models used to produce the source term activities for the shine models were adapted from the effluent models. The shine models and inputs were reviewed to ensure conservative shine doses.
4.2.4	Credit for control room engineered safety features may be taken in the operator dose models.	**Compliance.** Credit is taken for the automatic start of control room pressurization and filtered outside airflow. No credit is taken for any radiation monitor based action, automatic or manual, as McGuire is classified as a "dual intake" plant without manual or automatic actions.
4.2.5	Credit should not be taken for the use of personal protective equipment or prophylactic drugs.	**Compliance.** No credit is taken in the McGuire AST LOCA analysis for external personal protective equipment or prophylactic drugs including breathing apparatuses and KI tablets.

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
4.2.6	Control room dose receptor occupation and breathing model.	**Compliance.** The occupation factors and breathing rates were included in the analysis model without modification.
4.2.7	Use DCFs from position 4.1 for control room dose computations. Finite cloud correction is allowed for control room doses.	**Compliance.** DCFs from FGR 11 were used. LOCADOSE computes control room dose with the same geometry correction model provided in this regulatory position.
4.4	Acceptance Criteria.	**Compliance.** LOCA acceptance criteria from 10 CFR 50.67(b)(2) (and RG 1.183 Table 6) of 25 Rem TEDE is used as the acceptance criteria for off-site doses and the 5 Rem TEDE is used for the control room doses.
A.2	If pH of the sump is controlled at 7 or greater, iodine species can be modeled as 95% particulate, 4.85% elemental, and 0.15% organic.	**Compliance.** Both the minimum and the equilibrium sump pH (to standard conditions) are predicted to remain above 7 during a LOCA, so these iodine specie fractions are adopted for the containment releases.
A3.1	The release should be assumed to mix instantaneously and homogeneously throughout the free air volume of primary containment. This distribution should be adjusted if there are internal compartments with limited ventilation exchange. The release into containment should be assumed to terminate at the end of the early in-vessel release phase.	**Compliance.** Due to the separation of the operating deck and the lack of forced air flow prior to VX start between upper and lower containment, the initial release is confined to only lower containment. The release is instantaneous and homogeneous in lower containment. This is more conservative than an instantaneous and homogeneous release to the full containment volume since initially the concentration of radionuclides in lower containment will be much greater, resulting in the release of more activity per leakage volume. The model terminates the release of activity into containment at the end of the early in-vessel release phase.
A3.2	Natural deposition may be credited.	**Compliance. Natural deposition not credited.** It is not required to include natural deposition and Duke has opted to not take mitigation credit for natural processes including natural deposition. Not crediting this potential mitigation process increases the conservatism of the analysis since it results in increased releases.

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
A3.3	Containment spray removal may be credited. Should include: • Areas not covered by spray and the mixing rate between them by natural convection • Particulate iodine removal rates should be reduced by a factor of 10 when DF reaches 50. • Elemental iodine removal credited until maximum DF from SRP reached.	**Compliance.** The McGuire spray system achieves full spray coverage under minimum safeguards. The model removes credit for spray which could impact containment walls prior to falling to the operating deck. Spray flow is reduced to reflect the minimum permitted diesel generator frequency and voltage. Particulate iodine spray lambdas are reduced by a factor of 10 when a DF of 50 is first reached. Elemental iodine removal ceases when its DF reaches 200. Spray system is credited for the first 24 hours after the accident, although credit was not begun until the VX system starts 10 minutes into the accident.
A3.4	Reduction in airborne radioactivity from in containment recirculation filter systems may be credited. Revised filter loading should be addressed.	**Compliance. Filtration does not exist on VX system.** The Containment Air Return System, which transports air between upper and lower containment, has no filtered component and no filter credit is taken.
A3.5	BWR requirements.	N/A McGuire is a pressurized water reactor (PWR).
A3.6	Reduction in airborne radioactivity by ice condensers or other means may be considered.	**Compliance. Ice condenser mitigation not credited.** McGuire is an ice condenser containment plant. Mitigation of elemental iodine by the ice has previously been taken. However, due to the reduced emphasis on elemental iodine in the new source term, removal by ice is not credited. It is felt that similar credit could be taken for particulate removal by the ice based upon NUREG/CR-5768, *Ice Condenser Aerosol Tests*, but currently McGuire is not crediting any iodine mitigation by the ice condensers.
A3.7	Primary containment leakage model.	**Compliance.** The McGuire analysis models containment leakage at the full Technical Specification rate (L_a) at full containment pressure (P_a) for the first 24 hours after the accident and then the rate is cut in half.

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
A3.8	If containment is routinely purged, then releases from this system should be included for dose impact.	**Compliance. McGuire does not purge in response to a LOCA.** McGuire does not use its Containment Purge System in response to a LOCA. During normal operation (modes 1-4) the purge supply and exhaust valves are sealed closed per Technical Specification SR 3.6.3.1. Additionally, the Containment Air and Release System (VQ) was also evaluated in the LOCA analysis for similar impact. The potential release from this system was very small, and completed well before the onset of gap release. Its impact is not significant.
A4.1	Collect leakage from the primary containment into the secondary containment and process with ESF filters during period the secondary containment has a negative pressure. Credit for elevated release can be taken.	**Compliance.** Prior to the establishment of annulus vacuum, all containment leakage is modeled as bypass leakage. After the establishment of annulus vacuum, 93% of containment leakage (L_a at P_a) is modeled to flow to the annulus where it is processed (filtered) by the Annulus Ventilation System and released. This system discharges from the unit vent which is modeled as a ground release.
A4.2	Leakage from primary containment is released directly to the environment as a ground release when a negative pressure has not been established in the annulus.	**Compliance.** Prior to the establishment of annulus vacuum, all containment leakage is modeled as bypass leakage. After the establishment of annulus vacuum, 93% of containment leakage (L_a at P_a) is modeled to flow to the annulus where it is processed (filtered) by the Annulus Ventilation System and released. The release points for containment leakage are modeled as ground releases.

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
A4.3	The effect of high wind speeds on the ability of secondary containment to maintain a negative pressure should be evaluated on an individual case basis. Wind speed used should be a one hour average that is only exceeded 5% of the time. Ambient temperatures used should be one hour average which is exceeded only 5% and is conservative for the application	**Exception taken on wind speed requirement.** **Compliance with temperature requirement.** The effect of high wind speed on the ability of the Annulus Ventilation System to maintain a negative pressure in the annulus is small compared to its larger effect of reducing the χ/Q. Low wind speeds are used in the computation of the atmospheric dispersion factors. The conservatism introduced by the use of low wind speeds in computing the dispersion factors is more significant and easily compensates for neglecting high wind speed effect on Annulus Ventilation System performance. ARCON96 produces 95th percentile χ/Qs which are used in the radiological consequences analysis. This position is identical to that taken in the Catawba AST LOCA submittal. The 95% low temperature (18 °F) for the Charlotte region was used in modeling the Annulus Ventilation System post accident response. Since Catawba and McGuire share the same meteorological region, Charlotte, the same low temperature was used for both plants.
A4.4	Secondary containment dilution may be credited where supported and generally limited to 50%.	**Compliance.** Credit for annulus mixing is taken in the McGuire AST LOCA models. The configuration of the locations of the majority of the penetrations in lower containment along with the small width of the space and the location of the Annulus Ventilation System discharges low in the annulus and the return header of the Annulus Ventilation System high in the annulus all serve to promote mixing in the annulus of at least 50%.
A4.5	Bypass leakage should be evaluated at the Technical Specification rate.	**Compliance.** The full Technical Specification rate of 7% L_a is assigned to bypass leakage which is not processed by the Annulus Ventilation System. It is released directly to the environment without mitigation. No credit is taken for any removal mechanism associated with the fluid inside the pipes and ducts or the pipes and ducts themselves penetrating containment (postulated leakage paths).

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
A4.6	Radioactive material releases from secondary containment may be reduced by ESF filter systems which meet guidance of RG 1.52 and GL 99-02.	**Compliance.** Credit is taken for filtration of the Annulus Ventilation System recirculation and exhaust. These filters are tested in accordance with RG 1.52, revision 2 as required in Technical Specifications as part of the Ventilation Filter Testing Program. McGuire responded to GL 99-02 in November of 1999 and an SER was issued in November 2000. The Technical Specification requirements for penetration and bypass are combined with a safety factor of 2 to compute the filter efficiency of this (and other) filters using the methodology of RG 1.52 revision 2 and GL 99-02 to meet expectations for filter efficiency computations communicated during the review of the Catawba AST LOCA LAR by NRC Staff.
A5	Radiological consequences from ESF leakage should be analyzed and combined with the consequences from other release paths.	**Compliance.** The results from the containment and ECCS leakage effluents releases are combined to compute the total off-site dose consequences. The containment and ECCS effluent results are also added to the direct shine impact for the total control room operator dose consequence.
A5.1	With the exception of noble gases, all fission products released from the fuel to containment should be assumed to be instantaneously and homogeneously released to the sump at the time of release from the core.	**Compliance.** The same source term released to the containment atmosphere (with the exception of noble gases) is released to the sump with the same timing. The particulates are modeled to remain in solution in the sump water per A5.3.
A5.2	The ESF leakage modeled should be twice the permitted operational leakage rate. Releases should start at the earliest time recirculation flow occurs and end at the latest time releases from these systems are terminated.	**Compliance.** The modeled ESF leakage rate is twice the operational leakage rate. It begins as soon as sump recirculation flow beings in the accident response scenario. The leakage is modeled to occur for the duration of the accident.
A5.3	With the exception of iodine, all radioactive materials are retained in the sump water.	**Compliance.** Only iodines are released from ECCS leakage. Duke also includes the effect of iodine precursors to include iodine produced during the progression of the accident from decay of tellurium.
A5.4	If the temperature of the released sump fluid exceeds 212 °F, the flashing fraction can be calculated using a constant enthalpy process.	**Compliance.** The McGuire sump temperature is not predicted to exceed 212 °F during sump recirculation.

RG 1.183 Position	Synopsis of Requirement	McGuire LOCA Analysis Compliance Discussion
A5.5	If the temperature of the released sump fluid is less than 212 °F, partitioning of the iodine released should be 10% of the iodine activity unless a smaller amount can be justified.	**Compliance. Option to derive alternate model utilized.** McGuire has adopted the time dependent partitioning model used by Catawba in its AST LOCA submittal. This model is actually slightly conservative for McGuire which has a more favorable pH profile. This methodology is applied to ECCS releases from the Auxiliary Building and from the Refueling Water Storage Tank. The same values used for the approved Catawba analysis and AST submittal for releases to both locations are used for the McGuire analysis.
A5.6	Radioiodine available for release is to be 97% elemental and 3% organic. Reduction in released activity by dilution or holdup or filtration is permitted for systems which meet guidance of RG 1.52 and GL 99-02.	**Compliance.** Iodine specie model is adopted without modification. No credit is taken for holdup or dilution in either the Auxiliary Building or the FWST. The Auxiliary Building filters are credited to mitigate the release of ECCS leakage from the Auxiliary Building. These filters are tested in accordance with RG 1.52, revision 2 as required in Technical Specifications as part of the Ventilation Filter Testing Program. McGuire responded to GL 99-02 in November of 1999 and an SER was issued in November 2000. The Technical Specification requirements for penetration and bypass are combined with a safety factor of 2 to compute the filter efficiency of this (and other) filters using the methodology of RG 1.52 revision 2 and GL 99-02 to meet expectations for filter efficiency computations communicated during the review of the Catawba AST LOCA LAR by NRC Staff.
A7	The radiological consequences from containment purging should be analyzed. If containment purging capabilities are maintained for severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated.	**Compliance. McGuire does not purge in response to a LOCA.** McGuire does not use its Containment Purge System in response to a LOCA. During normal operation (modes 1-4) the purge supply and exhaust valves are sealed closed per Technical Specification SR 3.6.3.1. Neither this system nor any of its mitigation capabilities are modeled or credited in the LOCA analysis.

Appendix D

McGuire Meteorological Data Used in the Calculation of Atmospheric Dispersion Factors from the Current Meteorological Tower (2001 - 2005)

The attached compact disc (CD) contains McGuire meteorological data for each year from 2001 through 2005. Five data files are included. Each file corresponds to the year indicated in the file name. These files (*.met) are text files. They can be opened into any standard text, spreadsheet, or word processing software.