JAN 2 0 2004

L-PI-04-001
10 CFR 50.90
10 CFR 50.67

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
LICENSE Nos. DPR-42 AND DPR-60
LICENSE AMENDMENT REQUEST (LAR) DATED JAN 2 0 2004
SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO CONTAINMENT TECHNICAL SPECIFICATIONS

Attached is a request for change to the Technical Specifications (TS), Appendix A of the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. The Nuclear Management Company, LLC (NMC) submits this request in accordance with the provisions of 10 CFR 50.90 and 10 CFR 50.67.

This LAR proposes selective scope application of the alternate source term (AST) for the fuel handling accident (FHA) in accordance with the provisions of 10 CFR 50.67. NMC requests the NRC review and approve the AST FHA methodology for application to the Prairie Island Nuclear Generating Plant. This LAR also proposes revisions to TS associated with ensuring that safety analyses assumptions are met for a postulated FHA in containment. Based on the AST FHA analyses, this LAR proposes to modify TS 3.9.4, "Containment Penetrations", to apply during the handling of recently irradiated fuel and require all containment penetrations to be closed during handling of recently irradiated fuel. Since all containment penetrations are required to be closed during handling of recently irradiated fuel, the requirements of TS 3.3.5, "Containment Ventilation Isolation Instrumentation" relating to movement of irradiated fuel assemblies are proposed to be removed from TS. Bases changes are also proposed that support the proposed TS changes.

This change implements a portion of TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations" as it applies to TS 3.9.4. Consistent with TSTF-51, NMC commits to the guidelines of TSTF-51 Reviewer's Note for the assessment of systems removed from service during movement of irradiated fuel at PINGP. Specifically, the guidelines of NUMARC 93-01, Revision 3, Section 11.3.6,

Exhibit G Contains Proprietary Information

AP01

"Assessment Methods for Shutdown Conditions," Subsection 11.3.6.5, that will be adopted are:

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

Exhibit B contains the licensee's evaluation of this LAR. Exhibit C presents the TS and Bases pages marked up to show the proposed changes. Exhibit D presents the revised TS and Bases pages incorporating the proposed changes. Exhibit E provides the commitments made in this LAR. Exhibit F provides the safety analysis for adoption of the AST methodology. Exhibit G and Exhibit H provide Figures referenced in Exhibit F. Exhibit I provides draft containment controls which will implement the commitment to assess systems removed from service during movement of irradiated fuel. Changes to these draft containment controls will be made under plant processes.

Also provided, as Exhibit A, is an affidavit for information to be withheld from public disclosure. As Exhibit G contains information to be withheld from public disclosure according to 10 CFR 2.790 (d)(1), it is supported by an affidavit signed by NMC, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly it is respectfully requested that the information in Exhibit G be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations. A non-proprietary version of the information is provided in Exhibit H.

A compact disk (CD) has been provided as an enclosure to this submittal. This CD contains meteorological data in the ARCON96 format to facilitate NRC review of this submittal.

Exhibit G Contains Proprietary Information

NMC requests NRC review and approval of the proposed TS changes for the Prairie Island Nuclear Generating Plant by September 10, 2004, to support replacement of the Unit 1 steam generators.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments to the designated State Official.

Please address any comments or questions regarding this LAR to Mr. Dale Vincent at 1-651-388-1121.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on
JAN 2 0 2004

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant

CC Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
Glenn Wilson, State of Minnesota

Exhibits:

A. Affidavit
B. Licensee Evaluation
C Marked Up Pages
D. Revised Pages
E. List of Commitments
F. Safety Analysis for License Amendment Request, Selective Scope Implementation of Alternate Source Term for Prairie Island Nuclear Generating Plant, Units 1 and 2, Fuel Handling Accident Analysis
G. Figures 1, 2 and 3 (Proprietary Information)
H. Figures 1, 2 and 3 (Non-proprietary Version)
I. Draft Administrative Containment Closure Controls During Fuel Movement for the Prairie Island Nuclear Generating Plant

Enclosure:

Electronic media containing meteorological data in the ARCON96 format

Exhibit G Contains Proprietary Information

UNITED STATES NUCLEAR REGULATORY COMMISSION

NUCLEAR MANAGEMENT COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT DOCKET Nos. 50-282
50-306

Request to Withhold Proprietary Information from Public Disclosure

The Nuclear Management Company (NMC) hereby requests that Exhibit G to the letter entitled "Selective Scope Implementation of Alternate Source Term for Fuel Handling Accident Applied to Containment Technical Specifications" dated JAN 2 0 2004 , be withheld from public disclosure due to its proprietary nature. The details of this request are provided in the following affidavit:

AFFIDAVIT

I, Joseph M. Solymossy, being duly sworn, depose and state as follows:

(1) I am the Site Vice President for the Prairie Island Nuclear Generating Plant and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld consists of Figures 1, 2 and 3 that are included in Exhibit G to the Nuclear Management Company (NMC) letter to the NRC entitled "Selective Scope Implementation of Alternate Source Term for Fuel Handling Accident Applied to Containment Technical Specifications ". Exhibit G, a four page document, has the words "Proprietary Information" on the bottom of each page.

(3) In making this application for withholding of proprietary information of which it is the owner, NMC relies upon the exemption from disclosure set forth In the NRC regulation 10 CFR 2.790(b)(1) for confidential commercial information.

(4) Justification for the request for withholding from public disclosure is provided by addressing the five items identified in 10 CFR 2.790(b)(4).

To the best of my knowledge and belief:

a. This Information is considered company confidential and has been held in confidence by NMC.

b. This information is of the type customarily held in confidence by NMC and the rationale basis is that it meets the requirements of 10 CFR 2.790 (d)(1).

c. This information is transmitted in confidence to the NRC and the purpose of this request is to maintain its confidentiality.

d. These Figures are not available from public sources.

e. Public disclosure of the information sought to be withheld is likely to cause harm to NMCs competitive position within the meaning of 10 CFR 2.790 (d)(1).

This letter contains no restricted or other defense information.

NUCLEAR MANAGEMENT COMPANY

By Joseph M. Solymossy
Joseph M. Solymossy
Site Vice President,
Prairie Island Nuclear Generating Plant

State of Minnesota

County of Goodhue

On this 20th day of January 2004 before me a notary public acting in said County, personally appeared Joseph M. Solymossy, Site Vice President, Prairie Island Nuclear Generating Plant and being first duly sworn acknowledged that he is authorized to execute this document on behalf of the Nuclear Management Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true.

Dale M. Vincent

DALE M. VINCENT
NOTARY PUBLIC-MINNESOTA
My Commission Expires Jan. 31, 2005

Exhibit B

L-PI-04-001

LICENSEE EVALUATION

SELECTIVE SCOPE IMPLEMENTATION OF
ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO
CONTAINMENT TECHNICAL SPECIFICATIONS

1.0 DESCRIPTION

Pursuant to 10 CFR Part 50, Section 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following changes to the Technical Specifications (TS) contained in Appendix A of the Facility Operating Licenses:

As a holder of an operating license issued prior to January 10, 1997, and in accordance with 10 CFR 50.67, the Nuclear Management Company (NMC) is requesting to replace the accident source term used at the Prairie Island Nuclear Generating Plant for a fuel handling accident, occurring in containment or the fuel pool enclosure, by the selective implementation of Alternative Source Term (AST). The Nuclear Management Company (NMC) has revised the consequence analysis of postulated fuel handling accidents in containment and in the fuel pool enclosure using the alternate source term methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183. In the revised analysis for a fuel handling accident within containment, no credit is taken for containment integrity with respect to containment penetration closure, personnel airlock closure, equipment hatch closure or filtration by the Containment Purge or Inservice Purge Systems. Likewise, for a fuel handling accident within the spent fuel pool (SFP) enclosure, no credit is taken for SFP enclosure integrity or filtration by the SFP special ventilation system (SFPSVS).

The TS changes proposed in this LAR implement a portion of industry improved Standard Technical Specifications traveler, TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations." as it applies to TS 3.9.4, "Containment Penetrations." Consistent with TSTF-51, NMC commits to the guidelines of TSTF-51 Reviewer's Note for the assessment of systems removed from service during movement of irradiated fuel at PINGP. Specifically the guidelines of NUMARC 93-01, Revision 3, Section 11.3.6, "Assessment Methods for Shutdown Conditions," Subsection 11.3.6.5, that will be adopted are:

> During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS [reactor coolant system] decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in

doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay and to avoid unmonitored releases.

A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

2.0 PROPOSED CHANGES

A brief description of the proposed changes is provided below along with a discussion of the justification for each change. The specific wording changes to the Technical Specifications are provided in Exhibits C and D.

LCO 3.3.5, "Containment Ventilation Isolation Instrumentation" (CVI) and Bases: The Applicability Note (b) of Table 3.3.5-1 and Condition C have been removed from this Specification. Proposed TS 3.9.4 has been revised to require the Containment Purge and Inservice Purge Systems to be isolated at all times when recently irradiated fuel is handled which means isolation of these systems by the CVI is not required.

Analyses have been performed in accordance with the AST methodology of 10 CFR 50.67 for the FHA. These analyses, discussed below and in Exhibit F, demonstrate that releases from a fuel handling accident meet the acceptance criteria of 10 CFR 50.67 without credit for containment ventilation system isolation assuming the fuel has not been in a critical reactor within the last 50 hours.

CVI is not required when handling recently irradiated fuel since the Containment Purge and Inservice Purge Systems are required to be isolated as specified in proposed TS 3.9.4 and the Containment Purge and Inservice Purge Systems are not credited for filtration during handling of fuel which is not recently irradiated. Thus, CVI requirements during reactor shutdown are proposed to be removed from TS 3.3.5. The Bases for TS 3.3.5 have been revised to support the changes proposed for Specification 3.3.5.

LCO 3.9.4, "Containment Penetrations" and Bases: The Applicability of LCO 3.9.4 has been revised to require this TS to apply during the movement of recently irradiated fuel assemblies within containment. This Applicability change is also accompanied by a change to the Required Actions. This Specification has also been revised to require all penetrations to be closed during movement of recently irradiated fuel assemblies within containment, including one door in each air lock and the Containment Purge and

Inservice Purge System penetrations. Note, an exception is provided for opening penetrations under administrative controls consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants" Revision 2.

Analyses have performed in accordance with the AST methodology of 10 CFR 50.67 for the FHA. These analyses, discussed below and in Exhibit F, demonstrate that releases from a fuel handling accident meet the acceptance criteria of 10 CFR 50.67 without credit for containment integrity assuming the fuel has not been in a critical reactor within the last 50 hours. Thus the Bases define "recently" as 50 hours and the regulatory acceptance criteria are met without taking credit for containment integrity when handling fuel that has not occupied part of a critical core within the previous 50 hours.

Since the revised TS 3.9.4 requires at least one the containment air lock door to be closed during movement of recently irradiated fuel, the current TS requirements for the containment (high flow) purge system to be isolated, one air lock door to be OPERABLE and two containment fan coil unit fans capable of operating are not applicable. These requirements related to the current analysis assumptions which are superceded by the proposed AST FHA analyses proposed in this LAR. Thus, this LAR proposes to remove these requirements. This proposed change makes TS 3.9.4 LCO statement (b.) consistent with the guidance of NUREG-1431. This TS change also proposes to require all penetrations to be closed, including the Containment Purge and Inservice Purge Systems, thus the requirements for CVI operability is removed.

The Bases for TS 3.9.4 have been revised to support the changes proposed for Specification 3.9.4.and generally implement the guidance of TSTF-51.

3.0 BACKGROUND

The NRC methods for calculating accident doses were developed to be consistent with Technical Information Document (TID) –14844, "Calculation of Distance Factors for Power and Test Reactors." Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1999, the NRC promulgated a new regulation, 10 CFR 50.67 based on these advances. Specifically the NRC stated in the Statements of Consideration for 10 CFR 50.67, Published 12/23/99, 64 FR 71990:

> The Nuclear Regulatory Commission (NRC) is amending its regulations to allow holders of operating licenses for nuclear power plants to voluntarily replace the traditional source term used in design basis accident analyses with alternative source terms. This action will allow interested licensees to pursue cost beneficial licensing actions to reduce unnecessary regulatory burdens without compromising the margin of safety of the facility.

The Statements of Consideration provide further elaboration of this purpose for promulgation of 10 CFR 50.67 as follows:

> The NRC considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety, and that operating reactors licensed under this approach would not be required to reanalyze accidents using the revised source terms. The NRC concluded that some licensees may wish to use an alternative source term in analyses to support operational flexibility and cost-beneficial licensing actions and that some of these applications could provide concomitant improvements in overall safety and in reduced occupational exposure.

This LAR adopts the AST methodology of 10 CFR 50.67 for the FHA and proposes cost beneficial license amendments which reduce unnecessary regulatory burdens without compromising the margin of safety of the facility. This LAR proposes to revise the applicability of containment integrity during refueling operations and remove containment ventilation isolation system instrumentation requirements. These changes will allow outage scheduling flexibility, reduce the cost of maintaining containment integrity during refueling outages and allow cost savings during replacement of steam generators in Unit 1 during the Fall of 2004.

The AST analysis for the design basis accident presented in Exhibit F follows the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors" (Reference 1).

To facilitate NRC review and approval of this LAR by the Fall of 2004, the proposed TS changes implement a portion of TSTF-51 only as it applies to containment integrity, TS 3.9.4. The TSTF-51 changes to TS 3.9.4 can be made independent of the TSTF-51 changes in other TS. With the exception of TS 3.9.7 (PINGP TS 3.9.2), TSTF-51 adds "recently" to all TS with applicability, "During movement of irradiated fuel assemblies". With the exception of TS 3.3.5 for which changes are proposed in this LAR, the PINGP TS will continue to apply these other TS requirements during movements of irradiated fuel until such time that an LAR may be submitted to propose changes to these other TS.

The NRC has previously approved FHA implementation of AST for other nuclear power plants including Perry Nuclear Power Plant, Fitzpatrick Nuclear Power Plant and the H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson Plant). The changes proposed in this LAR are similar to the changes approved for the Robinson Plant in that they added "recently" to the applicability for their Refueling Operations Containment Penetrations Technical Specification (Robinson Plant TS 3.9.3) and defined "recently" as 56 hours in their TS Bases.

4.0 TECHNICAL ANALYSIS

PINGP is a two unit plant with a containment structure for each unit. A spent fuel storage enclosure is shared between the two units. Each containment has a Containment Purge System which TS does not allow to be operated when in Modes 1, 2, 3 and 4. Each containment is equipped with an Inservice Purge System which shares filtration trains with the SFPSVS. The SFPSVS has two complete redundant trains. One train exhausts through the Unit 1 Shield Building exhaust stack while the other train exhausts through the Unit 2 Shield Building exhaust stack.

This LAR proposes to change the design and licensing basis for the FHA by selective implementation of the AST methodology as allowed by 10 CFR 50.67. Based on the revised FHA, this LAR proposes to revise the TS 3.9.4 to apply during movement of recently irradiated fuel assemblies. Containment penetrations will be required to be closed during handling of recently irradiated fuel, including airlock doors and purge and exhaust penetrations. The analysis is based on the guidance provided by Regulatory Guide 1.183, the guideline for use of the AST methodology. The specifics of the analysis are contained in Exhibit F; however, a summary is provided below.

Current Licensing Basis FHA

The current licensing basis accident analysis for the FHA at the PINGP includes analysis for a FHA inside of containment or in the spent fuel pool (SFP). Both analyses assume a source term derived from reactor power operations at 1683 MWt (1650 MWt + 2% calorimetric uncertainty) and that the FHA occurs 100 hours after reactor shutdown. It is assumed that an assembly is dropped and the accident results in damage to all rods in the dropped assembly such that the gaseous fission products contained in the fuel cladding gap are released.

For the FHA inside containment, the radioactive material that escapes the containment refueling pool mixes in containment and is released to the outside atmosphere exponentially (based on 6000 cfm supplied by the Inservice Purge System) over a two hour time period through open containment airlock doors.

For the FHA in the SFP, the activity is released through the SFPSVS, where it is filtered (HEPA and charcoal filters) prior to release.

Control Room dose results were calculated assuming the Control Room Special Ventilation System (CRSVS) has been aligned to isolate outside air and filter a portion of the recirculated airflow.

The radiological consequences of the FHA as defined in the current licensing basis are well within the exclusion area boundary (EAB) and low population zone (LPZ) dose limits of 10 CFR 100. Control Room dose is within the limits of 10 CFR 50, Appendix A, General Design Criterion (GDC) 19.

Proposed Alternate Source Term FHA

Under the proposed AST accident methodology for the FHA, a fuel assembly is assumed to be dropped and damaged during refueling 50 hours after reactor shutdown. Analysis of the FHA accident is performed such that the results are bounding for the accident occurring either inside containment or the spent fuel pool. The activity from the damaged assembly is assumed to be released over two hours to the outside atmosphere.

No credit is taken for ventilation filtration system operation in the spent fuel area. Similarly, no credit is taken for containment ventilation system (Containment Purge or Inservice Purge) closure or filtration capability. In addition, no credit is taken for the containment equipment hatch placement or closure nor is credit taken for having air lock doors or other containment penetrations capable of closure.

Control Room dose results are calculated assuming the CRSVS is actuated on a high radiation signal, which aligns the system to isolate outside air and filter a portion of the recirculated airflow. Unfiltered inleakage and atmospheric dispersion factors are used which are more restrictive than the current analyses as discussed in Exhibit F.

The radiological consequences of the FHA under the revised AST methodology are within the dose limits in 10 CFR 50.67 and meet the acceptance criteria of RG 1.183 for the EAB, LPZ and the Control room.

Basis for Proposed TS Changes

The proposed AST methodology FHA analysis does not take credit for removal of radioactive materials by containment and spent fuel pool building ventilation systems' filters nor is credit taken for isolation of release paths. The activity is assumed to be released from the pool (spent fuel pool or containment refueling pool) to the outside atmosphere over a 2-hour period. Since no containment isolation is modeled, this analysis supports irradiated fuel handling operations without containment integrity, including the equipment hatch or personnel air lock remaining open assuming the fuel is not recently irradiated.

The current FHA analysis assumes an irradiated fuel assembly is dropped 100 hours after the reactor is shutdown. At the PINGP, irradiated fuel cannot be physically handled within 100 hours after reactor shutdown because the PINGP design does not allow the necessary reactor shutdown, cooldown, and disassembly within this time. Therefore this 100 hour irradiated fuel handling limitation is not included in the PINGP TS.

The AST FHA analysis assumes an irradiated fuel assembly is dropped 50 hours after the reactor is shutdown. NMC has evaluated the outage schedule impact of replacing the reactor vessel head which could accelerate reactor disassembly. With a new reactor vessel head and associated plant improvements, the plant may be ready for handling irradiated fuel assemblies within 72 hours after reactor shutdown. Since irradiated fuel cannot be handled in less than the 50 hour limitation assumed for the AST FHA, this limitation is not included in the PINGP TS.

Based on the above discussion and the analysis presented in Exhibit F, the proposed TS changes are consistent with 10 CFR 50.67 and do not adversely affect nuclear safety or plant operations.

Fuel Handling Accident Management and Release Monitoring

NMC has demonstrated in this LAR that the dose consequences of a FHA in the SFP or containment meet the regulatory guidance provided in 10 CFR 50.67. An analysis of the FHA based on expected conditions could be performed which would likely show a significantly lower dose consequence than using the worst case assumptions considered in the analyses presented in Exhibit F. Examples of some of the conservative assumptions follow. The FHA was assumed to occur at 50 hours which is earlier than can actually be physically achieved for fuel handling operations. If the FHA were to occur at 100 hours (a realistic minimum start time for fuel handling) or 200 hours, the dose would be significantly lower. All fuel rods in the dropped assembly are assumed to be breached and release all gap activity. More realistic assumptions would decrease the dose proportionately. The fission product inventory is calculated on the basis of the highest power assembly; the dose from an average assembly would be approximately 60% of that calculated. Conservative meteorological conditions were assumed; the dose during average conditions would be a small faction of that calculated. Releases are assumed to exit facility openings without mixing with ambient air. Quantifying and combining these and other conservatisms in the analysis would likely significantly reduce the dose consequences even further below the regulatory requirements provided in 10 CFR 50.67.

Notwithstanding the Technical Specification changes proposed in this license amendment, NMC remains committed to defense in depth. Exhibit I provides proposed administrative controls for managing and monitoring a fuel handling accident in containment including assurances that designated personnel are available to isolate or direct isolation of affected openings in the event of a FHA, that any obstruction which would prevent rapid closure of an open flow path can be quickly removed and that ventilation system and radiation monitor availability will be assessed.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The Nuclear Management Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   The proposed Technical Specification changes require containment integrity during movement of recently irradiated fuel. With this change, the Technical Specifications selectively implement 10 CFR 50.67 alternative source term methodologies for a fuel handling accident and implement portions of the approved industry improved Standard Technical Specification traveler, TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations" as it applies to TS 3.9.4, "Containment Penetrations." This change also removes requirements for containment ventilation isolation instrumentation during handling irradiated fuel from TS 3.3.5, "Containment Ventilation Isolation Instrumentation" since the containment purge and inservice purge system penetrations which are isolated by this instrumentation will be required to be isolated during movement of recently irradiated fuel. With the proposed 10 CFR 50.67 alternative source term methodologies, these filtration systems are not assumed to function during a fuel handling accident involving fuel which is not recently irradiated.

   This amendment does not alter the methodology or equipment used directly in fuel handling operations. None of the containment integrity features including the containment equipment hatch, personnel air locks or any other containment penetration are used to handle fuel. Therefore, containment integrity and ventilation systems, and spent fuel pool ventilation systems are not accident initiators and therefore these changes do not increase the probability of a previously evaluated accident.

   The total effective dose equivalent (TEDE) doses from the analysis supporting this amendment request have been compared to equivalent total effective dose equivalent (TEDE) doses estimated with the guidelines of Regulatory Guide 1.183 Footnote 7. The new values are shown to be comparable to the results of the previous analysis.

   A fuel handling accident analysis utilizing alternative source term methodologies allowed by 10 CFR 50.67 demonstrated that the dose consequences of a postulated fuel handling accident remain within the limits of 10 CFR 50.67 without taking credit for containment closure or ventilation systems assuming the fuel has not recently been in a critical reactor. The alternative source term fuel handling accident analysis also demonstrated that the more restrictive dose guidelines of Regulatory Guide 1.183 are also met without taking credit for these mitigation features. Since the alternative source term fuel handling accident analysis results are within the regulatory limits and regulatory guidelines without taking credit for these mitigation features, revising this Technical Specification for containment closure does not involve a significant increase in the consequences of a previously evaluated accident.

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

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

Response: No

The proposed Technical Specification changes require containment integrity during movement of recently irradiated fuel. With this change, the Technical Specifications selectively implement 10 CFR 50.67 alternative source term methodologies for a fuel handling accident and implement portions of the approved industry improved Standard Technical Specification traveler, TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations" as it applies to TS 3.9.4, "Containment Penetrations." This change also removes requirements for containment ventilation isolation instrumentation during handling irradiated fuel from TS 3.3.5, "Containment Ventilation Isolation Instrumentation" since the containment purge and inservice purge system penetrations which are isolated by this instrumentation will be required to be isolated during movement of recently irradiated fuel. With the proposed 10 CFR 50.67 alternative source term methodologies, these filtration systems are not assumed to function during a fuel handling accident involving fuel which is not recently irradiated.

The proposed Technical Specification changes do not involve plant design, hardware, system operation, or procedures involved with actual handling of irradiated fuel. The proposed changes include application of new methodology for fuel handling accident analysis and revises requirements for equipment operability during movement of irradiated fuel assemblies. These changes do not create the possibility for a new or different kind of accident.

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

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

Response: No

The proposed Technical Specification changes require containment integrity during movement of recently irradiated fuel. With this change, the Technical Specifications selectively implement 10 CFR 50.67 alternative source term methodologies for a fuel handling accident and implement portions of the approved industry improved Standard Technical Specification traveler, TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations" as it applies to TS 3.9.4, "Containment Penetrations." This change also removes requirements for containment ventilation isolation instrumentation during handling irradiated fuel from TS 3.3.5, "Containment Ventilation Isolation Instrumentation" since the containment purge and inservice purge system penetrations which are isolated by this instrumentation will be required to be isolated during movement of recently irradiated fuel. With the proposed 10 CFR 50.67 alternative source term methodologies, these filtration systems are not assumed to function during a fuel handling accident involving fuel which is not recently irradiated.

The assumptions and input used in the fuel handling accident analysis are conservative. The design basis fuel handling accident has been defined to identify conservative conditions. The source term and radioactivity releases have been calculated pursuant to Regulatory Guide 1.183, Appendix B and with conservative assumptions concerning prior reactor operations. The control room atmospheric dispersion factor has been calculated with conservative assumptions associated with the release. These conservative assumptions and input ensure that the radiation doses cited in this license amendment request are the upper bounds to radiological consequences of a fuel handling accident in containment or the spent fuel pool. The analysis shows that there is a significant margin between the offsite radiation doses calculated for the postulated fuel handling accident using the alternate source term and the dose limits of 10 CFR 50.67 and acceptance criteria of Regulatory Guide 1.183. The proposed changes will not degrade the plant protective boundaries, will not cause a release of fission products to the public, and will not degrade the performance of any structures, systems, and components important to safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.67

10 CFR 50.67 specifies in part the following requirements for revising the current accident source term:

(1) The requirements of 10 CFR 50.67 apply to holders of operating licenses issued prior to January 10, 1997 who seek to revise the current accident source term used in their design basis radiological analyses.

The Prairie Island Nuclear Generating Plant operating license was issued prior to January 10, 1997 and the Nuclear Management Company (licensee) seeks to revise the current accident source term used in the Prairie Island Nuclear Generating Plant design basis radiological analyses for a fuel handling accident.

(2) Licensees who seek to revise the current accident source term used in their design basis radiological analyses shall apply for a license amendment under 10 CFR 50.90.

This license amendment request seeks to revise the current accident source term used in the Prairie Island Nuclear Generating Plant design basis radiological analyses. This license amendment request seeks is submitted pursuant to 10 CFR 50.90.

(3) The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.

This license amendment request provides Exhibit F which contains an evaluation of the consequences of a fuel handling accident occurring in containment or the fuel pool enclosure. These accidents have been previously analyzed in the Prairie Island Nuclear Generating Plant Updated Safety Analysis Report.

(4) The NRC may issue the license amendment 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 25 rem total effective dose

equivalent (TEDE).

The Nuclear Regulatory Commission has issued Regulatory Guide 1.183 to provide guidance for performance of accident analyses to demonstrate with reasonable assurance that the dose limits of 10 CFR 50.67 are met. Exhibit F conforms to the guidance provided in Appendix B of Regulatory Guide 1.183. Exhibit F demonstrates 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 25 rem total effective dose equivalent (TEDE).

(5) ***The NRC may issue the license amendment if the applicant's analysis demonstrates with reasonable assurance that 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 25 rem total effective dose equivalent (TEDE).***

The Nuclear Regulatory Commission has issued Regulatory Guide 1.183 to provide guidance for performance of accident analyses to demonstrate with reasonable assurance that the dose limits of 10 CFR 50.67 are met. Exhibit F conforms to the guidance provided in Appendix B of Regulatory Guide 1.183. Exhibit F demonstrates that 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 25 rem total effective dose equivalent (TEDE).

(6) ***The NRC may issue the license amendment if the applicant's analysis demonstrates with reasonable assurance that 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 5 rem total effective dose equivalent (TEDE) for the duration of the accident.***

The Nuclear Regulatory Commission has issued Regulatory Guide 1.183 to provide guidance for performance of accident analyses to demonstrate with reasonable assurance that the dose limits of 10 CFR 50.67 are met. Exhibit F conforms to the guidance provided in Appendix B of Regulatory Guide 1.183. Exhibit F demonstrates that 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 5 rem total effective dose equivalent (TEDE) for the duration of the accident

This license amendment request meets the requirements of 10 CFR 50.67 for selective implementation of the alternate source term for a fuel handling accident.

General Design Criteria

The construction of the PINGP was significantly complete prior to issuance of 10 CFR 50 Appendix A General Design Criteria. The PINGP was designed and constructed to comply with the AEC General Design Criteria as proposed on July 10, 1967 (AEC GDC) as described in the plant Updated Safety Analysis Report (USAR). AEC GDC 69 and 70 provide design guidance for accident releases from spent fuel.

AEC draft GDC 69 states, "*Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.*" This GDC address radiological releases due to a fuel handling accident. The AST analyses of a FHA in the containment demonstrate that the dose consequences are within the limits of 10 CFR 50.67 without credit for the containment structures or ventilation systems assuming the fuel is not recently irradiated. Thus FHA accidents with the proposed TS changes do not lead to undue amounts of radioactivity to the public environs.

AEC draft GDC 70 states:

> *The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceeding low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities of very large cities can be affected by the radioactive effluents.*

This GDC, proposed in 1967, requires plant design features to maintain accident releases below 10 CFR 100 limits. For the FHA using the recently issued AST methodology, 10 CFR 50.67 provides the applicable limits for accident release design requirements. This LAR demonstrates that the limits of 10 CFR 50.67 are met without credit for containment confinement of pool releases, ventilation system filtration or ventilation system isolation assuming the fuel is not recently irradiated. Thus the facility design without containment integrity or ventilation system isolation TS includes the means necessary to maintain control over plant fuel handling accident radioactive effluents on the basis of 10 CFR 50.67 which establishes the applicable dose consequence limits.

Although the proposed TS require containment integrity and containment ventilation system isolation only during movement of recently irradiated fuel, NMC is making a commitment in this license amendment request to implement administrative controls for containment closure following a fuel handling accident in containment.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

The Nuclear Management Company has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCE

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors."

EXHIBIT C

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Letter PI-04-001
SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO CONTAINMENT TECHNICAL SPECIFICATIONS

Marked Up Pages
(shaded material to be added, strikethrough material to be removed)

Technical Specification Pages

3.3.5-3
3.3.5-5
3.9.4-1
3.9.4-2

Bases Pages

B 3.3.5-1
B 3.3.5-2
B 3.3.5-4
B 3.3.5-5
B 3.3.5-6
B 3.3.5-7
B 3.3.5-9
B 3.9.4-1
B 3.9.4-2
B 3.9.4-3
B 3.9.4-4
B 3.9.4-5
B 3.9.4-6
B 3.9.4-7
B 3.9.4-8

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C. ----------NOTE----------~~ ~~Only applicable during movement of irradiated fuel assemblies within containment when the Containment Purge or Inservice Purge Systems are not isolated.~~ ~~------------------------------~~ ~~One or more Functions with one or more manual or automatic actuation trains inoperable.~~ ~~OR~~ ~~Two required radiation monitoring channels inoperable.~~ ~~OR~~ ~~Required Action and associated Completion Time for Condition A not met.~~	~~C.1 Place and maintain containment purge (high flow) and inservice (low flow) purge valves in closed position.~~ ~~OR~~ ~~C.2 Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge (high flow) and inservice (low flow) purge valves made inoperable by isolation instrumentation.~~	~~Immediately~~ ~~Immediately~~

Table 3.3.5-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Manual Initiation	1(a), 2(a), 3(a), 4(a) ~~(b)~~	2	SR 3.3.5.5	NA
2.	Automatic Actuation Relay Logic	1(a), 2(a), 3(a), 4(a) ~~(b)~~	2 trains	SR 3.3.5.2 SR 3.3.5.4	NA
3.	High Radiation in Exhaust Air	1(a), 2(a), 3(a), 4(a) ~~(b)~~	2 (1 per train)	SR 3.3.5.1 SR 3.3.5.3 SR 3.3.5.6	(b~~e~~)
4.	Manual Containment Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a., for initiation functions and requirements.			
5.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for initiation functions and requirements.			
6.	Manual Containment Spray	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 2.a., for initiation functions and requirements.			

(a) When the Containment Inservice Purge System is not isolated.

(b) ~~During movement of irradiated fuel assemblies within containment when the Containment Purge or Inservice Purge System is not isolated.~~

~~(e)~~ ≤ count rate corresponding to 500 mrem/year whole body and 3000 mrem/year skin due to noble gases at the site boundary.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed~~, or both doors in each air lock may be open with:~~

~~1. containment (high flow) purge system isolated,~~

~~2. one air lock door OPERABLE, and~~

~~3. at least two containment fan coil unit fans capable of operating in the high speed mode~~; and

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

~~1.~~ closed by a manual or automatic isolation valve, blind flange, or equivalent~~, or~~

~~2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.~~

--------------------------NOTE--------------------------

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
~~SR 3.9.4.2~~	~~NOTE~~ ~~Not required to be met for containment purge (high flow) and inservice (low flow) purge valve(s) in penetrations closed to comply with LCO 3.9.4.c.1.~~ ~~Verify each required containment purge (high flow) and inservice (low flow) purge system valve actuates to the isolation position on an actual or simulated actuation signal.~~	~~24 months~~

B 3.3 INSTRUMENTATION

B 3.3.5 Containment Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment ventilation isolation (CVI) instrumentation closes the containment isolation valves in the Containment Purge (high flow) and Inservice (low flow) Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Containment Inservice (low flow) Purge System may be in use during reactor operation and with the reactor shutdown except during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 50 hours). The Containment Purge (high flow) System may be in use with the reactor shutdown except during handling of recently irradiated fuel.

Containment ventilation isolation initiates on a safety injection (SI) signal, by manual actuation of containment isolation, or by manual actuation of containment spray. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Three radiation monitoring channels are also provided as input to CVI. One channel measures gaseous radiation in containment exhaust air. This channel provides an input to one train of CVI actuation relay logic. The other two channels measure either gaseous or particulate containment exhaust air radiation. These two channels provide inputs to the other train of CVI actuation relay logic where either channel will actuate the train. These three detectors will respond to most events that release radiation to containment. Since the monitors constitute a sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY.

BASES

BACKGROUND (continued)

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from any one of the three channels initiates one train of CVI logic, which closes one supply and one exhaust containment isolation valve in the Containment Purge (high flow) System and Inservice (low flow) Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment exhaust air radiation monitors act as backup to the SI signal to ensure closing of the purge and exhaust valves. ~~They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown.~~ Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The CVI instrumentation satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate CVI, listed in Table 3.3.5-1, is OPERABLE.

1. Manual Initiation

   The LCO requires two channels OPERABLE. The operator can initiate CVI at any time by using either of two switches in

BASES

LCO (continued)

3. High Radiation in Exhaust Air

The LCO specifies two required channels of radiation monitors, one per train, to ensure that the radiation monitoring instrumentation necessary to initiate CVI remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups, and sample pump operation as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

4. Manual Containment Isolation

Refer to LCO 3.3.2, Function 3.a., for initiating Functions and requirements.

5. Safety Injection

Refer to LCO 3.3.2, Function 1, for initiating Functions and requirements.

6. Manual Containment Spray

Refer to LCO 3.3.2, Function 2, for initiating Functions and requirements.

APPLICABILITY

All Functions in Table 3.3.5-1 are required to be OPERABLE in MODES 1, 2, 3, and 4 when the Containment Inservice (low flow) Purge System is not isolated. ~~In addition, the Manual Initiation,~~

BASES

APPLICABILITY (continued)

~~Automatic Actuation Relay Logic, and High Radiation in Exhaust Air Functions are required OPERABLE during movement of irradiated fuel assemblies within containment, when the Containment Purge (high flow) and Inservice (low flow) Purge Systems are not isolated. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the CVI instrumentation must be OPERABLE in these MODES.~~

~~While in MODES 5 and 6 without irradiated fuel handling in progress, the CVI instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.~~

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the Allowable Value, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

BASES

ACTIONS (continued)

A.1

Condition A applies to the failure of one CVI radiation monitor channel.

The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that the remaining channels will respond to events.

B.1

Condition B applies to all CVI Functions and addresses the train orientation for these Functions.

If a train is inoperable, two required radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4 when the Containment Inservice Purge System is not isolated.

~~C.1 and C.2~~

~~Condition C applies to all CVI Functions and addresses the train orientation for these Functions. If a train is inoperable, two required radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge (high flow) and inservice (low flow) purge and exhaust isolation valves in their closed position is met~~

BASES

ACTIONS

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

~~or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.~~

~~A Note states that Condition C is only applicable during movement of irradiated fuel assemblies within containment when the Containment Purge and Inservice Purge Systems are not isolated.~~

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which CVI Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

BASES ~~(continued)~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.5

SR 3.3.5.5 is the performance of a TADOT. This test is a check of the Manual Initiation Function and is performed every 24 months. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.5.6

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 50 hours) within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent and ~~.~~ ~~The LCO~~ requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed ~~or capable of being closed~~. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 50.67~~100~~. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

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

BACKGROUND

The containment air locks, which are also part of the containment

BASES

(continued)

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

During movement of recently irradiated fuel assemblies within containment, containment closure ~~or closure capability~~ is required; therefore, the door interlock mechanism may remain disabled but one air lock door must always remain closed~~and both doors may be open provided one door can be closed within 30 minutes with at least two containment fan coil unit fans capable of operating in high speed~~.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will ~~restrict fission product radioactivity release from containment to~~ be restricted to within regulatory limits.

The Containment Purge and Exhaust System includes two subsystems, Containment Purge and Containment Inservice Purge. The containment purge subsystem includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a minipurge system referred to as containment inservice purge, includes a 14 inch purge penetration and an 18 inch exhaust penetration.

During MODES 1, 2, 3, and 4, the two valves in each of the containment purge and exhaust penetrations are secured in the closed position, or the penetrations may be blank flanged. The two valves in each of the two containment inservice purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation System.

BASES

BACKGROUND (continued)

In MODE 6 during handling of recent irradiated fuel, the Containment Purge and Containment Inservice Purge Systems must remain closed.~~sufficient air flow rates are necessary to conduct refueling operations. The inservice purge system is used for this purpose, and each of the four valves is closed by the radiation monitors associated with the containment inservice purge system in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation." The 36 inch subsystem is normally blank flanged, although the option for use is allowed during outages, except during movement of irradiated fuel with the air lock doors open. All four containment purge valves are also closed by the Containment Ventilation Isolation Instrumentation~~.

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

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.2, "Refueling Cavity Water Level," and~~in conjunction with the minimum decay time of 100 hours prior to~~ irradiated fuel movement with containment closure capability or a minimum decay time of 50 hours without containment closure capability ensure that the release of fission product radioactivity,

BASES

subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 50.67~~100~~. The acceptance limit for offsite radiation exposure is 25% of 10 CFR 50.67~~100~~ values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 50.67 limits).

APPLICABLE SAFETY ANALYSES (continued)

~~The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will restrict fission product release from containment to be well within regulatory limits. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.~~

~~A fuel handling accident does not cause containment pressurization; however, with an assumed single failure, the operating purge system supply fan is assumed to continue supplying air to containment. To maintain post-fuel handling accident releases well within the limits of 10 CFR 100, only the inservice purge system is allowed to be operating during fuel movement. Two fan coil unit fans are required to operate in the high speed mode following a fuel handling accident to assure that radioactive material in containment is well mixed and any releases will leave containment at a lower concentration over the duration of the accident. The provision that one air lock door is OPERABLE and under procedural control will assure that at least one door remains capable of being closed as required, thus assuring radioactive releases are well within the limits of 10 CFR 100 (Ref. 1).~~

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

LCO

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of irradiated fuel assemblies within

BASES

containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

This LCO limits the consequences of a fuel handling accident involving recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment.

LCO (continued)

The LCO requires ~~containment~~ penetrations providing direct access from the containment atmosphere to the outside atmosphere to be closed, including containment purge and exhaust penetrations. ~~to meet the following requirements:~~

~~a. The equipment hatch is closed and held in place by at least 4 bolts;~~

~~b. One door in each air lock is closed, or both doors in each air lock may be open with:~~

~~1. containment (high flow) purge system isolated,~~

~~2. one air lock door capable of being closed, and~~

~~3. at least two containment fan coil unit fans capable of operating in the high speed mode; and~~

~~c. Each penetration (including the containment (high flow) purge system and inservice (low flow) purge system.) providing direct access from the containment atmosphere to the outside atmosphere is either:~~

~~1. closed by a manual valve, or automatic isolation valve, blind flange, or equivalent: or~~

~~2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.~~

~~A penetration with direct access from the containment~~

BASES

~~atmosphere to the outside atmosphere includes all penetrations open to the containment atmosphere that provide a flow path that leads anywhere outside containment and are open to the atmosphere.~~

LCO
(continued)

~~For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO require that the automatic purge and exhaust valve closure can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.~~

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident.

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Due to radioactive decay, a fuel handling accident involving handling fuel not recently irradiated (i.e., fuel that has not occupied part of a critical reactor core within the previous 50 hours) will result in doses that are well within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

BASES

ACTIONS

A.1

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, ~~including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open,~~ the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. ~~The Surveillance on the open purge and exhaust valves will demonstrate that the valves will function if required during a fuel handling accident. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic Containment Ventilation Isolation signal.~~

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

SURVEILLANCE ~~SR 3.9.4.2~~

BASES

REQUIREMENTS (continued)

~~This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. A CHANNEL CALIBRATION is performed every 24 months. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances, when performed, will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.~~

~~The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.~~

REFERENCES

1. USAR, Section 14.5.

EXHIBIT D

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Letter PI-04-001
SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO CONTAINMENT TECHNICAL SPECIFICATIONS

Revised Pages

Technical Specification Pages

3.3.5-2
3.3.5-3
3.3.5-4
3.9.4-1
3.9.4-2

Bases Pages

B 3.3.5-1
B 3.3.5-2
B 3.3.5-4
B 3.3.5-5
B 3.3.5-7
B 3.3.5-8
B 3.9.4-1
B 3.9.4-2
B 3.9.4-3
B 3.9.4-4
B 3.9.4-5
B 3.9.4-6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. -----------NOTE----------- Only applicable in MODE 1, 2, 3, or 4 when the Containment Inservice Purge System is not isolated. ------------------------------ One or more Functions with one or more manual or automatic actuation trains inoperable. <u>OR</u> Two required radiation monitoring channels inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment inservice (low flow) purge valves made inoperable by isolation instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.5-1 to determine which SRs apply for each Containment Ventilation Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.5.2	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.5.3	Perform COT.	31 days
SR 3.3.5.4	Perform SLAVE RELAY TEST.	24 months
SR 3.3.5.5	--------------------------NOTE-------------------------- Verification of setpoint is not required. -- Perform TADOT.	 24 months
SR 3.3.5.6	Perform CHANNEL CALIBRATION.	24 months

Table 3.3.5-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Manual Initiation	1[a], 2[a], 3[a], 4[a]	2	SR 3.3.5.5	NA
2.	Automatic Actuation Relay Logic	1[a], 2[a], 3[a], 4[a]	2 trains	SR 3.3.5.2 SR 3.3.5.4	NA
3.	High Radiation in Exhaust Air	1[a], 2[a], 3[a], 4[a]	2 (1 per train)	SR 3.3.5.1 SR 3.3.5.3 SR 3.3.5.6	(b)
4.	Manual Containment Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a., for initiation functions and requirements.			
5.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for initiation functions and requirements.			
6.	Manual Containment Spray	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 2.a., for initiation functions and requirements.			

(a) When the Containment Inservice Purge System is not isolated.
(b) $\leq$ count rate corresponding to 500 mrem/year whole body and 3000 mrem/year skin due to noble gases at the site boundary.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere closed by a manual or automatic isolation valve, blind flange, or equivalent, or

---------------------------NOTE---------------------------
Penetration flow path(s) providing access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days

B 3.3 INSTRUMENTATION

B 3.3.5 Containment Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment ventilation isolation (CVI) instrumentation closes the containment isolation valves in the Containment Purge (high flow) and Inservice (low flow) Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Containment Inservice (low flow) Purge System may be in use during reactor operation and with the reactor shutdown except during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 50 hours). The Containment Purge (high flow) System may be in use with the reactor shutdown, except during handling of recently irradiated fuel.

Containment ventilation isolation initiates on a safety injection (SI) signal, by manual actuation of containment isolation, or by manual actuation of containment spray. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Three radiation monitoring channels are also provided as input to CVI. One channel measures gaseous radiation in containment exhaust air. This channel provides an input to one train of CVI actuation relay logic. The other two channels measure either gaseous or particulate containment exhaust air radiation. These two channels provide inputs to the other train of CVI actuation relay logic where either channel will actuate the train. These three detectors will respond to most events that release radiation to containment. Since the monitors constitute a sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY.

BASES

BACKGROUND (continued)	Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from any one of the three channels initiates one train of CVI logic, which closes one supply and one exhaust containment isolation valve in the Containment Purge (high flow) System and Inservice (low flow) Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."
APPLICABLE SAFETY ANALYSES	The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment exhaust air radiation monitors act as backup to the SI signal to ensure closing of the purge and exhaust valves. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits. Due to radioactive decay, containment is only required to isolate during a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 50 hours). The CVI instrumentation satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).
LCO	The LCO requirements ensure that the instrumentation necessary to initiate CVI, listed in Table 3.3.5-1, is OPERABLE. 1. <u>Manual Initiation</u> The LCO requires two channels OPERABLE. The operator can initiate CVI at any time by using either of two switches in

BASES

LCO

1. Manual Initiation (continued)

the control room. This action will cause actuation of one train of Containment Purge and Inservice Purge System containment isolation valves in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one switch and the interconnecting wiring to the valves.

2. Automatic Actuation Relay Logic

The LCO requires two trains of CVI Relay Logic OPERABLE to ensure that no single random failure can prevent automatic actuation.

The CVI Automatic Actuation Relay Logic consists of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3.b, Containment Isolation. The applicable MODES and specified conditions for the CVI portion of these Functions are different and less restrictive than those for their containment isolation and SI roles. If one or more of the SI or containment isolation Functions becomes inoperable in such a manner that only the CVI Function is affected, the Conditions applicable to their SI and containment isolation Functions need not be entered. The less restrictive Actions specified for inoperability of the CVI Functions specify sufficient compensatory measures for this case.

BASES

LCO (continued)

3. High Radiation in Exhaust Air

The LCO specifies two required channels of radiation monitors, one per train, to ensure that the radiation monitoring instrumentation necessary to initiate CVI remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups, and sample pump operation as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

4. Manual Containment Isolation

Refer to LCO 3.3.2, Function 3.a., for initiating Functions and requirements.

5. Safety Injection

Refer to LCO 3.3.2, Function 1, for initiating Functions and requirements.

6. Manual Containment Spray

Refer to LCO 3.3.2, Function 2, for initiating Functions and requirements.

APPLICABILITY

All Functions in Table 3.3.5-1 are required to be OPERABLE in MODES 1, 2, 3, and 4 when the Containment Inservice (low flow) Purge System is not isolated.

BASES (continued)

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the trip setpoint is less conservative than the Allowable Value, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one CVI radiation monitor channel.

The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that the remaining channels will respond to events.

B.1

Condition B applies to all CVI Functions and addresses the train orientation for these Functions.

If a train is inoperable, two required radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is

BASES

ACTIONS (continued)

met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4 when the Containment Inservice Purge System is not isolated.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which CVI Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1 (continued)

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.5.2

SR 3.3.5.2 is the performance of an ACTUATION LOGIC TEST. This test is performed every 31 days on a STAGGERED TEST BASES. The test includes actuation of the master and slave relays whose contact outputs remain within the logic. The test condition inhibits actuation of the masters whose contact outputs provide direct equipment actuation. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.5.3

A COT is performed every 31 days on each required channel to ensure the entire channel will perform the intended Function. The setpoint shall be left consistent with the current unit specific procedure tolerance.

SR 3.3.5.4

SR 3.3.5.4 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. This test is performed every 24 months.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.5

SR 3.3.5.5 is the performance of a TADOT. This test is a check of the Manual Initiation Function and is performed every 24 months. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.5.6

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies (i.e. fuel that has occupied part of a critical reactor core within the previous 50 hours) within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent and requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

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

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

BASES

BACKGROUND (continued)

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

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

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Containment Purge and Exhaust System includes two subsystems, Containment Purge and Containment Inservice Purge. The containment purge subsystem includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a minipurge system referred to as containment inservice purge, includes a 14 inch purge penetration and an 18 inch exhaust penetration.

During MODES 1, 2, 3, and 4, the two valves in each of the containment purge and exhaust penetrations are secured in the closed position, or the penetrations may be blank flanged. The two valves in each of the two containment inservice purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation System.

BASES

BACKGROUND (continued)	In MODE 6, during handling of recent irradiated fuel, the Containment Purge and Containment Inservice Purge systems must remain closed. The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, or by a manual isolation valve, or blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during recently irradiated fuel movements.
APPLICABLE SAFETY ANALYSES	During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.2, "Refueling Cavity Water Level," and irradiated fuel movement with containment closure capability or a minimum decay time of 50 hours without containment closure capability ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 50.67. The acceptance limit for offsite radiation exposure is 25% of 10 CFR 50.67 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 50.67 limits). Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

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

This LCO limits the consequences of a fuel handling accident involving irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment.

The LCO requires penetrations providing direct access from the containment atmosphere to the outside atmosphere to be closed, including containment purge and exhaust penetrations.

BASES (continued)

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident.

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Due to radioactive decay, a fuel handling accident involving handling fuel not recently irradiated (i.e., fuel that has not been occupied part of a critical reactor core within the previous 50 hours) will results in doses that are well within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

BASES (continued)

ACTIONS

A.1

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

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position.

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

REFERENCES

1. USAR, Section 14.5.

EXHIBIT E

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Letter PI-04-001
SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO CONTAINMENT TECHNICAL SPECIFICATIONS

LIST OF COMMITMENTS

The following table identifies those actions to which NMC committed in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct question regarding these commitments to Mr. Gabe Salamon at Prairie Island Nuclear Generating Plant, (651) 388-1121.

REGULATORY COMMITMENT	DUE DATE
Implement the guidelines of NUMARC 93-01, Revision 3, Section 11.3.6, "Assessment Methods for Shutdown Conditions," Subsection 11.3.6.5 quoted in letter PI-04-001	Implementation date of the license amendment requested in letter PI-04-001

EXHIBIT F

SAFETY ANALYSIS

FOR

LICENSE AMENDMENT REQUEST

SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

FUEL HANDLING ACCIDENT ANALYSIS

License Amendment Request Letter PI-04-001

SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO CONTAINMENT TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

1.0 Introduction

1.1 Evaluation Overview and Objective

The objective of this evaluation is to document the Prairie Island Nuclear Generating Plant (PINGP) selective implementation of the Alternative Source Terms (AST) for the Fuel Handling Accident (FHA) offsite and control room doses in accordance with 10 CFR 50.67 (Reference 3) as described in Regulatory Guide (RG) 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." (Reference 2) The revised analysis may be used to support Technical Specifications (TS) changes proposed in this license amendment request and future changes which may be proposed. Included in the analysis is the use of updated control room atmospheric dispersion factors based on the ARCON96 methodology. (Reference 8)

1.2 Changes to the PINGP Design and Licensing Basis

The following denotes the more significant changes to the PINGP design and licensing bases related to the FHA that are to be considered:

1. The AST methodology is adopted for the composition and timing of radiation releases, as well as accident specific modeling in lieu of RG 1.25.
2. Atmospheric dispersion factors for the control room intake are reanalyzed for existing pathways using ARCON96.
3. The assumed unfiltered inleakage to the Control Room is increased from 44 cfm to 410 cfm (400 cfm for unfiltered inleakage and 10 cfm for ingress and egress).
4. The discharge/decay time prior to fuel movement considered in the FHA is reduced from 100 hours to 50 hours.
5. No credit is taken for containment closure or ventilation filtration/isolation for the FHA occurring inside of containment.
6. No credit is taken for isolation or ventilation filtration for the FHA occurring in the spent fuel pool.

1.3 Deviations from the Regulatory Guideline

No exceptions were taken from the analysis guidance provided in Appendix B of RG 1.183 for the FHA.

2.0 Fuel Handling Accident Scenario

2.1 Introduction

The Prairie Island Nuclear Generating Plant (PINGP) licensing basis for the FHA is currently based on the methodology and assumptions that are derived from RG 1.25 and Standard Review Plan 15.7.4. (References 1 and 2) This analysis is presented in Chapter 14 of the PINGP Updated Safety Analysis Report (USAR), Section 14.5.1 (Reference 14).

RG 1.183 (Reference 3) provides guidance on use of an alternative source term (AST) for use in design basis radiological consequences analyses, as allowed by 10 CFR 50.67 (Reference 4). The offsite and control room radiological consequences for PINGP are re-evaluated using the AST methodology as established in RG 1.183. A brief description of the events addressed by the FHA analysis, input values and assumptions, and the consequences of the accident are presented.

The current licensed maximum power level is 1650 MWt; the analysis in this evaluation is based on a core power of 1683 MWt to account for 2% calorimetric uncertainty.

2.2 Current Licensing Basis Description

The possibility of a fuel handling accident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, the rupture of all fuel elements in a withdrawn assembly is assumed as a conservative limit for evaluating the offsite and control room radiological consequences of a fuel handling accident.

The PINGP USAR discusses FHA dose analyses performed using methods prior to RG 1.25 and using the methods in RG 1.25. The following discussion will describe the licensing basis analyses performed using the methods in RG 1.25. The current licensing basis accident analysis for the FHA at the PINGP includes analysis for a FHA inside of Containment or in the Spent Fuel Pool. Both analyses assume a source term derived from reactor power operations at 1683 MWt (1650 MWt + 2% calorimetric uncertainty) and that the fuel handling accident occurs 100 hours after reactor shutdown. It is assumed that an assembly is dropped and the accident results in damage to all rods in the dropped assembly such that the gaseous fission products contained in the fuel-cladding gap are released. The fission product noble gas gap inventories and halogen activities are based on RG 1.25. The damaged fuel assembly is assumed to have been operating at 1.65 times average core power (based on maximum fuel rod radial peaking factor). For the FHA inside of Containment, the activity is released directly to the atmosphere through open Containment Airlock doors. For the FHA in the SFP, the activity is released through the SFP special ventilation system, where it is filtered (HEPA filters and charcoal adsorbers) prior to release. An overall pool decontamination factor (DF) of 100 for iodine is applied, but no DF is applied to the noble gas releases. Thyroid doses are calculated at the exclusion area boundary (EAB) and the control room (for the FHA inside of Containment). Whole body and thyroid dose are calculated at the EAB for the FHA inside of the SFP. The thyroid doses for the current licensing basis are determined from the iodine dose conversion factors from ICRP Publication 30 for the FHA inside of Containment and ICRP Publication 2 for the FHA inside of the SFP. The doses are summarized in Table 2-1.

Table 2-1: PINGP Current Licensing Basis FHA Dose Summary

FHA Inside Containment

Location	Whole Body (rem)	Thyroid (rem)
Exclusion Area Boundary	Not Calculated	61
Control Room	Not Calculated	2.7

FHA Inside Spent Fuel Pool

Location	Whole Body (rem)	Thyroid (rem)
Exclusion Area Boundary	0.555	23.6

The radiological consequences of the FHA as defined in the current licensing basis are well within the EAB and LPZ dose limits of 10 CFR 100 (300 rem thyroid and 25 rem whole body). "Well within" is defined by SRP 15.7.4 (Reference 2) as 25% or less of the 10 CFR 100 limits.

2.3 Proposed Licensing Basis Description

As discussed above, FHA methodology used in the existing design basis accident analysis discussed in the PINGP USAR is to be updated to reflect the guidance provided in RG 1.183 (Reference 3). The new FHA analysis includes new control room atmospheric dispersion factors developed using ARCON96. The 30-day doses to an operator in the control room due to inhalation of and submersion in the airborne radioactivity releases are developed for FHA. The worst 2-hour period dose at the EAB and the dose at the LPZ for the duration of the release are calculated for the postulated airborne radioactivity releases. For the EAB and the LPZ the reported dose is based on the 30 day time period; which is essentially the same as the 2-hour dose due to the assumption that all of the activity is released within two hours. This represents the post accident dose to the public due to inhalation and submersion.

Under the proposed accident methodology for the FHA, a fuel assembly is assumed to be dropped and damaged during fuel handling. Analysis of the accident is performed with assumptions selected such that the results are bounding for the accident occurring either inside containment or the spent fuel pool. The activity from the damaged assembly is released over two hours to the outside atmosphere taking no credit for hold-up or ventilation system filtration.

This section describes the assumptions and analyses performed to determine the amount of activity released and the resultant offsite and control room doses.

FHA Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 2-4. The analysis involves dropping a recently discharged fuel assembly. Consistent with the current licensing basis, it is assumed that all of the activity in the damaged assembly is released to the pool. Furthermore, it is assumed that the activity that escapes from the containment refueling cavity or the spent fuel pool is released to the environment over a two-hour time period per the guidance of RG 1.183. A constant release rate is assumed for the two-hour time period.

No credit is taken for ventilation filtration system operation in the spent fuel area (i.e., no credit is taken for spent fuel pool special ventilation). Similarly, no credit is taken for containment purge or in-service purge supply and exhaust system closure or filtration capability. In addition, no credit is taken for the containment equipment hatch placement or closure nor is credit taken for having the containment air lock doors closed. Since the assumptions and parameters used to model the release due to a FHA inside containment are identical to those for a FHA in the spent fuel pool, except for different control room intake atmospheric dispersion factor values (χ/Qs) for the different release paths, the activity released is the same regardless of the location of the accident. In order to bound the accident occurring either inside containment or in the spent fuel pool, the location with the highest χ/Q value is assumed. Discussion of the control room and offsite χ/Qs is in Section 5.0.

Consistent with RG 1.183 (Position 1.2 of Appendix B), the radionuclides considered for release are xenons, kryptons, halogens, cesiums, and rubidiums. The list of xenons, kryptons, and halogens considered is given in Table 2-4. These values are based on 1683 MWt core power. The alkali metals, cesium and rubidium are not included in this analysis because they are not assumed to be released from the pool. Per RG 1.183, Appendix B, the cesium and rubidium (particulate radionuclides) released from the damaged fuel rods are assumed to be retained by the water in the refueling cavity and would not be available for release.

Consistent with the current licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. The inventory in the damaged assembly is based on the assumption that the subject fuel assembly has been operated at the maximum radial peaking factor of 1.65 times the average core power. It is assumed that the dropped assembly has been discharged from the core 50 hours after reactor shutdown; therefore, a decay time of 50 hours is applied to the activities in the analysis. The basis for the core activity is described in Section 4.1

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, 10% for Kr-85 and 5% for all other noble gas and iodine nuclides. Footnote 11 to Table 3 in RG 1.183 indicates that these gap fractions are acceptable "with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU." PINGP fuel design and fuel management practices provide for exceeding linear heat generation rate (LHGR) of 6.3 kw/ft with burnups exceeding 54 GWD/MTU. Site-specific analyses were performed that show that the gap fractions in Table 3 of RG 1.183 are bounding. The site-specific analysis of the gap fractions is described in Section 4.2.

In accordance with RG 1.183, the iodine species in the pool is 99.85% elemental and 0.15% organic. This is based on the chemical form of the halogens released from the fuel to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. It assumed that all CsI instantaneously dissociates in the water and re-evolves as elemental. Thus, 99.85% of the iodine released is elemental.

An effective decontamination factor (DF) of 200 for iodine, as provided in RG 1.183, is used in the analysis to account for scrubbing of the iodine in the pool liquid. A DF of 200 is applicable to PINGP as the minimum water level requirement of RG 1.183, Appendix B, Section 2 is satisfied. Specifically, PINGP Technical Specification Surveillance Requirement (SR) 3.9.2.1, "Refueling Cavity Water Level," requires that a minimum of 23 feet of water above the top of the reactor vessel flange be maintained during movement of irradiated assemblies within containment. Similarly, SR 3.7.15.1, "Fuel Storage Pool Water Level," requires a minimum of 23 feet of water over the top of the assemblies be maintained during movement of irradiated fuel assemblies in the spent fuel storage pool. Because 99.85% of the iodine is in the elemental form, an elemental DF of 285 is applied in order to achieve an overall DF of 200. No DF is applied to the noble gas releases and an infinite DF is applied to the particulate radionuclides (i.e., the cesium and rubidium).

No credit is taken for removal of iodine by containment and spent fuel pool building ventilation systems' filters nor is credit taken for isolation of release paths. It is assumed that the activity is released from the pool to the outside atmosphere over a 2-hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open (and other TS changes).

The EAB dose is calculated for the worst 2-hour period, the LPZ dose is calculated for the release duration (i.e., two hours), and the control room doses are calculated for 30 days. As seen in the results, the EAB and LPZ dose are reported for the entire 30-day duration.

The Fauske & Associates (FAI) MAAP-AST software code was used to calculate the isotopic releases and resulting radiation doses offsite and in the control room (Reference 15).

Control Room Ventilation Operation

It is assumed that the control room (CR) HVAC system is initially operating in normal mode, whereby fresh air is being brought into the CR unfiltered at a rate of 2000 cfm. Post-accident, the activity level in the Control Room would cause a high radiation signal within the first few seconds. The high radiation signal causes dampers to close automatically isolating the control room envelope (CRE) from the outside air and directing a portion of the recirculated air through PAC filters. Actuation of the system in this manner due to the high radiation signal is conservatively delayed to 2 minutes after event initiation to increase the margin of safety. After isolation and initiation of filtered recirculation, 410 cfm of unfiltered air inleakage is assumed. The 410 cfm of unfiltered inleakage includes 400 cfm for boundary inleakage and 10 cfm for ingress and egress.

Acceptance Criteria

According to RG 1.183, the EAB and LPZ dose acceptance criteria for a fuel handling accident is 6.3 rem TEDE, which is approximately 25% of the 10 CFR 50.67 limit of 25 rem. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

2.4 Results and Conclusions

2.4.1 Offsite

The offsite doses due to a design basis FHA are presented in Table 2-2. These doses are well within the dose limits 10 CFR 50.67 and less than the acceptance criteria of RG 1.183.

Table 2-2: FHA Offsite Dose Results Assuming AST

Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	6.3	1.95
Low Population Zone	6.3	0.53

2.4.2 Control Room

The Control Room dose due to a design basis FHA is presented in Table 2-3. The doses are less than the dose limit of 10 CFR 50.67 and acceptance criteria of RG 1.183.

Table 2-3: FHA Control Room Dose Results Assuming AST

Unfiltered Inleakage (cfm)	Acceptance Criteria (rem)	TEDE (rem)
410	5	4.4

Table 2-4: Assumptions Used for FHA In Containment Dose Analysis (AST)

Core Power Level	1683 MWt
Radial Peaking Factor	1.65
Fuel Damaged	All Rods in 1 assembly
Time from Shutdown before Fuel Movement	50 hrs
Activity in the Damaged Fuel Assembly[1]	
I-131	5.333E+05 Ci
I-132	5.790E+05 Ci
I-133	2.352E+05 Ci
I-134	3.382E-11 Ci
I-135	6.070E+03 Ci
Kr-85m	5.524E+01 Ci
Kr-85	9.850E+03 Ci
Kr-88	1.613E+00 Ci
Xe-131m	6.942E+03 Ci
Xe-133m	2.782E+04 Ci
Xe-133	1.072E+06 Ci
Xe-135m	9.723E+02 Ci
Xe-135	5.722E+04 Ci
Gap Fractions	
I-131	8% of activity
Kr-85	10% of activity
Other Iodine and Noble Gas	5% of activity
Chemical Form of Iodine in Pool	
Cesium iodide (CsI)	95%
Elemental	4.85%
Organic	0.15%
Water Depth (minimum)	23 feet
Overall Pool Iodine Scrubbing Factor (DF)	200
Noble Gas Scrubbing Factor (DF)	1.0
Particulate Scrubbing Factor (DF)	Infinite
Filter Efficiency – (SFP Special Vent)	No filtration assumed
Isolation of Release	No isolation assumed
Time to Release All Activity	2 hrs
Atmospheric Dispersion Factors (χ/Q)	
Control Room, Fuel Handling Structure	1.44E-02 sec/m^3
Exclusion Boundary Area (715 m)	6.49E-04 sec/m^3
Low Population Zone (2414 m)	1.77E-04 sec/m^3
Control Room Isolation: Actuation Signal/Timing	
Radiation Monitor High Set-point	<1E-04 µCi/cc Xe-133
Actuation of High Radiation Signal	2 minutes

[1] The activity values have been decayed by 50 hours and adjusted by the radial peaking factor, but have not been adjusted for the release fractions or the pool scrubbing factors.

3.0 Dose Calculation

3.1 Input Parameters and Assumptions

The total effective dose equivalent (TEDE) doses are determined at the exclusion area boundary (EAB) for the worst 2-hour interval. The TEDE doses at the low population zone (LPZ) are determined for the duration of the release. The dose reported for the EAB and the LPZ is for the entire 30 days; which, given the two hour release time frame, will bound the worst 2-hour interval. For the control room (CR) personnel, dose is determined for the duration of the event (i.e., 30 days). The interval for determining control room dose is extended beyond the time when the releases are terminated in order to account for the additional dose to the operators in the control room due to the activity that is assumed to be circulating within the control room envelope.

The TEDE dose is equivalent to the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. Effective dose equivalent (EDE) is used in lieu of DDE in determining the contribution of external dose to the TEDE consistent with RG 1.183. The dose conversion factors (DCFs) used in determining the CEDE dose are from the EPA Federal Guidance Report No. 11 (Reference 5) and are given in Table 3-1. The dose conversion factors used in determining the EDE dose are from the EPA Federal Guidance Report No. 12 (Reference 6) and are listed in Table 3-2.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table 3-3 and Table 3-4, respectively. The offsite atmospheric dispersion factors are identical to those currently used in the PINGP USAR radiological accident analyses, previously found acceptable by the AEC in Reference 12.

Parameters used in the control room personnel dose calculations are provided in Table 3-5, and include the normal operation flow rates, the post-accident operation flow rates, control room volume, filter efficiencies and control room operator breathing rates. Atmospheric dispersion factor is for the most limiting release point and is calculated to the control room intake. The limiting atmospheric dispersion factor is applied to the unfiltered inleakage value as well.

The FHA accident assumes an unfiltered inleakage value of 410 cfm into the control room. This includes 10 cfm for ingress and egress, and 400 cfm for inleakage through the CRE. Inleakage testing performed, using tracer gas testing methods, in 1998 showed that the actual inleakage was much less than 400 cfm. Following repairs and enhancements to the system boundary (primarily door seal replacements and replacing louver style outside air dampers with bubble-tight dampers), the limiting inleakage result from the testing was 160 ± 5 cfm. Plans for future testing are addressed in the PINGP 180 day response to NRC Generic Letter 2003-01 (References 13, 25).

No credit is taken for the radioactive decay during release and transport or for cloud depletion by ground deposition during transport to the control room, exclusion area boundary (EAB) or the low population zone (LPZ). Decay is a depletion mechanism credited only for a source term prior to release to the atmosphere and for activity after it enters the control room. Decay constants for each nuclide are provided in Table 3-6.

3.2 Dose Calculation Models

Offsite Dose Calculation Models

The TEDE dose is calculated for the worst 2-hour period at the EAB. At the LPZ the TEDE dose is calculated up to the time all releases are terminated. The TEDE doses are obtained by combining the CEDE doses and the EDE doses.

Offsite inhalation doses (CEDE) are calculated using the following equation:

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} (BR)_j (\chi/Q)_j \right) \right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem).
DCF_i = CEDE dose conversion factor (rem/Ci) via inhalation for isotope i
$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)
$(BR)_j$ = breathing rate (m^3/sec) during time interval j
$(\chi/Q)_j$ = atmospheric dispersion factor (sec/m^3) during time interval j

Offsite external exposure (EDE) doses are calculated using the following equation:

$$D_{EDE} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} (\chi/Q)_j \right) \right]$$

where:

D_{EDE} = external exposure dose via cloud immersion (rem)
DCF_i = EDE dose conversion factor (rem·m^3/Ci·sec) via external exposure for isotope i
$(IAR)_{ij}$ = integrated activity (Ci) of isotope i released during the time interval j
$(\chi/Q)_j$ = atmospheric dispersion factor (sec/m^3) during time interval j

Control Room Dose Calculation Models

CEDE (doses due to inhalation) and EDE (doses due to external exposure) are calculated for 30 days in the control room. The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake are used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room is used to calculate the concentration of activity in the control room. Control room parameters used in the analyses are presented in Table 3-5. Control room atmospheric dispersion factors used in the FHA are provided in Table 2-4.

Control room inhalation doses are calculated using the following equation:

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j * (OF)_j \right) \right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem)
DCF_i = CEDE dose conversion factor (rem/Ci) via inhalation for isotope i
$Conc_{ij}$ = concentration (Ci-sec/m^3) in the control room of isotope i, during time interval j, calculated dependent upon inleakage
$(BR)_j$ = breathing rate (m^3/sec) during time interval j
$(OF)_j$ = occupancy factor during time interval j

Control room external exposure doses are calculated using the following equation:

$$D_{EDE} = \left(\frac{1}{GF} \right) * \sum_i DCF_i \left(\sum_j Conc_{ij} * (OF)_j \right)$$

where:

D_{EDE} = external exposure dose via cloud immersion in rem.
GF = geometry factor, calculated based on RG 1.183, using the equation:
$GF = \frac{1173}{V^{0.338}}$, where V is the control room volume in ft^3
DCF_i = EDE dose conversion factor (rem·m^3/Ci·sec) via external exposure for isotope i
$Conc_{ij}$ = concentration (Ci-sec/m^3) in the control room of isotope i, during time interval j, calculated dependent upon inleakage
$(OF)_j$ = occupancy factor during time interval j

Table 3-1: Committed Effective Dose Equivalent Dose Conversion Factors

Isotope	DCF (rem/curie)	Isotope	DCF (rem/curie)
I-131	3.29E+04	Cs-134	4.63E+04
I-132	3.81E+02	Cs-136	7.33E+03
I-133	5.85E+03	Cs-137	3.19E+04
I-134	1.31E+02	Rb-86	6.62E+03
I-135	1.23E+03	Rb-88	8.36E+01
		Rb-89	4.29E+01
Kr-85m	N/A		
Kr-85	N/A	Ru-103	8.95E+03
Kr-87	N/A	Ru-105	4.55E+02
Kr-88	N/A	Ru-106	4.77E+05
Xe-131m	N/A	Rh-105	9.55E+02
Xe-133m	N/A	Mo-99	3.96E+03
Xe-133	N/A	Tc-99m	3.26E+01
Xe-135m	N/A		
Xe-135	N/A	Y-90	8.44E+03
Xe-138	N/A	Y-91	4.89E+04
		Y-92	7.80E+02
Te-127	3.18E+02	Y-93	2.15E+03
Te-127m	2.15E+04	Nb-95	5.81E+03
Te-129m	2.39E+04	Zr-95	2.37E+04
Te-129	8.95E+01	Zr-97	4.33E+03
Te-131m	6.40E+03	La-140	4.85E+03
Te-132	9.44E+03	La-141	5.81E+02
Te-134	1.27E+02	La-142	2.53E+02
Sb-127	6.03E+03	Nd-147	6.85E+03
Sb-129	6.44E+02	Pr-143	1.09E+04
		Am-241	4.44E+08
Ce-141	8.96E+03	Cm-242	1.73E+07
Ce-143	3.39E+03	Cm-244	2.48E+08
Ce-144	3.74E+05		
Pu-238	3.92E+08	Sr-89	4.14E+04
Pu-239	4.29E+08	Sr-90	1.3E+06
Pu-240	4.29E+08	Sr-91	1.66E+03
Pu-241	8.25E+06	Sr-92	8.07E+02
Np-239	2.51E+03	Ba-139	1.72E+02
		Ba-140	3.74E+03

Table 3-2: Effective Dose Equivalent Dose Conversion Factors

Isotope	DCF (rem·m³/Ci·sec)*	Isotope	DCF (rem·m³/Ci·sec)*
I-131	6.734E-2	Cs-134	0.2801
I-132	0.4144	Cs-136	0.3922
I-133	0.1088	Cs-137	2.864E-05
I-134	0.4810	Rb-86	1.780E-02
I-135	0.2953	Rb-88	0.1243
		Rb-89	0.3922
Kr-85m	2.768E-02		
Kr-85	4.403E-04	Ru-103	8.325E-02
Kr-87	0.1524	Ru-105	0.1410
Kr-88	0.3774	Ru-106	0.0
Xe-131m	1.439E-03	Rh-105	1.376E-02
Xe-133m	5.069E-03	Mo-99	2.694E-02
Xe-133	5.772E-03	Tc-99m	2.179E-02
Xe-135m	7.548E-02		
Xe-135	4.403E-02	Y-90	7.030E-04
Xe-138	0.2135	Y-91	9.620E-04
		Y-92	4.810E-02
Te-127	8.954E-04	Y-93	1.776E-02
Te-127m	5.439E-04	Nb-95	0.1384
Te-129m	5.735E-03	Zr-95	0.1332
Te-129	1.018E-02	Zr-97	3.337E-02
Te-131m	0.2594	La-140	0.4329
Te-132	3.811E-02	La-141	8.843E-03
Te-134	0.1569	La-142	0.5328
Sb-127	0.1232	Nd-147	2.290E-02
Sb-129	0.2642		
		Pr-143	7.770E-05
Ce-141	1.269E-02	Am-241	3.027E-03
Ce-143	4.773E-02	Cm-242	2.105E-05
Ce-144	3.156E-03	Cm-244	1.817E-05
Pu-238	1.806E-05		
Pu-239	1.569E-05	Sr-89	2.860E-04
Pu-240	1.758E-05	Sr-90	2.786E-05
Pu-241	2.683E-07	Sr-91	0.1277
Np-239	2.845E-02	Sr-92	0.2512
		Ba-139	8.029E-03
		Ba-140	3.175E-02

* Table III.1 in FGR 12 (Reference 5) gives dose conversion factors in Sv·m³/Bq·sec; therefore, each value was multiplied by 3.7E+12 rem/Sv·Bq/Ci to get the units of rem·m³/Ci·sec.

Table 3-3: Offsite Breathing Rates

Time Period	Value
0 - 8 hours	3.5E-04 m^3/sec
8 – 24 hours	1.8E-04 m^3/sec
>24 hours	2.3E-04 m^3/sec

Table 3-4: Offsite Atmospheric Dispersion Factors

Location/Time Interval	Value
Exclusion Area Boundary [ŧ]	6.49E-04 sec/m^3
Low Population Zone [ŧ]	
0 - 8 hours	1.77E-04 sec/m^3

[ŧ] This exclusion area boundary atmospheric dispersion factor is conservatively applied during the entire 30-day duration in the determination of the limiting 2-hour period. The low population zone atmospheric dispersion factor for the 0-8 hour time frame is conservatively applied during the entire 30-day duration.

Table 3-5: Control Room Parameters

Volume	61,315 ft^3
Control Room Unfiltered In-Leakage	410 cfm
Normal Mode Ventilation Flow Rates	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	2000 cfm
Emergency Mode Ventilation Flow Rates	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate *	4000 cfm ± 10%
Unfiltered Makeup Flow Rate	0 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
CR Radiation Monitor Setpoint	< 1.0E-04 μCi/cc for Xe-133
CR Radiation Monitor Location (R-23 & R-24)	HVAC Duct downstream of filter
CR HVAC Emergency Mode Actuation Delay **	2 minutes
Breathing Rate	3.5E-04 m^3/sec
Occupancy Factors	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

* The intake value of 3600 cfm was found to calculate the bounding dose.

** This is a conservative time for aligning the CR HVAC from Normal Mode of operation to Emergency Mode and includes delay time to reach the high radiation setpoint.

Table 3-6: Nuclide Decay Constants

Isotope	Decay Constant (hr^{-1})	Isotope	Decay Constant (hr^{-1})
I-131	0.00359	Cs-134	3.77E-05
I-132	0.301	Cs-136	2.2E-03
I-133	0.0333	Cs-137	2.64E-06
I-134	0.792	Rb-86	1.55E-03
I-135	0.103	Rb-88	2.34
		Rb-89	2.77
Kr-85m	0.155		
Kr-85	7.38E-06	Ru-103	7.31E-04
Kr-87	0.547	Ru-105	0.156
Kr-88	0.244	Ru-106	7.85E-05
Xe-131m	0.00242	Rh-105	1.93E-02
Xe-133m	0.0126	Mo-99	1.03E-02
Xe-133	0.00548	Tc-99m	0.116
Xe-135m	2.63		
Xe-135	0.0753	Y-90	1.08E-02
Xe-138	2.45	Y-91	4.89E-04
		Y-92	0.196
Te-127	7.45E-02	Y-93	0.0686
Te-127m	2.65E-04	Nb-95	8.25E-04
Te-129m	8.5E-04	Zr-95	4.44E-04
Te-129	0.621	Zr-97	4.1E-02
Te-131m	2.31E-02	La-140	1.72E-02
Te-132	8.88E-03	La-141	0.177
Te-134	0.99	La-142	0.495
Sb-127	7.4E-03	Nd-147	2.60E-03
Sb-129	0.16	Pr-143	2.11E-03
		Am-241	1.72E-07
Ce-141	8.89E-04	Cm-242	1.77E-04
Ce-143	0.021	Cm-244	4.37E-06
Ce-144	1.01E-04		
Pu-238	9.8E-07	Sr-89	5.71E-04
Pu-239	3.25E-09	Sr-90	2.75E-06
Pu-240	1.17E-08	Sr-91	0.071
Pu-241	6.0E-06	Sr-92	0.257
Np-239	0.0123	Ba-139	0.501
		Ba-140	2.26E-03

4.0 Radiation Source Terms

4.1 Core Inventory

A new core source term has been calculated for use in the fuel handling accident analysis (References 16 and 17). The inventory of the fission products in the reactor core is based on maximum full-power operation of the core at a power level equal to 1683 MWt and current licensed values of fuel enrichment and burnup. 1683 MWt includes 2% uncertainty above the current licensed core power level 1650 MWt.

The ORIGEN2 computer code (Reference 22) was used to determine the equilibrium core inventory. ORIGEN2 is a versatile point depletion and radioactive decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein.

For the FHA case, the core wide fission product inventories are for a fuel assembly burn-up of 65,000 MWD/MTU. For the current operating cycles, the maximum pin exposure limit is 62,000 MWD/MTU. Thus, the core wide fission product inventory was selected to bound current allowed operating conditions.

The core inventory developed using ORIGEN2 based on the above methodology includes many isotopes that are not dose significant. Those dose significant isotopes relative to the FHA are presented in Table 4-1.

Table 4-1: Equilibrium Core Fission Product Activities at 1683 MWt

Isotope	Activity (Ci)
I-131	4.586E+07
I-132	6.537E+07
I-133	8.927E+07
I-134	9.739E+07
I-135	8.415E+07
Kr-85m	9.154E+06
Kr-85	7.225E+05
Kr-87	1.691E+07
Kr-88	2.366E+07
Xe-131m	5.139E+05
Xe-133m	2.852E+06
Xe-133	8.961E+07
Xe-135m	1.836E+07
Xe-135	1.848E+07
Xe-138	7.061E+07
Cs-134	1.862E+07
Cs-136	4.781E+06
Cs-137	8.643E+06
Rb-86	1.862E+05
Rb-88	2.422E+07
Rb-89	3.053E+07

4.2 Gap Fractions

RG 1.183, Table 3, specifies Fission Product gap inventories for non-LOCA events. Footnote (11) to Table 3 reads:

> "The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission product gas release calculations performed using NRC approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases."

The Prairie Island fuel management program can result in some fuel assemblies being exposed to a maximum linear heat generation rate (LHGR) that exceeds 6.3 kw/ft at fuel burn-ups between 54 and 62 GWD/MTU. Thus, to account for the higher LHGR a site specific analysis was conducted (Reference 18). A computer code was developed by Fauske and Associates, Inc., referred to as GAP (Reference 24), to perform the site specific gap fraction analysis.

The GAP code was developed and qualified as a safety related computer code. The GAP code implements the gap fractional release methodology presented in ANSI/ANS-5.4-1982, "American National Standard Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel" (Reference 23). The nodal input data to GAP for the temperature and specific power distribution as a function of burnup were developed with the Westinghouse PAD 4.0 code based on bounding power histories.

The GAP code is used to determine the gap release fractions for the short-lived and long-lived radionuclides. Both the ANS-5.4-1982 low-temperature and high-temperature release models are used. The bounding value as obtained for either release model is selected for the final result. These specific gap release fractions were then compared to the gap release fractions provided in Table 3 of RG 1.183. The bounding values between the regulatory value and the specific value are then used in the FHA dose analysis. The comparison of the results is shown in Table 4-2.

Table 4-2
Comparison of Gap Fractional Releases for Prairie Island Fuel Rods to RG 1.183, Table 3

Radionuclide or Radionuclide Group	Fractional Release	
	Bounding Prairie Island Result	RG 1.183, Table 3
I-131	0.036	0.08
Kr-85	0.056	0.10
Other Noble Gases	0.024	0.05
Other Halogens	0.013	0.05
Alkali Metals	0.072	0.12

The GAP code predicts that in all cases, the fractions in Table 3 of RG 1.183 are bounding. Therefore, the fractions in Table 3 of RG 1.183 are used in the FHA dose analysis.

5.0 Accident Atmospheric Dispersion Factors (χ/Q)

5.1 Control Room Atmospheric Dispersion Factors

The control room intake χ/Q values for the potential FHA release points are determined using ARCON96, Atmospheric Relative Concentrations in Building Wakes methodology. (Reference 8). Input data consists of hourly on-site meteorological data, release characteristic such as release height, the building area affecting the release, and various receptor parameters such as its distance and direction from the release to the control room air intake and intake height.

A continuous temporally representative 5-year period of hourly average data from the PINGP meteorological tower (i.e., January 1, 1993 through December 31, 1997) is used in this calculation. The PINGP meteorological tower is located such that it satisfies the guidance in RG 1.23 (Reference 9). Redundant wind direction, wind speed and temperature instruments are located at the 10-meter and the 60-meter elevations. Instrument accuracies satisfy RG 1.23, Section C.4. Data collected from these instruments was reviewed daily and weekly by on-site staff. The daily review was performed to identify missing data and identify any irregularities in the data. The weekly review was performed to examine the data for consistency, trends and to ensure that the data is reasonable. The plant procedure for the weekly examination of the data provides criteria for gauging the reasonableness of the data. Quality assurance measures such as monthly tests and annual calibrations of the instruments are performed to ensure data quality and identify any problems. The monthly tests ensure proper operation of the instruments by checking the signals at the instrument rack modules. The monthly tests allow for adjustments if signals are found outside of the desired calibration values. If the results of tests indicate that the instruments are not functioning properly, then corrective actions such as calibrations, repairs, or replacements, are performed. The annual calibrations ensure proper calibration and operation of the instrumentation; which includes physical inspections and maintenance of the transmitters.

Each hour of data, at a minimum, has wind direction, wind speed and temperature at the 10-meter level and at the 60-meter level. The data recovery for each of the five years is greater than 90%. During the Jan. 1, 1993 to Dec. 31, 1997 time period the data was reviewed annually by an independent meteorological organization to ensure that the data is of high quality. The annual review included a general check of all parameters for completeness and reasonableness, comparison to previous years data to identify any trends or notable changes in values, and consistency checks between different heights, parameters, and measurement systems. The conclusions from the annual independent assessments is that the "meteorological data was judged complete, accurate and representative."

<u>Source</u>

The FHA could be postulated to occur in either the spent fuel pool or in containment.

The spent fuel pool (SFP) enclosure is inside of the Auxiliary Building, but outside of the Auxiliary Building Special Ventilation Zone (ABSVZ). This portion of the Auxiliary Building is a steel structure with metal siding that is not leak tight. This will be referred to as the "common area of the Aux Bldg" hence forth. Figure 2 shows the SFP enclosure in relation to this common area of the Aux Bldg. If the Spent Fuel Pool special ventilation system were credited in the dose analyses the release is filtered before being exhausted through the Shield Building Stack. If normal ventilation is operating and credit is not taken for isolation by the high radiation signal, the release is out the normal ventilation exhaust stack; which is farther away than other potential release locations. Without the ventilation systems operating, the radioactivity released from the damaged fuel assembly could exit the SFP enclosure and enter the common area of the Aux Bldg. Activity exiting the common area of the Aux Bldg at the closest point to the CR ventilation intake would provide a bounding atmospheric dispersion factor. Figure 3 shows the relative relationship between the common area of the Aux Bldg and the CR ventilation intakes. Therefore, the analysis is performed assuming that the radioactivity is released through the common area of the Aux Bldg closest to the CR ventilation intake and no credit is taken for isolation of the spent fuel pool structure or operation of the spent fuel pool ventilation systems.

The analysis for the FHA inside of containment is performed assuming that there are no controls on containment boundary during fuel handling. Thus, the leakage could exit Containment and enter the ABSVZ through open containment penetrations, exit containment directly to the atmosphere through the open Equipment Hatch or enter this same common area of the Aux Bldg through an open Containment Maintenance Air Lock. If the leakage entered this common area of the Aux Bldg through an open Maintenance Air Lock it could have the same release path as that described above for the FHA in the SFP enclosure. Leakage into the ABSVZ would need to traverse a torturous path to exit the building and most likely would be filtered by the Auxiliary Building special ventilation system and released through the Shield Building Ventilation Stack. Leakage through the open Equipment Hatch would enter the Annulus and be released to the Shield Building stack or released directly to the outside environment. The distance from Shield Building Ventilation Stack and the Equipment Hatch to the CR Vent Intake is much further than the distance from the common area of the Aux Bldg to the CR Vent Intake. Thus, similar to the FHA in the spent fuel pool assuming all of the leakage escapes through the common area of the Aux Bldg in the area of the building closest to the CR ventilation intake provides a bounding result.

Receptor

The CR ventilation intakes use bubble tight dampers to isolate the control room from the outside environment. However, the FHA analysis conservatively assumes that all of the inleakage to the Control Room Envelope (CRE) occurs through the ventilation intake. The CR ventilation intake is the closest point to the source of the leakage and provides the limiting χ/Q. Using a single point for the inleakage to the CRE also simplifies this determination, as a single receptor location can be used to bound other potential receptor locations for the CRE.

The following assumptions are made for these calculations:

1. The plume centerline from each release is conservatively transported directly over the control room air intake.
2. All releases are assumed to under the influence of the containment building wake.
3. The ARCON96 default wind direction range of 90°, centered on the direction that transports the gaseous effluents from the release points to the receptors is used in the calculation
4. The ARCON96 values for surface roughness length (i.e., 0.20 meter) and sector averaging constant (i.e., 4.3) are consistent with RG 1.194. (Reference 11)
5. All releases are conservatively treated as ground level as there are no release conditions that merit categorization as an elevated release with respect to the PINGP configuration.
6. The release from the common area of the Aux Bldg is assumed to be at the same elevation as the Control Room ventilation intake. This minimizes the source/receptor pair distance and results in a bounding atmospheric dispersion factor.

As part of the AST work, new atmospheric dispersion factors were determined accounting for several possible release locations (Reference 19). For each release location, the receptor is the control room fresh air intake (both 121 and 122 trains were considered). This receptor location is also used conservatively for unfiltered inleakage. Figure 3 shows the release locations with respect to the receptor location.

1. Unit 1 Safety and Relief Valve Group 1
2. Unit 2 Safety and Relief Valve Group 1
3. Unit 1 Safety and Relief Valve Group 2
4. Unit 2 Safety and Relief Valve Group 2
5. Unit 1 Side SFP Normal Vent Supply Damper
6. Unit 2 Side SFP Normal Vent Supply Damper
7. Unit 1 ADV
8. Unit 2 ADV

The ARCON96 input parameters for these source locations to the 121 CR Vent Intake (receptor) are listed in Tables 5-1 through 5-8, respectively. The ARCON96 input parameters for these source locations to the 122 CR Vent Intake (receptor) are listed in Tables 5-9 through 5-16, respectively.

Based on similar direction and distance, the χ/Q values for the common area of the Aux Bldg to the 121 and 122 control room intake can be determined. Table 5-17 and 5-18 show the key parameters of the common area of the Aux Bldg to the 121 and 122 CR vent intake, respectively. As shown the distances and directions are encompassed by the source/receptor pairs analyzed using ARCON96. The initial diffusion coefficients (σ_y and σ_z) for the common area of the Aux Bldg are much larger than the initial diffusion coefficients for the source/receptor pairs analyzed using ARCON96.

The χ/Q values for the 0 - 2 hour time period for all release locations to 121 CR Vent Intake are summarized in Table 5-19. The χ/Q value for the 0 - 2 hour time period for the common area of the Aux Bldg to the 121 CR Vent Intake is determined by plotting the data in Table 5-19 on Figure 5-1 and developing a curve fit equation for calculating the χ/Q value as a function of distance. Figure 5-1 shows very good curve fit to the points in Table 5-19. The minimum distance from the common area of the Aux Bldg to the 121 CR Vent Intake is 18.3 meters (assuming the release location is at the same elevation as the 121 CR Vent Intake; i.e., no accounting for elevation difference when determining the minimum distance from the common area of the Aux Bldg to 121 CR Vent Intake). The calculated χ/Q value for the 0 – 2 hour time frame is shown on Table 5-21.

The χ/Q values for the 0 - 2 hour time period for all release locations to 122 CR Vent Intake are summarized in Table 5-20. The χ/Q value for the 0 - 2 hour time period for the common area of the Aux Bldg to the 122 CR Vent Intake is determined by plotting the data in Table 5-20 on Figure 5-2 and developing a curve fit equation for calculating the χ/Q value as a function of distance. Figure 5-2 shows very good curve fit to the points in Table 5-20. The minimum distance from the common area of the Aux Bldg to the 122 CR Vent Intake is 25.7 meters (assuming the release location is at the same elevation as the 122 CR Vent Intake; i.e., no accounting for elevation difference when determining the minimum distance from the common area of the Aux Bldg to 122 CR Vent Intake). The calculated χ/Q value for the 0 – 2 hour time frame is shown on Table 5-21.

As shown in Table 5-21, the most conservative χ/Q value is for the common area of the Aux Bldg to the 121 CR Vent Intake. This value is used for all CR unfiltered inleakage in the FHA dose analysis.

5.2 Offsite Atmospheric Dispersion Factors

The χ/Q values for the PINGP EAB and the LPZ are those from the current licensing basis (Reference 14, Appendix H, Table XIV). The offsite χ/Q values are presented in Table 5-22. The χ/Q value for the 0 – 8 hour time period is used for the duration of the analysis for the EAB and the LPZ.

Table 5-1: Unit 1 Safety and Relief Valve Group 1 to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	180°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	16.1 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-2: Unit 2 Safety and Relief Valve Group 1 to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	253 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	53.2 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-3: Unit 1 Safety and Relief Valve Group 2 to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	183°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	53.7 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-4: Unit 2 Safety and Relief Valve Group 2 to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	221 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	70.1 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-5: Unit 1 Side SFP Normal Vent Supply Damper to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	23 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	181°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	55.9 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0.4, respectively

Table 5-6: Unit 2 Side SFP Normal Vent Supply Damper to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	23 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	220 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	73.2 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0.4, respectively

Table 5-7: Unit 1 ADV to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	270°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	9.6 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.7, 0, respectively

Table 5-8: Unit 2 ADV to 121 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	270 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	39.8 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.7, 0, respectively

Table 5-9: Unit 1 Safety and Relief Valve Group 1 to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	97°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	67.1 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-10: Unit 2 Safety and Relief Valve Group 1 to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	129 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	12.8 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-11: Unit 1 Safety and Relief Valve Group 2 to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	126°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	75.6 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-12: Unit 2 Safety and Relief Valve Group 2 to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	160 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	48.7 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0, respectively

Table 5-13: Unit 1 Side SFP Normal Vent Supply Damper to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	23 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	128°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	78.6 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0.4, respectively

Table 5-14: Unit 2 Side SFP Normal Vent Supply Damper to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	23 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	161 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	50.6 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.4, 0.4, respectively

Table 5-15: Unit 1 ADV to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	89°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	51.6 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.7, 0, respectively

Table 5-16: Unit 2 ADV to 122 CR Vent Intake

Input Parameter	Value
Meteorological Data	Determined from data collected: 1993-1997
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	60 m
Release Type	Ground
Release Height	28 m
Building Area Perpendicular to Wind Direction	2176 m^2
Effluent Vertical Velocity	0 m/sec
Vent or Stack Flow	0 m^3/sec
Vent or Stack Radius	0 m
Direction to Source	70 °
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	21.9 m
Control Room Air Intake Height	25 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/sec
Surface Roughness Length	0.2 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0.7, 0, respectively

Table 5-17: Common Area of Aux Bldg to 121 CR Vent Intake

Input Parameter	Value
Release Height	29.6 m
Direction to Source	218°
Distance to Control Room Air Intake	18.3 m
Control Room Air Intake Height	25 m
Initial Values of sigma y and sigma z	3.1, 1.7, respectively

Table 5-18: Common Area of Aux Bldg to 122 CR Vent Intake

Input Parameter	Value
Release Height	29.6 m
Direction to Source	105°
Distance to Control Room Air Intake	25.7 m
Control Room Air Intake Height	25 m
Initial Values of sigma y and sigma z	2.6, 1.7, respectively

Table 5-19, Sources to 121 CR Vent Intake

Source/Receptor	Distance (m)[1]	χ/Q (sec/m^3) 0-2 hrs
Unit 1 Safety and Relief Valve Group 1 to 121 CR Vent Intake	16.4	1.87E-02
Unit 2 Safety and Relief Valve Group 1 to 121 CR Vent Intake	53.3	2.01E-03
Unit 1 Safety and Relief Valve Group 2 to 121 CR Vent Intake	53.8	1.89E-03
Unit 2 Safety and Relief Valve Group 2 to 121 CR Vent Intake	70.2	1.10E-03
Unit 1 side SFP Normal Vent Supply Damper to 121 CR Vent Intake	55.9	1.73E-03
Unit 2 side SFP Normal Vent Supply Damper to 121 CR Vent Intake	73.2	9.75E-04
Unit 1 ADV to 121 CR Vent Intake	10.1	4.66E-02
Unit 2 ADV to 121 CR Vent Intake	39.9	3.60E-03

1. Slant distance is used in lieu of horizontal distance to maximize the distance.

Figure 5-1, Plot of Sources to 121 CR Vent Intake

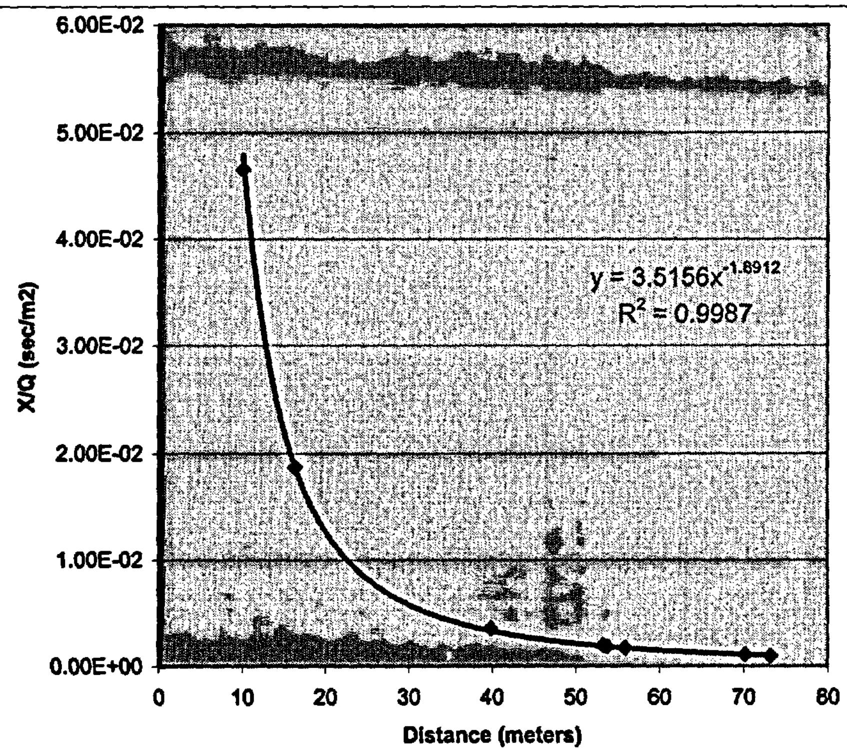

Table 5-20 , Sources to 122 CR Vent Intake

Source/Receptor	Distance (m)[1]	χ/Q (sec/m^3) 0-2 hrs
Unit 1 Safety and Relief Valve Group 1 to 122 CR Vent Intake	67.2	1.34E-03
Unit 2 Safety and Relief Valve Group 1 to 122 CR Vent Intake	13.1	3.06E-02
Unit 1 Safety and Relief Valve Group 2 to 122 CR Vent Intake	75.7	1.11E-03
Unit 2 Safety and Relief Valve Group 2 to 122 CR Vent Intake	48.8	2.47E-03
Unit 1 side SFP Normal Vent Supply Damper to 122 CR Vent Intake	78.6	1.00E-03
Unit 2 side SFP Normal Vent Supply Damper to 122 CR Vent Intake	50.6	2.18E-03
Unit 1 ADV to 122 CR Vent Intake	51.7	2.12E-03
Unit 2 ADV to 122 CR Vent Intake	22.1	9.89E-03

1. Slant distance is used in lieu of horizontal distance to maximize the distance.

Figure 5-2, Plot of Sources to 122 CR Vent Intake

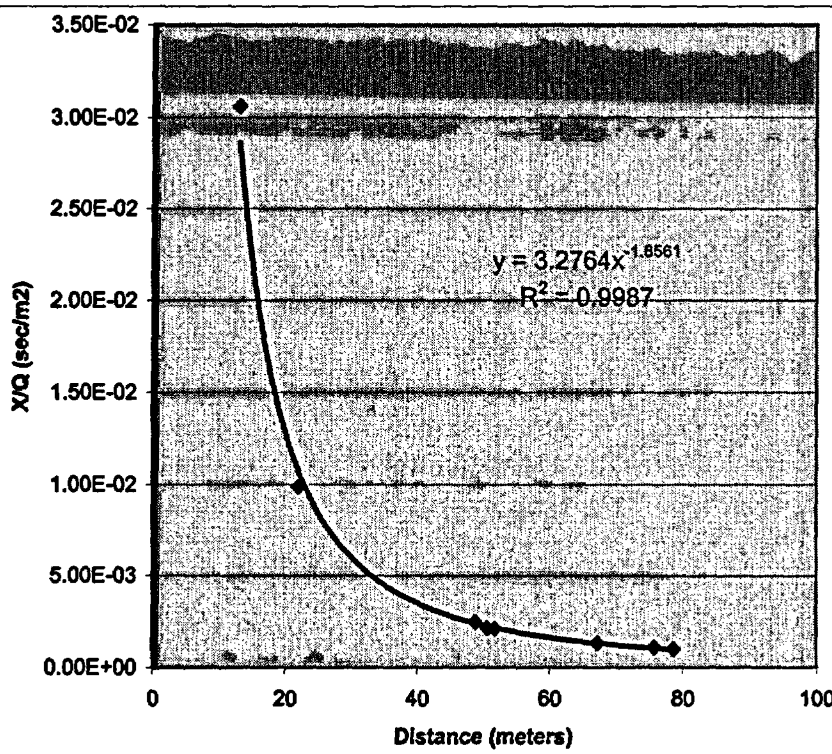

Table 5-21: Prairie Island Control Room Atmospheric Dispersion Factors (sec/m³)

Source/Receptor Pair	0 – 2 hr Time Period
Common Area of Aux Bldg to 121 CR Vent Intake	1.44E-02
Common Area of Aux Bldg to 122 CR Vent Intake	7.91E-03

Table 5-22: Prairie Island Offsite Atmospheric Dispersion Factors (sec/m³)

	Averaging Period			
Receptor Location	0 – 8 hr	8 – 24 hr	1 – 4 d	4 – 30 d
Exclusion Area Boundary	6.49E-04	2.59E-04	4.65E-05	6.90E-06
Low Population Zone	1.77E-04	3.99E-05	7.12E-06	1.04E-07

The Atmospheric Dispersion Factor for the 0 – 8 hour time period is used for the entire duration of the FHA to determine the dose at both the EAB and the LPZ.

6.0 Control Room Envelope

6.1 Control Room Licensing Basis

The PINGP control room design was implemented and licensed under AEC draft General Design Criterion (GDC) 11, dated July 1967, which existed before the issuance of the GDC in 10 CFR 50, Appendix A. Simply stated, AEC Draft GDC 11 requires that the facility shall be provided with a control room from which actions to maintain safe operational status can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the CR under any credible post-accident condition.

Design and system reviews stemming from the post-TMI initiatives demonstrate that the system is capable of meeting the dose limits of 10 CFR 50 Appendix A GDC-19 as required by NUREG-0737, Item III.D.3.4. The design factors affecting the system's ability to meet the above dose limits include: actuation of the ventilation system to the emergency mode on a Safety Injection or High Radiation signal, emergency filtration flow rate 4000 cfm ± 10%, and meeting minimum filtration efficiencies specified for the HEPA filters and charcoal adsorbers.

6.2 Control Room Design and Ventilation System Description

The PINGP control room envelope is located at elevation 735' within the Auxiliary Building approximately equidistance between Unit 1 and Unit 2. The control room envelope consists of the control room and the two mechanical equipment rooms (referred to as the 121 and 122 Chiller Rooms). The control room ventilation system is entirely located within the two chiller rooms, with the exception of the outside air supply. The outside air supply dampers are located at the envelope boundary. The chiller rooms are located directly above the control room at elevation 755'. The cable spreading room on the 715' elevation (directly below the control room) is not part of the control room envelope. Figures 1 and 2 show the relation of the control room and the chiller rooms within the Auxiliary Building.

Two redundant radiation monitors with control functions are located within the control room envelope. The radiation monitors (R-23 and R-24) are calibrated to Xe-133 and physically located within the Control Room. The radiation monitors sensing lines penetrate the control room supply ductwork downstream of the control room HVAC filter unit inside of the Control Room. A "high" signal from either detector will automatically switch the control room ventilation system from the normal mode of operation to the emergency mode. The descriptions of these modes are given in the following discussion.

<u>Normal Mode</u>

During normal operation one train is running and the other train is in standby. For the operating train, the air handler would be operating and the clean-up fan would be in standby with no air flow through the PAC filter. During normal operation, the operating train recirculates the control room envelope air and draws in fresh air. Air is exhausted from the Control Room Envelope at a rate equivalent to the quantity of fresh air brought in. The design flow rates are 10,000 cfm recirculation flow rate and 2000 cfm fresh air for a total air handler flow rate of 12,000 cfm. The input for normal operation in the dose analysis is the fresh air supply flow rate. An equivalent input is also included for the exhaust flow rate. The normal recirculation flow rate is not an input in the dose analysis.

Emergency Mode

In response to a Safety Injection or high radiation signal, both trains start and are automatically aligned to isolate the fresh air and start and align a portion of the recirculation air flow through the Clean-Up fan. The portion of the air that is drawn by the clean-up fan passes through a PAC filter that is credited in the dose analysis. In this alignment, the system is recirculating and filtering the control room atmosphere. To account for a single active failure, only one train of control room ventilation system is credited in the dose analysis. In the emergency mode, the clean-up fan is designed to provide 4000 cfm ± 10%. For the FHA dose analysis, the lower recirculated air flow rate, 3600 cfm, results in the bounding dose consequence to the control room operator.

6.3 Control Room Habitability

In 1998, PINGP conducted tracer gas testing of the control room envelope to determine unfiltered inleakage rates. Several different integrated system configurations were used for the testing based on train A, train B or both trains of control room ventilation operating coupled with different alignments for the ventilation systems in the ABSVZ and Turbine Building. Two different sets of tests were performed. The initial set of testing in January 1998 predicted unfiltered inleakage rates greater than the values used in the dose analyses. This was reported in LER 98-02-00, dated Feb. 18, 1998. Repairs were made to the boundary, primarily replacing door seals. Subsequent testing in July 1998 produced the following results.

System Configuration	Inleakage Rate (cfm)
High Rad	160 ± 5
SI	145 ± 5

The system configuration for the High-Rad vs. the SI signal affects the ventilation systems in the adjacent spaces (ABSVZ and Turbine Bldg); however, there is no difference in the alignment of the emergency configuration of the control room ventilation system in response to a High Rad vs a SI signal. With the High Rad signal, the configuration of the ventilation systems in the adjacent areas is more representative of a FHA. With the SI signal, the configuration of the ventilation systems in the adjacent areas is more representative of a LOCA event. Thus, for the FHA scenario, the maximum unfiltered inleakage from testing (including uncertainties) is 165 cfm. The FHA dose analysis assumes an unfiltered inleakage of 400 cfm plus an additional 10 cfm for ingress/egress, for a total unfiltered inleakage of 410 cfm.

In addition, PINGP has undertaken other actions to assure control room habitability. Outside air dampers were originally of a louver style design. These have been replaced with bubble tight dampers (extremely low-leakage dampers). As discussed previously, during the initial testing, door seals were identified as a significant vulnerability to CRE integrity and that replacing the door seals greatly reduced the inleakage. To maintain the CRE, preventative maintenance procedures have been implemented to inspect and replace these door seals on a regular basis.

Any commitments for future testing of the CRE are addressed in the 180-day response to NRC Generic Letter 2003-01, "Control Room Habitability" (References 13 and 25).

7.0 Conclusion

An assessment of the radiological consequences due to a FHA using the AST methodology concludes that the EAB, LPZ, and control room doses are within the limits of 10 CFR 50.67 and within the acceptance criteria of RG 1.183 without crediting closure of the containment equipment hatch, personnel and maintenance air lock doors, and SFP or containment ventilation filtration capabilities. Because the FHA can occur in the either containment or the spent fuel pool, the most limiting atmospheric dispersion factor was chosen to bound this accident. Based on results from the ARCON96 code, the common area of the Aux Bldg to the 121 CR vent intake provides the most limiting atmospheric dispersion.

In conclusion, there will be no adverse impact on the public health and safety.

8.0 References

1. USNRC, "Assumptions Used for Evaluation the Potential Radiological consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling And Pressurized Water Reactors," Regulatory Guide 1.25, March 23, 1972.

2. USNRC, "Standard Review Plan," NUREG-0800, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, July 1981.

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

4. 10 CFR 50.67, "Accident Source Term."

5. USEPA, "Limiting values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation Submersion, and Ingestion," Federal Guidance Report No. 11, September 1988.

6. USEPA, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, September 1993.

7. Prairie Island Nuclear Generating Plant Technical Specifications.

8. NUREG/CR-6331, Rev. 1, Atmospheric Relative Concentrations in Building Wakes, J.V. Ramsdell, C.A. Simonen, Pacific Northwest National Laboratory, 1997.

9. USNRC, Onsite Meteorological Programs, Regulatory Guide 1.23, February 17, 1972.

10. Not Used.

11. USNRC, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Regulatory Guide 1.194, June 2003.

12. U.S. Atomic Energy Commission, Directorate of Licensing, Safety Evaluation of the Prairie Island Nuclear Generating Plant, Units 1&2.

13. NRC Generic Letter 2003-01, Control Room Habitability, dated June 12, 2003.

14. Prairie Island Nuclear Generating Plant Updated Safety Analysis Report.

15. Fauske and Associates, Inc. (FAI) MAAP-AST Computer Code.

16. Calculation GEN-PI-046, Fission Product Inventories for Use in Prairie Island Alternate Source Term Assessments.

17. Calculation GEN-PI-047, Fuel Handling Accident (FHA) Fission Product Inventories for Use in Prairie Island Alternate Source Term Assessment.

18. Calculation GEN-PI-048, Non-LOCA Gap Release Factions for Use in Prairie Island Alternate Source Term Assessments.

19. Calculation GEN-PI-049, Prairie Island Control Room Atmospheric Dispersion Factors (χ/Q).

20. Calculation GEN-PI-050, Shielding Calculation for AST.

21. Calculation GEN-PI-051, Fuel Handling Accident Dose Analysis.

22. ORNL/TM-11018, Standard – and Extended – Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code, ORNL Chemical Technology Division, December.

23. ANSI/ANS-5.4-1982, American National Standard Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel, November.

24. Fauske and Associates Inc. (FAI), GAP Computer Code.

25. NMC Letter to the NRC dated December 9, 2003, "Response to GL 2003-01, Control Room Habitability."

EXHIBIT I

DRAFT

ADMINISTRATIVE CONTAINMENT CLOSURE CONTROLS

DURING FUEL MOVEMENT

FOR

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Letter PI-04-001

SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO CONTAINMENT TECHNICAL SPECIFICATIONS

The following requirements shall be maintained to ensure defense in depth. Containment Closure Controls are in effect whenever containment penetrations are open with movement of irradiated fuel assemblies in progress. The definition of an open containment penetration is a penetration that provides direct access from the containment atmosphere to the outside environment with no automatic closure available. Containment penetrations will be closed expeditiously $\leq$ 1 hour.

1.0 The equipment necessary to implement containment closure shall be appropriately staged prior to maintaining any containment penetration open including airlock doors and the containment equipment hatch.

2.0 Hoses and cables running through any open penetration, airlock, or equipment hatch shall be configured to facilitate rapid removal (e.g. quick disconnects, isolation valves) in the event that containment closure is required. Any cables or hoses to be disconnected should not be supplying services that support personnel safety (e.g., breathing air).

The designated personnel shall be trained, as required, on:

2.1 De-energizing/isolating the line prior to disconnecting.

2.2 Where to perform the de-energizing or isolation function.

2.3 Directions for disconnecting the line.

2.4 The location of any tools required for disconnection.

3.0 The containment airlocks may be open provided the following conditions exist:

3.1 One door in each airlock or an appropriate temporary door is capable of being closed.

3.2 The airlock door opening is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/airlock doors or devices to keep the door open/supported do not violate this provision.

3.3 Trained personnel are designated each shift with the responsibility for expeditious closure of at least one door on each airlock or closure of an appropriate temporary door following containment evacuation.

3.4 The airlocks are procedurally controlled by procedure.

4.0 The containment equipment hatch may be open provided the following conditions exist:

4.1 The containment equipment hatch is capable of being closed or a temporary closure method is available and can be implemented.

4.2 The equipment hatch is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/ equipment hatch or devices to keep the hatch open/flange supported do not violate this provision.

4.3 Necessary tools to install the equipment hatch flange and tighten at least four equipment hatch closure bolts are staged in the area or other methods to close the equipment hatch opening (i.e., restrict air flow out of the containment), such as an air curtain, are fabricated and staged at the work area along with the necessary installation tools.

4.4 Sufficient number of personnel are designated each shift with the responsibility for expeditious closure (≤ 1 hour) of the containment equipment hatch opening following containment evacuation.

5.0 Other containment penetrations may be open provided the following conditions exist:

5.1 One valve in each open containment penetration is capable of being closed, or

5.2 Other methods to close the open penetrations (i.e., restrict air flow out of the containment), such as a closure cover, shall be fabricated and staged along with the necessary installation tools.

5.3 Personnel are designated each shift with the responsibility for expeditious closure of open penetrations(s) following a fuel handling accident inside containment.

6.0 If containment closure would be hampered by an outage activity, compensatory actions will be developed.

7.0 The Containment Inservice Purge and Shield Building Ventilation Systems, with associated radiation release monitoring, will be available for the release path, whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open.

If for any reason this ventilation requirement cannot be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished, the equipment hatch closed, or a temporary cover is placed over the equipment hatch opening.

8.0 Actions by personnel following a fuel handling accident shall comply with plant requirements.

EXHIBIT H

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Letter PI-04-001
SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENT APPLIED TO CONTAINMENT TECHNICAL SPECIFICATIONS

FIGURES 1, 2 AND 3

NON-PROPRIETARY VERSION

AUX. BUILDING ELEV.735'-0"

FIGURE 1

NON-PROPRIETARY INFORMATION

AUX. BUILDING ELEV.755'-0"

FIGURE 2

NON-PROPRIETARY INFORMATION

...\PRI\Steve Thomas\AUX755.DGN 01/05/2004 10:41:13 AM

ROOF PLAN

FIGURE 3

NON-PROPRIETARY INFORMATION

...\CAD\PRI\Steve Thomas\ROOF.DGN 01/05/2004 11:18:05 AM