Xcel Energy®

AUG 2 1 2014

L-PI-14-072
10 CFR 50.90

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
Renewed License Nos. DPR-42 and DPR-60

License Amendment Request (LAR) to Revise the Licensing Basis Analysis for a Waste Gas Decay Tank Rupture

Pursuant to 10 CFR 50.90, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), hereby requests an amendment to revise the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, licensing basis analysis for a waste gas decay tank rupture. NSPM evaluated the changes proposed in this LAR in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

The enclosure to this letter, "Evaluation of the Proposed Changes", contains the licensee's evaluation of the proposed changes.

NSPM requests approval of this LAR within one calendar year of the submittal date. Upon NRC approval, NSPM requests 90 days to implement the associated changes. In accordance with 10 CFR 50.91, NSPM is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Mr. Dale Vincent, P.E., at 651-267-1736.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

1717 Wakonade Drive East • Welch, Minnesota 55089-9642
Telephone: 651.388.1121

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

AUG 2 1 2014

Kevin Davison
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosures (1)

cc: Administrator, Region III, USNRC
Project Manager, PINGP, USNRC
Resident Inspector, PINGP, USNRC
State of Minnesota

ENCLOSURE

Evaluation of the Proposed Changes

License Amendment Request (LAR) to Revise the Licensing Basis Analysis for a Waste Gas Decay Tank Rupture

---

ATTACHMENTS:

1. Updated Safety Analysis Report (USAR) page markup
2. Calculation 12400604-UR(B)-001, Revision 0A, "Waste Gas Tank Rupture Dose Consequences"

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, respectively.

Pursuant to 10 CFR 50.90, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), hereby requests an amendment to revise licensing basis analysis for the PINGP, Units 1 and 2, waste gas decay tank (WGDT) rupture.

2. DETAILED DESCRIPTION

2.1 Proposed Changes

This LAR proposes changes to the PINGP licensing basis and does not include any material changes to the Facility Operating License, Technical Specifications (TS) or TS Bases or any physical changes to the PINGP. Upon approval, the licensing basis changes proposed in this LAR will be made to the Updated Safety Analysis Report (USAR). A brief description of the proposed changes is provided below. Draft wording changes to the USAR are provided in Attachment 1.

USAR Section 14.5.3, "Accidental Release-Waste Gas": The discussion of the WGDT rupture analysis and dose consequences will be revised to document NRC approval of this licensing basis change through this requested license amendment.

2.2 Background

During implementation of the license amendments to increase PINGP power through measurement uncertainty recapture (ML102030573), NSPM determined that the WGDT rupture analysis required revision to provide missing analytical basis for dose consequences, incorporate the impact of changes to the fuel type and revise fuel cycle lengths. Calculation 12400604-UR(B)-001, Revision 0, was issued September 2, 2010 to address these changes and 10 CFR 50.59 Evaluation 1074, Revision 0 determined the revised calculation could be incorporated into the plant licensing basis without prior NRC review and approval.

During implementation of the PINGP renewed facility license (ML11147A141), NSPM determined that calculation 12400604-UR(B)-001 required revision to include 60 years of plant operation. Calculation 12400604-UR(B)-001, Minor Revision 0A, was issued September 5, 2013 to address this change and 10 CFR 50.59 Evaluation 1102, Revision 0 determined the revised calculation could be incorporated into the plant licensing basis without prior NRC review and approval.

A later NSPM internal review of 10 CFR 50.59 Evaluation 1074, Revision 0 and Evaluation 1102, Revision 0 determined that these evaluations incorrectly concluded that the revised calculation could be incorporated into the licensing basis without prior NRC review and approval. This issue was documented in the NSPM corrective action program (CAP) 01417573.

To resolve CAP01417573, 10 CFR 50.59 Evaluation 1102, Revision 1 was performed which concluded that NRC review and approval is required to incorporate calculation 12400604-UR(B)-001, Minor Revision 0A, into the plant licensing basis. This LAR provides calculation 12400604-UR(B)-001, Minor Revision 0A, for NRC review and approval.

With the licensing basis changes proposed in this LAR, the plant will continue to operate safely and the health and welfare of the public is protected.

3. TECHNICAL EVALUATION

PINGP is a two unit plant located on the right bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned and operated by Northern States Power Company, a Minnesota corporation (NSPM). Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with the licensee's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. PINGP was not licensed to NUREG-0800, "Standard Review Plan (SRP)", and was not part of the NRC Systematic Evaluation Program (SEP).

Waste Gas Decay Tank Description

The waste gas decay tanks contain gases vented from the reactor coolant system, the volume control tank, and the liquid holdup tanks. Two independent process loops are provided to accumulate and store radioactive gases. The first system is designed to strip fission gases from the reactor coolant while the second accumulates all other potentially radioactive gases. The fission gases stripped from the reactor coolant will represent the more significant portion of the radioactive source. Nonvolatile fission product concentrations are greatly reduced as the cooled reactor coolant system liquid is passed through purification demineralizers.

Current Licensing Basis

The activity in a gas decay tank is taken to be the maximum amount that could accumulate over the plant lifetime (40 years) from operation with one percent of the rated core thermal power being generated by rods with clad defects. For all isotopes except Kr-85, the postulated amount of activity is taken to be one reactor coolant system equilibrium cycle inventory. The Kr-85 inventory represents the activity at the end of a 40 year plant life.

To define the maximum doses, the release is assumed to result from gross failure of any process system storage tank, represented in the analysis by a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

The following tabulation summarizes the whole body and thyroid doses at the exclusion distance, consistent with a receptor on the plume centerline.

Current licensing basis

	Thyroid Dose		**Whole Body Dose**	
	EAB[1]	**LPZ**[2]	**EAB**	**LPZ**
Gas Decay Tank Rupture	negligible	N/A	1.5 rem	N/A
10 CFR 100 Guidelines	300 rem	300 rem	25 rem	25 rem

1. Exclusion Area Boundary
2. Low Population Zone

Note that the original plant licensing basis did not report values for the LPZ. These reported values demonstrated that a rupture in the waste gas system would present no undue hazard to public health and safety.

Proposed Licensing Basis Change

This license amendment proposes to incorporate the updated waste gas system tank rupture analysis in Calculation 12400604-UR(B)-001, Minor Revision 0A, (Attachment 2) into the PINGP licensing basis.

In the updated analysis, the activity in a gas decay tank is taken to be the maximum amount that could accumulate over the plant lifetime (60 years) from operation with one percent of the rated core thermal power being generated by rods with clad defects. For all isotopes except Kr-85, the postulated amount of activity is taken to be one reactor coolant system equilibrium cycle inventory. The Kr-85 inventory represents the activity at the end of a 60 year plant life.

As in the original licensing basis analysis, to define the maximum doses, the release is assumed to result from gross failure of any process system storage tank, represented in the analysis by a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

The following tabulation summarizes the updated analysis whole body and thyroid doses at the exclusion distances, consistent with a receptor on the plume centerline.

Calculation 12400604-UR(B)-001, Minor Revision 0A, results

	Thyroid Dose		**Whole Body Dose**	
	EAB	**LPZ**	**EAB**	**LPZ**
Gas Decay Tank Rupture	N/A	N/A	4.32 rem	1.18 rem
10 CFR 100 Guidelines	300 rem	300 rem	25 rem	25 rem

Although the original analysis did not report LPZ values, the original AEC safety evaluation (SE) did report LPZ values. To be consistent with the AEC SE and current dose analysis requirements, LPZ values were calculated and will be included in the plant licensing basis. These analysis results continue to demonstrate that a rupture in the waste gas system would present no undue hazard to public health and safety, and the calculated doses are well within the regulatory guidelines.

Even though the calculated doses in the proposed licensing basis calculations are well within the regulatory guidelines, this change requires NRC review and approval under the NSPM 10 CFR 50.59 program. The NSPM 10 CFR 50.59 program is based on NRC Regulatory Guide (RG) 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments" which endorsed Nuclear Energy Institute topical report 96-07 (NEI 96-07), "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, without exception.

The NEI 96-07 guidance provides a method for determining if increased dose consequences are more than minimal. Section 4.3.3 of NEI 96-07, Revision 1, establishes the current SRP guideline values as a basis for minimal increases for all facilities, not just those that were specifically licensed against those guidelines. Although PINGP was not licensed to the SRP and associated Branch Technical Positions (BTPs), these guidelines do apply for purposes of performing 10 CFR 50.59 evaluations per NEI 96-07, Section 4.3.3.

Per NUREG-0800, BTP 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure", "...the dose criterion in every case should not exceed 25 mSv (2.5 rem) at the exclusion area boundary given that such systems are fortified to withstand the effects of a hydrogen explosion and earthquakes. However, for systems not designed to withstand explosions and earthquakes, the criterion is 1 mSv (0.1 rem) at the exclusion area boundary."

Since the PINGP waste gas system design was not demonstrated to withstand explosions, the applicable BTP dose criterion is 0.1 rem at the exclusion area boundary. Since the dose consequence increase in the proposed licensing basis calculation is greater than 0.1 rem, this change is more than a minimal increase. Although PINGP was designed and licensed prior to issuance of NUREG-0800 and the concomitant BTPs, the guidance in BTP 11-5 is applied through the NSPM 10 CFR 50.59 program as a criterion for determining if a proposed change is more than a minimal increase and thus this proposed licensing basis waste gas tank rupture analysis is a more than minimal increase which requires NRC review and approval.

Technical Basis for Licensing Basis Change

The methods, inputs and assumptions used in Calculation 12400604-UR(B)-001, Minor Revision 0A, are summarized below. Specific details are provided in the calculation which is included with this LAR as Attachment 2 to this enclosure.

A rupture or failure of the waste gas system is highly unlikely since the components of the system are not subjected to high pressures or stresses, are a Class I design, and are designed to American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Class C standards. However, a rupture of a gas decay tank is analyzed to define the limit of the hazard that could result from any malfunction in the radioactive waste disposal system.

The activity in a gas decay tank is taken to be the maximum amount that could accumulate over the plant lifetime from operation with one percent of the rated core thermal power being generated by rods with clad defects. For all isotopes except Kr-85, this postulated amount of activity is taken to be one Reactor Coolant System equilibrium cycle inventory. This value is particularly conservative because some of this activity would normally remain in the coolant, some would have been dispersed earlier through the stack via equipment leakage, and the shorter-lived isotopes will have decayed substantially. The Kr-85 inventory represents the activity at the end of the 60 year plant life. To define the maximum doses, the release is assumed to result from gross failure of any process system storage tank, represented by a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

In calculating off-site plume center-line exposure, the activity was assumed to be discharged to the atmosphere at ground level. Dispersion coefficients based on the on-site meteorology program are used. The dispersion characteristics are discussed in the PINGP USAR Appendix H. Curves corrected for building wake effects by the volumetric source method, are presented on Figure 8 of USAR Appendix H.

The following parameters were used in the dose assessment:

a. 0-8 hour EAB dispersion factor value (X/Q) of 6.49E-4 sec/m^3
b. 0-8 hour LPZ X/Q of 1.77E-04 sec/m^3

The purpose of this calculation and its previous revisions was to incorporate changes at PINGP since the original calculation was performed. This calculation has been updated to include the impacts of fuel design and fuel cycle changes, and also account for an additional twenty years of plant operations due to plant life extension. The methodology used for this new waste gas tank rupture analysis is the same as that used in the original plant analysis.

In an LAR dated October 27, 2009 (ML093160583), NSPM proposed to adopt the alternative source term (AST) methodology allowed by 10 CFR 50.67. However, NSPM chose not to apply the AST methodology to the waste gas tank rupture analysis and thus the dose consequence acceptance criteria for this accident continues to be 10 CFR Part 100.

As shown in the table on Page 5, the calculated doses are well within the regulatory guidelines in 10 CFR Part 100; they are significantly less than the acceptance criteria in the original plant Safety Evaluation, dated September 28, 1972; and, these analysis results continue to demonstrate that a rupture in the waste gas system would present no undue hazard to public health and safety.

Conclusions

The Prairie Island Nuclear Generating Plant WGDT licensing basis rupture analysis is revised to incorporate plant changes including different fuel design and plant life extension to 60 years. The revised analysis results demonstrate that a WGDT rupture would result in doses that are well within the applicable regulatory guidelines and that the waste gas system design prevents release of undue amounts of radioactivity. However, the revised analysis requires NRC review and approval because the dose increase is more than minimal under the NSPM 10 CFR 50.59 program criteria. Operation of the Prairie Island Nuclear Generating Plant with the proposed licensing basis calculation will continue to protect the health and safety of the public.

4. REGULATORY SAFETY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 100.11 Determination of exclusion area, low population zone, and population center distance.

> (a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

The Prairie Island Nuclear Generating Plant (PINGP) was originally licensed to the post-accident guidelines in 10 CFR 100.11. License amendments 206 and 193, for Units 1 and 2 respectively, (LA-206/193) (ML112521289), applied the alternative source term methodology to most Prairie Island Nuclear Generating Plant accident analyses. As stated in the license amendment request dated October 27, 2009 (ML093160583) which initiated LA-206/193, the analysis of releases from the waste gas system was not included in the scope of the proposed license amendments. Since PINGP was licensed prior to issuance of NUREG-0800 and is not committed to the Standard Review Plan, 10 CFR 100.11 provides the applicable regulatory guidelines for the waste gas decay tank (WGDT) rupture analysis. The updated WGDT rupture analysis demonstrates that the doses are well within the applicable 10 CFR 100.11 regulatory guidelines and the health and safety of the public is not affected.

General Design Criteria

The construction of the Prairie Island Nuclear Generating Plant was significantly complete prior to issuance of 10 CFR 50, Appendix A, General Design Criteria. The Prairie Island Nuclear Generating Plant was designed and constructed to comply with the Atomic Energy Commission General Design Criteria as proposed on July 10, 1967, (AEC GDC) as described in the plant Updated Safety Analysis Report. AEC GDC proposed Criterion 69 provides design guidance for protection against radioactivity release from waste storage.

Criterion 69 – Protection Against Radioactivity Release from Spent Fuel and Waste Storage

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

AEC GDC Criterion 69 is met for the waste storage at the Prairie Island Nuclear Generating Plant through the design to prevent the release of undue radioactivity to the public. The original plant analysis of the WGDT rupture demonstrated that releases were a small fraction of the applicable regulatory guidelines in 10 CFR 100.11. The analysis proposed in this license amendment request, which was updated to include the current fuel types, fuel cycle length and operation to 60 years, demonstrate that doses from a WGDT rupture are well within the applicable regulatory guidelines and that the waste gas system design prevents release of undue amounts of radioactivity. With the changes proposed in this license amendment request, AEC GDC 69 continues to be met.

4.2 Precedent

The NRC has reviewed and approved license amendments for the Prairie Island Nuclear Generating Plant which revise licensing basis dose consequence analyses. A recent example are license amendments 191 and 180 (LA-191/180), for Units 1 and 2 respectively, (ML091490611) which revised the accident analyses for loss of coolant and main steam line break accidents and their concomitant dose consequences. This proposed license amendment differs from LA-191/180 in that it does not involve any associated plant Technical Specification changes.

4.3 Significant Hazards Consideration

Northern States Power Company, a Minnesota corporation (NSPM) 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. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   This license amendment request proposes to revise the licensing basis waste gas decay tank rupture analysis. The proposed analysis was updated to include the current fuel type, current fuel cycle lengths and plant operation to sixty years.

   The proposed waste gas decay tank rupture analysis changes are not accident initiators, and therefore the proposed changes do not involve an increase in the probability of an accident.

   The original waste gas decay tank rupture analysis demonstrated that the doses were a small fraction of the regulatory guidelines and that the waste gas system design prevents release of undue amounts of radioactivity. The revised waste gas decay tank rupture analysis demonstrates that the doses are well within the regulatory guidelines and that the waste gas system design continues to prevent release of undue amounts of radioactivity, and thus the proposed changes do not involve a significant increase in the consequences of an accident.

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

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

Response: No

This license amendment request proposes to revise the licensing basis waste gas decay tank rupture analysis. The proposed analysis was updated to include the current fuel type, current fuel cycle lengths and plant operation to sixty years.

The proposed waste gas decay tank rupture analysis includes plant changes that have previously been evaluated. This analysis applies the same methodology as the previous analysis. The proposed revision to the waste gas decay tank rupture analysis does not change any system operations or maintenance activities. The changes do not involve physical alteration of the plant; that is, no new or different type of equipment will be installed. These changes do not create new failure modes or mechanisms which are not identifiable during testing and no new accident precursors are generated.

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

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

This license amendment request proposes to revise the licensing basis waste gas decay tank rupture analysis. The proposed analysis was updated to include the current fuel type, current fuel cycle lengths and plant operation to sixty years.

This revised analysis applies the same methodology as the original waste gas decay tank rupture analysis. The original waste gas decay tank rupture analysis demonstrated that the doses were a small fraction of the regulatory guidelines and that the waste gas system design prevents release of undue amounts of radioactivity. The revised waste gas decay tank rupture analysis demonstrates that the doses are well within the regulatory guidelines and that the waste gas system design continues to prevent release of undue amounts of radioactivity.

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

Based on the above, the Northern States Power Company, a Minnesota corporation (NSPM) concludes that the proposed amendment does not involve a 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.

4.4 **Conclusions**

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.

5. ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, 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 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

None

ENCLOSURE, ATTACHMENT 1

Updated Safety Analysis Report Pages (Markup)

Page 14.5-11
Page 14.5-12
Page 14.11-8

3 pages follow

(Reference A)

The activity in a gas decay tank is taken to be the maximum amount that could accumulate over the plant lifetime from operation with one percent of the rated core thermal power being generated by rods with clad defects. For all isotopes except Kr 85, this postulated amount of activity is taken to be one Reactor Coolant System equilibrium cycle inventory as given in Appendix D, Table D.7-1. This value is particularly conservative because some of this activity would normally remain in the coolant, some would have been dispersed earlier through the stack via equipment leakage, and the shorter-lived isotopes will have decayed substantially. The Kr 85 inventory given in Appendix D, Table D.7-1, represents the activity at the end of the 60 year plant life. 01390545

To define the maximum doses, the release is assumed to result from gross failure of any process system storage tank, here represented by a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

Gas decay tank rupture maximum doses are provided along those for volume control tank rupture, below. 01390545

14.5.3.2 Volume Control Tank Rupture

The volume control tank contains fission gases and low concentrations of halogens which are normally a source of waste gas activity vented to a gas decay tank. The iodine concentrations and volatility are quite low at the temperature, pH and pressure of the fluid in the volume control tank. The same assumptions detailed in the preceding subsection apply to this tank. As the volume control tank and associated piping are not subjected to any high pressures or stresses, failure is very unlikely. However, a rupture of the volume control tank is analyzed to define the maximum exposure that could result from such an occurrence.

Rupture of the volume control tank is assumed to release all the contained noble gases and 1% of the halogen inventory of the tank plus that amount contained in the 40 gpm flow from the demineralizers, which would continue for up to fifteen minutes before isolation would occur. The 1% halogen release is a very conservative estimate of the decontamination factor expected for these conditions.

Based on 1% fuel defects, the activities available for release are 7700 Ci of Xe^{133} dose equivalent noble gases and .022 Ci of I^{131} dose equivalent halogens.

The WGDT rupture analysis results have been approved by the NRC in License Amendment ___ and ___, for Units 1 and 2, respectively. (Reference B)

Method of Analysis

In calculating off-site plume center-line exposure it is assumed that the activity is discharged to the atmosphere at ground level and is dispersed as a Gaussian plume downwind taking into account building wake dilution.

Dispersion coefficients based on the on-site meteorology program are used. A wind velocity of 0.89 meters per second is assumed to remain in one direction for the duration of the accident under Pasquill F conditions. The dispersion characteristics are discussed in Appendix H. Curves corrected for building wake effects by the volumetric source method, are presented on Figure 8 of Appendix H.

The following parameters have been used in the dose assessment:

a. A 0-8 hour EAB X/Q value of 6.49×10^{-4} sec/m^3

b. A 0-8 hour LPZ X/Q value of 1.77E-04 sec/m^3

c. Breathing rate equal to 3.47×10^{-4} m^3/sec

d. An I^{131} equivalent dose conversion factor equal to 1.48×10^6 rem/curie

e. A Kr^{85} dose equivalent conversion factor equal to 6.20×10^{-2} rem-m^3/curie-sec

f. A Xe^{133} dose equivalent conversion factor equal to 3.57×10^{-2} rem-m^3/curie-sec

(Ref. A)

The following tabulation summarizes the whole body and thyroid doses at the exclusion distance, consistent with a receptor on the plume centerline.

	Thyroid Dose		Whole Body Dose	
	EAB	LPZ	EAB	LPZ
Gas Decay Tank Rupture	N/A	N/A	4.32 rem	1.18 rem
Volume Control Tank Rupture	7.3E-03 rem	1.7E-03	0.18 rem	0.05 rem
10CFR100 Limits	300 rem	300 rem	25 rem	25 rem

01390545

It is concluded that a rupture in the waste gas system or in the volume control tank would present no undue hazard to public health and safety.

107. S.L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, USNRC, December 2007.

108. Calculation GEN-PI-080, "Prairie Island Atmospheric Dispersion Factors (X/Qs) - AST Additional Releases," Revision 1.

109. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000.

110. Thomas J. Wengert (USNRC) to Mr. James E. Lynch (NSPMN), "Subject: Prairie Island Nuclear Generating Plants, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternative Source Term Methodology (TAC Nos. ME2609 and ME2610)," January 22, 2013. [ML112521289] 01406407

111. LTR-LIS-13-274, "Prairie Island Units 1 and 2, 10 CFR 50.46 Summary Sheets for the Evaluation to Support the Unit 2 Installation of AREVA Model 56/19 Replacement Steam Generators (RSGs)," June 28, 2013. 01386642

A. Calculation # 12400604-UR(B)-001, Waste Gas Tank Rupture Dose Consequences, Revision 0A.

B. NRC SER for License Amendment _ _ _ / _ _ _ , dated _ _ /_ _ /_ _.

Use next available numbers.

ENCLOSURE, ATTACHMENT 2

Calculation 12400604-UR(B)-001
Revision 0A

Waste Gas Tank Rupture Dose Consequences

28 pages follow

Xcel Energy — Calculation Signature Sheet

Document Information	
NSPM Calculation (Doc) No: 12400604-UR(B)-001	Revision: 0
Title: Waste Gas Tank Rupture Dose Consequences	
Facility: ☐ MT ☒ PI	Unit: ☒ 1 ☒ 2
Safety Class: ☒ SR ☐ Aug Q ☐ Non SR	
Special Codes: ☐ Safeguards ☐ Proprietary	
Type: **Calc** Sub-Type:	

NOTE:	Print **and** sign name in signature blocks, as required.

Major Revisions	☒ N/A
EC Number:	☐ Vendor Calc
Vendor Name or Code:	Vendor Doc No:
Description of Revision:	
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record	☒
Prepared by: (sign) / (print)	Date:
Reviewed by:(sign) / (print)	Date:
Type of Review: ☐ Design Verification ☐ Tech Review ☐ Suitability Review	
Method Used (For DV Only): ☐ Review ☐ Alternate Calc ☐ Test	
Approved by: (sign) / (print)	Date:

Minor Revisions	☐ N/A
EC No: 22469	☒ Vendor Calc:
Minor Rev. No: A	
Description of Change: Update WGDT Rupture analysis based on 60 yr plant life	
Pages Affected:	
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record	☒
Prepared by: (sign) / (print) Vendor	Date:
Reviewed by: (sign) [signature] / (print) Jason Loeffler	Date: 7/19/13
Type of Review: ☒ Design Verification ☐ Tech Review ☒ Suitability Review	
Method Used (For DV Only): ☐ Review ☐ Alternate Calc ☐ Test	
Approved by: (sign) [signature] / (print) 7/22/2013	Date:

Tony Perez

Record Retention: Retain this form with the associated calculation for the life of the plant.

Xcel Energy

Calculation Signature Sheet

NOTE:	This reference table is used for data entry into the PassPort Controlled Documents Module reference tables (C012 Panel). It may also be used as the reference section of the calculation. The input documents, output documents and other references should all be listed here. Add additional lines as needed by using the "TAB" key and filling in the appropriate information in each column.

Reference Documents (PassPort C012 Panel from C020)

#	Controlled* Doc? +	Type	Document Name	Document Number	Doc Rev	Ref Type** INPUT	OUTPUT
1	x	LIC	Safety Analysis	USAR Section 14	32	X	X
2	x	LIC	Radioactive Source Bases	USAR Section – APP D	32	X	X
3							
4							
5							
6							
7							
8							
9							
10							
11							
12							
13							
14							
15							
16							
17							

* Controlled Doc marked with an "X" means the reference can be entered on the C012 panel in black. Unmarked lines will be yellow. If marked with an "X", also list the Doc Type, e.g., CALC, DRAW, VTM, PROC, etc.

Record Retention: Retain this form with the associated calculation for the life of the plant.

Xcel Energy

Calculation Signature Sheet

** Mark with an "X" if the calculation provides inputs and/or outputs or both. If not, leave blank. (Corresponds to PassPort "Ref Type" codes: **Inputs / Both =** "ICALC", **Outputs** = "OCALC", **Other / Unknown** = blank)

Other PassPort Data

Associated System (PassPort C011, first three columns) **OR** **Equipment References** (PassPort C025, all five columns):

Facility	Unit	System	Equipment Type	Equipment Number
PI	0	WG	SYS	SYS 0WG

Superseded Calculations (PassPort C019):

Facility	Calc Document Number	Title

Description Codes - Optional (PassPort C018):

Code	Description (optional)	Code	Description (optional)

Notes (Nts) - Optional (PassPort X293 from C020):

Topic Notes	Text
☐ Calc Introduction	☒ Copy directly from the calculation Intro Paragraph **or** ☐ See write-up below
☐ (Specify)	

Record Retention: Retain this form with the associated calculation for the life of the plant.

Xcel Energy

Calculation Signature Sheet

Monticello Specific Information

☐ YES ☐ N/A Topic Code(s) (See MT Form 3805): _______
☐ YES ☐ N/A Structural Code(s) (See MT Form 3805):_______

Does the Calculation:

☐ YES ☐ No Require Fire Protection Review? (Using MT Form 3765, "Fire Protection Program Checklist", determine if a Fire Protection Review is required.) If YES, document the engineering review in the EC. If NO, then attach completed MT Form 3765 to the associated EC.

☐ YES ☐ No Affect piping or supports? (If Yes, Attach MT Form 3544.)

☐ YES ☐ No Affect IST Program Valve or Pump Reference Values, and/or Acceptance Criteria? (If Yes, inform IST Coordinator and provide copy of calculation.)

Record Retention: Retain this form with the associated calculation for the life of the plant.

XcelEnergy	**Design Review Comment Form**

Sheet 1 of 2

DOCUMENT NUMBER/ TITLE: 179876.51.2002, Waste Gas Decay Tank Rupture Dose Analysis

REVISION: 0 DATE: 7/16/2013

ITEM #	REVIEWER'S COMMENTS	PREPARER'S RESOLUTION	REVIEWER'S DISPOSITION
1	Prairie Island has already applied for a license extension and is about to enter the period of extended operation. Recommend revising the first sentence of the "Purpose and Scope" section to reflect this.	Agreed. Revised the first sentence of Section 1.0 to clarify the PINGP license application status.	Acceptable
2	Design Input 4.1.3 and 4.1.4: Recommend referencing USAR Section 14.5.3 in lieu of 14.5.3.2. The subsection 2 under 14.5.3 is discussing the VCT rupture.	Agreed. Revised referenced section to be 14.5.3 instead of 14.5.3.2.	Acceptable
3	Section 5.1, Paragraph 2: Typo – "…the Kr-85 activity give in Reference 2…" should read "given in Reference 2"	Agreed. Revised text.	Acceptable
4	Eq-1 used to predict the decay of Kr-85m into Kr-85 is different than the equation used in 12400606-UR(B)-001. The exponential decay component is ignored such that the equation simplifies to: $A_2(t) = A_1 * \frac{\lambda_2}{\lambda_{1-\lambda_2}}$ The equation in the current Analysis of Record is given as this:	No Resolution Required	N/A

	$A_2(t) = A_1 * \frac{\lambda_2}{\lambda_1}$ Note: this error has no impact on the results as the decay constant for Kr-85m is so much larger than that of Kr-85. It appears that the analysis of record simplified the equation and didn't describe what was done. No resolution required this comment is only for documentation.		
5	Recommend condensing USAR Appendix D References down into one. Still list the specific table when the reference is cited in the design inputs/ analysis section but the individual tables don't need to be included in the references section. The reference section should just list PINGP USAR, Appendix D, Radioactive Source Bases, Revision 32.	Agreed. Reference 2 is now PINGP USAR, Appendix D, Radioactive Source Bases, Revision 32. References 3, 4, and 7 have been removed and the other references have been renumbered as a result.	Acceptable
6	Section 6.1.2, Editorial Change – Remove "quite" from second sentence (i.e., "…Kr-85 doesn't reach equilibrium..").	Agreed. Text revised to remove "quite."	Acceptable

Reviewer: Jason Loeffler Date: 7/18/13	Preparer: Jason D. Olson Date: 07/17/13

Xcel Energy	External Design Document Suitability Review Checklist

External Design Document Being Reviewed:Calculation

Title: **Waste Gas Decay Tank Rupture Dose Analysis**

Number: 179876.51.2002 Rev: 0 Date: 7-18-13

This design document was received from:

Organization Name: Black & Veatch PO or DIA Reference: 46458

The purpose of the suitability review is to ensure that a calculation, analysis or other design document provided by an External Design Organization complies with the conditions of the purchase order and/or Design Interface Agreement (DIA) and is appropriate for its intended use. The suitability review does not serve as an independent verification. Independent verification of the design document supplied by the External Design Organization should be evident in the document, if required.

The reviewer should use the criteria below as a guide to assess the overall quality, completeness and usefulness of the design document. The reviewer is not required to check calculations in detail.

REVIEW

		Reviewed	N/A
1	Design inputs correspond to those that were transmitted to the External Design Organization.	☒	☐
2	Assumptions are described and reasonable.	☒	☐
3	Applicable codes, standards and regulations are identified and met.	☒	☐
4	Applicable construction and operating experience is considered.	☐	☒
5	Applicable structure(s), system(s), and component(s) are listed.	☒	☐
6	Formulae and equations are documented. Unusual symbols are defined.	☒	☐
7	Acceptance criteria are identified, adequate and satisfied.	☒	☐
8	Results are reasonable compared to inputs.	☒	☐
9	Source documents are referenced.	☒	☐
10	The document is appropriate for its intended use.	☒	☐
11	The document complies with the terms of the Purchase Order and/or DIA.	☒	☐
12	Inputs, assumptions, outputs, etc. which could affect plant operation are enforced by adequate procedural controls. List any affected procedures in the Comments block below.	☒	☐
13	Plant impact has been identified and either implemented or controlled. List all identified plant impacts, their associated tracking A/Rs and a description in the PLANT IMPACT(S) table.	☒	☐
14	Design and Operational Margin have been considered and documented.	☒	☐

Xcel Energy	External Design Document Suitability Review Checklist

Comments: This calculation calculates whole body dose rates of 4.32 and 1.18 REM at the EAB and LPZ, respectively, based on the 60 year plant life. This falls below the 10CFR 100 Limits of 25 REM. Additionally, the increase in predicted dose rate is less than 10% of the difference between the current calculated dose value and the regulatory limit as discussed in NEI 96-07, Revision 1. The current dose rates predicted in USAR chapter 14 are 3.3 REM and 0.88 REM at the EAB and LPZ. The 1.02 REM increase in dose at the EAB is less than 10% of the existing margin to the regulatory limit (2.17 REM). The 0.3 REM increase in dose at the LPZ is less than 10% of the existing margin to the regulatory limit (2.41 REM).

Completed by: ______________________ Date: 07/19/2013

Xcel Energy	External Design Document Suitability Review Checklist

PLANT IMPACT(S)

Calculation Number(1): _12400604-UR(B)-001 Minor Rev 0A (B&V Calc 179876.51.2002)_

Item No.	AR Tracking Number(2)	PLANT IMPACT REQUIRED ACTION DESCRIPTION
1	1343799-02	Update USAR Table D.7-1 to account for Kr-85 activity at end of life
2	1343799-02	Update USAR Section 14.5.3 to address updated dose analysis
3		
4		
5		
6		
7		
8		
9		
10		
11		
12		
13		
14		
15		

Table Notes:

(1) If an identified plant impact is a result of a vendor calculation, provide the calculation number and list and describe any follow on actions resulting from the calculation AND any documents that enforce calculation inputs, assumptions or conclusions.

(2) Initiate an AR to track open items and plant impacts (e.g., procedure revisions, validation of assumptions, database updates, etc.), if any.

BLACK & VEATCH Building a world of difference.	CALCULATION COVER SHEET

Client/Dept. Xcel Energy

Record No. 179876.51.2002

Project Xcel Energy AST Implementation

Project No. 179876

Calculation No. 179876.51.2002

File No. 51.2000

Calculation Title Waste Gas Decay Tank Rupture Dose Analysis

Rev. 0

Calculation Type: ☐ Preliminary ☒ Final

Seismic Classification: ☒ I ☐ II ☐ NS ☐ Other

Safety-Related: ☒ Yes ☐ No ☐ Other

OBJECTIVE

The objective of this calculation is to determine the whole body dose at the EAB and LPZ due to a rupture and release of the radioactive noble gases contained in the gas decay tank.

DEFERRED DESIGN VERIFICATION ITEMS

No.	Deferred Design Verification Description	DVR No.*	Date
	None		

COMPUTER PROGRAM CALCULATIONS

Computer Program Title ______ Version ______

REVIEW AND APPROVAL

Rev.	Prepared By		DVR No.*	Approved By	
	Print and Sign	Date		Print and Sign	Date
0	Jason D. Olson Jason D Olson	07/17/2013	DVR-0003	Steve Thomas SDK	7/17/2013

*Indicate Design Verification Review (DVR) number.

This calculation supersedes Calculation Number None

This calculation is superseded by Calculation Number None

BLACK & VEATCH Building a world of difference.	CALCULATION COVER SHEET

Client/Dept. Xcel Energy — Record No. 179876.51.2002

Project Xcel Energy AST Implementation — Project No. 179876

Calculation No. 179876.51.2002 — File No. 51.2000

Calculation Title Waste Gas Decay Tank Rupture Dose Analysis — Rev. 0

Table of Contents:

BLACK & VEATCH Building a world of difference.	CALCULATION COVER SHEET

Client/Dept.	Xcel Energy	Record No.	179876.51.2002
Project	Xcel Energy AST Implementation	Project No.	179876
Calculation No.	179876.51.2002	File No.	51.2000
Calculation Title	Waste Gas Decay Tank Rupture Dose Analysis	Rev.	0

1.0 Purpose and Scope:

Prairie Island Nuclear Generating Plant (PINGP) is in the process of implementing an extension to their operating license, which extends the 40 year plant life to 60 years. The PINGP Updated Safety Analysis Report (USAR) Section 14 (Ref. 1) currently considers a waste gas decay tank (WGDT) rupture accident with the release of the radionuclide inventory at the end of 40 years of plant operation. In order to support the extension of the PINGP operating license, the WGDT rupture analysis must be updated to consider 60 years of plant operation. The purpose of this calculation is to evaluate the offsite dose due to a postulated WGDT rupture and release of the radionuclide inventory at the end of 60 years of plant operation. The scope of this calculation does not include the volume control tank rupture accident.

2.0 References:

1. PINGP USAR, Section 14, Revision 33P.

2. PINGP USAR, Appendix D, Radioactive Source Bases, Revision 32.

3. "Radioactive Decay Data Tables," David C. Kocher, DOE/TIC-11026, 1981.

4. "Waste Gas Tank Rupture Dose Consequences," Xcel Energy, Calculation No. 12400604-UR(B)-001, Revision 0.

5. "Fundamentals of Nuclear Science and Engineering," J. Kenneth Shultis and Richard E. Faw, 2002, Marcel Dekker, Inc.

6. Code of Federal Regulations, 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

7. Email from Jason Loeffler, PINGP, Subject: Reactor Coolant System Equilibrium Cycle, dated July 16, 2013 (see Attachment 7.3).

3.0 Summary and Conclusions:

The whole body doses (gamma + beta) at the exclusion area boundary (EAB) and low population zone (LPZ) boundary due to a postulated WGDT accident following 60 years of plant operation are presented below in Table 3.1. Refer to Section 6.0 for further details.

Table 3.1 – Whole Body Doses due to WGDT Accident

Location	Whole Body Dose (rem)	Acceptance Criteria
EAB	4.32	25 rem
LPZ	1.18	25 rem

All radionuclide activities in the Waste Gas Decay Tank remain unchanged from those given in USAR Table D.7-1 (Ref. 2), except for Kr-85, which is 8.15×10^4 Ci at the end of the 60 year plant life. The EAB and LPZ doses are within the acceptance criteria derived from 10CFR100.11 (Ref. 6).

4.0 Design

4.1 Design Inputs

4.1.1 Waste Gas Decay Tank radionuclide inventory as presented in USAR, Table D.7-1 (Ref. 2).

Isotope	Total Activity (Curies)
Kr-85	5.43×10^4
Kr-85m	1.98×10^2
Kr-87	1.28×10^2
Kr-88	3.60×10^2
Xe-133m	4.54×10^2
Xe-133	3.21×10^4
Xe-135	1.02×10^3
Xe-135m	5.89×10^1
Xe-138	7.60×10^1

4.1.2 Effective Decay Energies for Noble Gases as presented in USAR, Table D.8-1 (Ref. 2)

Isotope	Gamma (MeV)	Beta (MeV)
Kr-85	0.0022	0.25
Kr-85m	0.158	0.229
Kr-87	0.614	1.24
Kr-88	1.95	0.36
Xe-133m	0.041	0.0
Xe-133	0.045	0.1
Xe-135	0.248	0.303
Xe-135m	0.431	0.0
Xe-138	1.13	0.615

4.1.3 The 0-8 hour exclusion area boundary (EAB) atmospheric dispersion factor (χ/Q) is 6.49×10^{-4} sec/m^3 per USAR Section 14.5.3 (Ref. 1).

4.1.4 The 0-8 hour low population zone (LPZ) atmospheric dispersion factor (χ/Q) is 1.77×10^{-4} sec/m^3 per USAR Section 14.5.3 (Ref. 1).

4.1.5 The half-lives of the noble gas radionuclides in the WGDT are given in Reference 3:

Isotope	Half-Life
Kr-85	10.72 yr
Kr-85m	4.48 hr
Kr-87	1.27 hr
Kr-88	2.84 hr
Xe-133m	2.19 d
Xe-133	5.245 d
Xe-135	9.11 hr
Xe-135m	15.36 min
Xe-138	14.13 min

4.1.6 The branching fraction of Kr-85m to Kr-85 by isomeric transition (IT) is 0.211 (Ref. 3).

4.2 Design Margins

Client Margin: No margin has been specified by the client.
Safety Margin: No safety margin has been specified.
Design Margin: No design margin has been specified.
Operation Margin: No operation margin has been specified.
Other margin: The method used to calculate the Kr-85 activity in the WGDT after 60 years of plant operation is inherently conservative because radioactive decay of Kr-85 was not included.

4.3 Acceptance Criteria

4.3.1 The calculated doses at the EAB and LPZ are below the regulatory dose limit of 25 rem to the whole body, as given in 10CFR100.11 (Ref. 6) and USAR Section 14 (Ref. 1).

4.4 Assumptions

4.4.1 This calculation adopts any underlying assumptions used in the preparation of the WGDT radionuclide inventories and accident parameters given in the referenced portions of the PINGP USAR (Refs. 1-2).

4.4.2 The noble gas radionuclides in the WGDT are fission products that have accumulated over the plant lifetime from operation with one percent of the rated core thermal power being generated by rods with clad defects (Ref. 1). Therefore it is assumed that the average annual buildup rate for all the noble gas radionuclides in the WGDT will remain constant over the 60 year plant life.

5.0 Analysis

USAR Section 14 (Ref. 1) currently provides the results of a WGDT rupture analysis (Ref. 4) that considered an accumulation of noble gas radionuclides over a 40 year plant life. The accident considered in Reference 1 assumes that the entire inventory of a single WGDT is instantaneously released to the atmosphere. The whole body doses at the EAB and LPZ were then calculated based on this release and the atmospheric dispersion factors for the Prairie Island site. The objective of this calculation is to perform an analysis similar to Reference 4, but extend the analysis to reflect a 60 year operating life.

5.1 WGDT Radionuclide Inventory

The radionuclide inventory of the WGDT consists of noble gases (krypton and xenon isotopes). Per USAR Section 14.5.3.1 (Ref. 1), the activity in a gas decay tank is taken to be the maximum amount that could accumulate over the plant lifetime from operation with one percent (1%) of the rated core thermal power being generated by rods with clad defects. For all isotopes except Kr-85, this postulated amount of activity is taken to be one Reactor Coolant System equilibrium cycle inventory as given in USAR Appendix D, Table D.7-1 (Ref. 2). As stated in Section 14.5.3.1 of Reference 1, this value is conservative because some of this activity would normally remain in the coolant, some would have been dispersed earlier through the stack via equipment leakage, and the shorter-lived isotopes would have decayed substantially.

The Kr-85 activity given in Reference 2, as determined in Reference 4, represents the activity at the end of the 40 year plant life. For this calculation, the Kr-85 activity will be calculated for the end of the 60 year plant life. The remainder of the shorter-lived isotopes will still remain at their equilibrium activities through the additional 20 years of operation.

In order to calculate the Kr-85 activity at 60 years, the buildup rate of Kr-85 in the WGDT will be calculated from the existing 40-year data and then integrated over 60 years instead of 40 years. It is assumed that the average annual buildup rate for all the noble gas radionuclides in the WGDT will remain constant over the 60 year plant life. One must also consider that Kr-85m may decay into Kr-85 by isomeric transition and contribute to the total activity of Kr-85 in the WGDT. The following equation provided by Reference 5, Equation 5.66, is used to calculate the activity of a daughter nuclide (Kr-85) from a known activity of a parent nuclide (Kr-85m) at time t:

$$A_2(t) = A_1(0)\frac{\lambda_2}{\lambda_1-\lambda_2}\left[e^{-\lambda_2 t}\right] \qquad \text{Eq. 1}$$

Where:

$A_2(t)$ = activity of daughter radionuclide at time t, Curies
$A_1(0)$ = activity of parent radionuclide at time zero, Curies
λ_1 = decay constant of parent radionuclide, yr^{-1}
λ_2 = decay constant of daughter radionuclide, yr^{-1}
t = time elapsed from time zero at which daughter activity is determined, yr^{-1}

The governing equation for cases where radionuclides are being continuously introduced into a system or vessel is derived from Equation 5.53 of Reference 5:

$$A(t) = \frac{P}{\lambda}\left(1 - e^{-\lambda t}\right) \qquad \text{Eq. 2}$$

Where:

$A(t)$ = activity of radionuclide at time t, Curies
P = production or rate at which activity is added to the WGDT, Curies per unit time
λ = decay constant of radionuclide, $(\text{unit of time})^{-1}$

t = time elapsed from time zero at which activity is determined, yr^{-1}

The radioactive decay constant is a function of the radionuclide's half-life, $T_{1/2}$, and is given in Equation 5.36 of Reference 5:

$$\lambda = \frac{0.693}{T_{1/2}} \qquad \text{Eq. 3}$$

5.2 EAB and LPZ Doses

Once the total activities are calculated for each noble gas radionuclide, the whole body dose at the site boundaries of interest can be calculated based on the methodology given in USAR Section D.8.3 (Ref. 2). The whole body gamma dose and the skin beta dose delivered to a dose receptor is obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane (i.e., an "infinite semispherical cloud"). The general equation for calculating the whole body dose due to gamma radiation is given in Reference 2 as follows:

$$D_\gamma = 0.246 \left(\frac{X}{Q}\right) \sum_i Q_i E_{i\gamma} \qquad \text{Eq. 4}$$

Where:

D_γ = whole body dose due to gamma radiation, rem
X/Q = site dispersion factor, sec/m^3
Q_i = total activity of isotope i released, Curies
$E_{i\gamma}$ = effective gamma decay energy from isotope i, MeV/disintegration

Similarly, Reference 2 also provides the general equation for calculating the whole body skin beta dose as follows:

$$D_\beta = 0.246 \left(\frac{X}{Q}\right) \sum_i Q_i E_{i\beta} \qquad \text{Eq. 5}$$

Where:

D_β = whole body skin dose due to beta radiation, rem
X/Q = site dispersion factor, sec/m^3
Q_i = total activity of isotope i released, Curies
$E_{i\beta}$ = effective beta decay energy from isotope i, MeV/disintegration

6.0 Results

6.1 WGDT Radionuclide Inventory at 60 years

6.1.1 Shorter-Lived Radionuclides

The WGDT inventory given in Section 4.1.1 consists of the following radionuclides and activities, along with their respective half-lives from Section 4.1.5:

Table 6.1-1 WGDT Activity

Isotope	Total Activity (Curies)	Half-Life
Kr-85	5.43×10^{4}	10.72 yr
Kr-85m	1.98×10^{2}	4.48 hr
Kr-87	1.28×10^{2}	1.27 hr
Kr-88	3.60×10^{2}	2.84 hr
Xe-133m	4.54×10^{2}	2.19 d
Xe-133	3.21×10^{4}	5.245 d
Xe-135	1.02×10^{3}	9.11 hr
Xe-135m	5.89×10^{1}	15.36 min
Xe-138	7.60×10^{1}	14.13 min

As described in Reference 1, these activities, with the exception of Kr-85, are based on one Reactor Coolant System equilibrium cycle inventory. Although Reference 1 does not specify the rate at which these radionuclides are introduced into the WGDT, it can be demonstrated that given a certain production rate (Ci/s), the shorter-lived noble gas radionuclides will quickly reach an equilibrium activity (saturation activity) in the WGDT. As an example, consider Kr-88 using Eq. 2 from Section 5.1:

$$A(t) = \frac{P}{\lambda}\left(1 - e^{-\lambda t}\right)$$

From this equation it can be seen by inspection that as time approaches infinity, the maximum activity is the production rate divided by the decay constant. The production rate is then given by:

$$P = \lambda A$$

For Kr-88, the production rate is determined as follows, substituting Eq. 3 from Section 5.1 into the formula for λ :

$$P = \frac{0.693}{2.84\ hr} x\ 360\ Ci$$

$$P = 87.845\ Ci/hr$$

Now substituting this production rate back into Eq. 2 from Section 5.1 and choosing various buildup times demonstrates how quickly the saturation activity is reached for short-lived radionuclides. Below is a sample calculation at t = 1 hr. Attachment 7.1 contains the activity calculations for other buildup times and the results are tabulated below in Table 6.1-2.

$$A(1\,hr) = \frac{87.845\ Ci/hr}{\frac{0.693}{2.84\ hr}}\left(1 - e^{-\frac{0.693}{2.84\ hr}x1hr}\right)$$

$$A(1\,hr) = 77.95\ Ci$$

Table 6.1-2 Kr-88 Activity in WGDT

Time (hr)	Activity (Ci)
0	0
1	77.95
2	139.02
4	224.35
6	276.74
8	308.89
10	328.63
12	340.74
14	348.18
16	352.74
18	355.55
24	358.97
36	359.94
48	360.00
720	360.00

As can be seen from the data in Table 6.1-2, the Kr-88 activity quickly approaches the saturation activity for the given constant production rate. The equilibrium point, where the decay rate of Kr-88 is equivalent to the production rate, is reached in two days. Thus the total activity remains unchanged after this point, unless the production rate changes. This result can be generalized for all of the shorter-lived isotopes in the WGDT and the equilibrium activities in Table 6.1-1 may be used directly in the calculation of the EAB and LPZ doses for a 60 year operating plant.

6.1.2 Krypton 85

Due to its longer half-life, Kr-85 must be dealt with in a different manner than the shorter-lived isotopes. Unlike the shorter-lived isotopes, Kr-85 doesn't reach equilibrium over the 40 year plant life, thus necessitating the calculation of the total activity after 60 years of accumulation in the WGDT. The previous WGDT rupture analysis (Ref. 4), on which the current USAR activities and doses are based, conservatively assumed that Kr-85 would undergo no radioactive decay over the 40 year plant life. This methodology is retained for this calculation.

In addition to accumulation of Kr-85 from the Reactor Coolant System, decay from Kr-85m into Kr-85 by IT must also be considered. The governing equation is given in Section 5.1, Eq. 1:

$$A_2(t) = A_1(0)\frac{\lambda_2}{\lambda_1 - \lambda_2}\left[e^{-\lambda_2 t}\right]$$

Ignoring the exponential decay of the daughter (Kr-85) yields the following simplified form of the equation:

$$A_{Kr85} = A_{Kr85m}(0)\frac{\lambda_{Kr85}}{\lambda_{Kr85m} - \lambda_{Kr85}}$$

Kr-85m may decay by beta emission or IT. From Section 4.1.6, the branching fraction for IT to Kr-85 is 0.211. Thus, the contribution of Kr-85m to the total activity of Kr-85 is calculated as follows:

$$A_{85}(60yr) = 0.211\ x\ 198\ Ci\frac{\dfrac{0.693}{10.72\ yr\ x\ 365.25\frac{d}{yr}x24\frac{hr}{d}}}{\dfrac{0.693}{4.48\ hr} - \dfrac{0.693}{10.72\ yr\ x\ 365.25\frac{d}{yr}x24\frac{hr}{d}}}$$

$$A_{85}(60yr) = 41.778\ Ci\ \frac{7.375\ x\ 10^{-6}hr^{-1}}{0.1547\ hr^{-1} - 7.375\ x\ 10^{-6}hr^{-1}}$$

$$A_{85}(60yr) = 41.778\ Ci\ x\ 4.77\ x10^{-5}$$

$$A_{85}(60yr) = 2.0\ x10^{-3}\ Ci$$

Thus, the buildup of Kr-85 from the decay of Kr-85m is insignificant.

The activity of Kr-85 at the end of the 40 year plant life is given as 5.43×10^4 Ci in Table D.7 of Reference 2. Because decay is not considered, the production rate of Kr-85 is simply calculated by dividing the total activity by the duration of 40 years:

$$P = \frac{5.43\ x\ 10^4\ Ci}{40\ yr}$$

$$P = 1357.5\frac{Ci}{yr}$$

PINGP currently operates using an 18-month fuel cycle, so using a 12-month fuel cycle will result in a conservative accumulation of Kr-85 in the WGDT (Ref. 7). The activity of Kr-85 at the end of 60 years is then given by multiplying the production rate (P) by 60 12-month fuel cycles:

$$P = 1357.5 \frac{Ci}{yr} \, x \, 60 \, yr$$

$$P = 8.145 \, x \, 10^4 \, Ci$$

6.2 EAB and LPZ Doses

The EAB and LPZ doses are calculated using Eqs. 4 and 5 from Section 5.1:

$$D_\gamma = 0.246 \left(\frac{X}{Q}\right) \sum_i Q_i E_{i\gamma}$$

$$D_\beta = 0.246 \left(\frac{X}{Q}\right) \sum_i Q_i E_{i\beta}$$

Where the X/Q values are given in Sections 4.1.3 and 4.1.4, the nuclide activities are given above in Table 6.1-1, except for Kr-85 which was determined in Section 6.1.2 to be 8.145 x 10^4 Ci, and the effective beta and gamma decay energies are given in Section 4.1.2.

6.2.1 Whole Body Doses

Using the formulae and inputs described above, the whole body dose at the EAB and LPZ are calculated in Attachment 7.2 and the results are presented below.

Table 6.2-1 Whole Body Dose at the EAB

Isotope	Activity (Ci)	Effective Gamma Decay Energy (MeV/dis)	Effective Beta Decay Energy (MeV/dis)	Whole Body Gamma Dose (rem)	Whole Body Skin Beta Dose (rem)
Kr-85	8.15E+04	0.0022	0.25	2.86E-02	3.25E+00
Kr-85m	1.98E+02	0.158	0.229	4.99E-03	7.24E-03
Kr-87	1.28E+02	0.614	1.24	1.25E-02	2.53E-02
Kr-88	3.60E+02	1.95	0.36	1.12E-01	2.07E-02
Xe-133m	4.54E+02	0.041	0.0	2.97E-03	0.00E+00
Xe-133	3.21E+04	0.045	0.1	2.31E-01	5.12E-01
Xe-135	1.02E+03	0.248	0.303	4.04E-02	4.93E-02
Xe-135m	5.89E+01	0.431	0.0	4.05E-03	0.00E+00
Xe-138	7.60E+01	1.13	0.615	1.37E-02	7.46E-03
EAB X/Q = 6.49E-04 sec/m^3			Total Dose:	4.50E-01	3.87E+00
		Total Gamma + Beta Dose:		4.32E+00	

Table 6.2-2 Whole Body Dose at the LPZ

Isotope	Activity (Ci)	Effective Gamma Decay Energy (MeV/dis)	Effective Beta Decay Energy (MeV/dis)	Whole Body Gamma Dose (rem)	Whole Body Skin Beta Dose (rem)
Kr-85	8.15E+04	0.0022	0.25	7.80E-03	8.87E-01
Kr-85m	1.98E+02	0.158	0.229	1.36E-03	1.97E-03
Kr-87	1.28E+02	0.614	1.24	3.42E-03	6.91E-03
Kr-88	3.60E+02	1.95	0.36	3.06E-02	5.64E-03
Xe-133m	4.54E+02	0.041	0.0	8.10E-04	0.00E+00
Xe-133	3.21E+04	0.045	0.1	6.29E-02	1.40E-01
Xe-135	1.02E+03	0.248	0.303	1.10E-02	1.35E-02
Xe-135m	5.89E+01	0.431	0.0	1.11E-03	0.00E+00
Xe-138	7.60E+01	1.13	0.615	3.74E-03	2.04E-03
LPZ X/Q = 1.77E-04 sec/m^3			Total Dose:	1.23E-01	1.06E+00
		Total Gamma + Beta Dose:		1.18E+00	

Client/Dept.	Xcel Energy	Record No.	179876.51.2002
Project	Xcel Energy AST Implementation	Project No.	179876
Calculation No.	179876.51.2002	File No.	51.2000
Calculation Title	Waste Gas Decay Tank Rupture Dose Analysis	Rev.	0

7.0 Attachments

7.1 Excel spreadsheet – Kr-88 Activities (2 pages)
7.2 Excel spreadsheet – EAB and LPZ Doses (2 pages)
7.3 Email – PINGP, Jason Loeffler to B&V, Reactor Coolant System Equilibrium Cycle (2 pages)

time (hr)	Kr-88 Activity (Ci)
0	0
1	77.95
2	139.02
3	186.87
4	224.35
5	253.72
6	276.74
7	294.76
8	308.89
9	319.96
10	328.63
11	335.42
12	340.74
13	344.91
14	348.18
15	350.74
16	352.74
17	354.31
18	355.55
19	356.51
20	357.27
21	357.86
22	358.32
23	358.68
24	358.97
25	359.19
26	359.37
27	359.50
28	359.61
29	359.70
30	359.76
31	359.81
32	359.85
33	359.89
34	359.91
35	359.93
36	359.94
37	359.96
38	359.97
39	359.97
40	359.98
41	359.98
42	359.99
43	359.99
44	359.99

45	359.99
46	359.99
47	360.00
48	360.00
49	360.00
50	360.00
720	360.00

Isotope	Activity (Ci)	Eff. Gamma Decay Energy (MeV/dis)	Eff. Beta Decay Energy (MeV/dis)	WBD Gamma (rem)	WBSD Beta (rem)
Kr-85	8.15E+04	0.0022	0.25	2.86E-02	3.25E+00
Kr-85m	1.98E+02	0.158	0.229	4.99E-03	7.24E-03
Kr-87	1.28E+02	0.614	1.24	1.25E-02	2.53E-02
Kr-88	3.60E+02	1.95	0.36	1.12E-01	2.07E-02
Xe-133m	4.54E+02	0.041	0	2.97E-03	0.00E+00
Xe-133	3.21E+04	0.045	0.1	2.31E-01	5.12E-01
Xe-135	1.02E+03	0.248	0.303	4.04E-02	4.93E-02
Xe-135m	5.89E+01	0.431	0	4.05E-03	0.00E+00
Xe-138	7.60E+01	1.13	0.615	1.37E-02	7.46E-03
			Total Dose:	4.50E-01	3.87E+00
	Total Gamma + Beta Dose:			4.32E+00	
EAB X/Q =	6.49E-04 sec/m^3				

Isotope	Activity (Ci)	Eff. Gamma Decay Energy (MeV/dis)	Eff. Beta Decay Energy (MeV/dis)	WBD Gamma (rem)	WBSD Beta (rem)
Kr-85	8.15E+04	0.0022	0.25	7.80E-03	8.87E-01
Kr-85m	1.98E+02	0.158	0.229	1.36E-03	1.97E-03
Kr-87	1.28E+02	0.614	1.24	3.42E-03	6.91E-03
Kr-88	3.60E+02	1.95	0.36	3.06E-02	5.64E-03
Xe-133m	4.54E+02	0.041	0	8.10E-04	0.00E+00
Xe-133	3.21E+04	0.045	0.1	6.29E-02	1.40E-01
Xe-135	1.02E+03	0.248	0.303	1.10E-02	1.35E-02
Xe-135m	5.89E+01	0.431	0	1.11E-03	0.00E+00
Xe-138	7.60E+01	1.13	0.615	3.74E-03	2.04E-03
			Total Dose:	1.23E-01	1.06E+00
			Total Gamma + Beta Dose:	1.18E+00	
LPZ X/Q =	1.77E-04 sec/m^3				

Olson, Jason D.

Subject: FW: Initial Contact - PINGP Waste Gas Decay Tank Rupture Analysis

From: Loeffler, Jason W. [mailto:Jason.Loeffler@xenuclear.com]
Sent: Tuesday, July 16, 2013 3:48 PM
To: Olson, Jason D.
Subject: RE: Initial Contact - PINGP Waste Gas Decay Tank Rupture Analysis

Jason:

I was able to discuss this question with the people involved in the 12400604-UR(B)-001 Rev 0 calculation and found out the following. The 1.11 uCi/cc is the maximum concentration of Kr-85 which would be produced during each fuel cycle. The equilibrium cycle for Prairie Island is 280 days after which the activity levels will no longer continue to increase in the RCS. To determine the maximum amount of activity released to the WGDT you would multiply the number of fuel cycles. Originally the plant was licensed to 12 month fuel cycles. However, at some point we switched to 18 month fuel cycles. To determine a conservative activity level the integration through 40 years was based on a 12 month fuel cycle rather than the 18 month fuel cycle.

Let me know if you have any additional questions.

Thanks,

Jason Loeffler
P: 651.388.1121 ext. 4694

From: Olson, Jason D. [mailto:OlsonJD@bv.com]
Sent: Monday, July 15, 2013 2:08 PM
To: Loeffler, Jason W.
Cc: Dendinger, Travis; Towner, David L.; Perez, Antonio J.
Subject: RE: Initial Contact - PINGP Waste Gas Decay Tank Rupture Analysis

Jason,

In the current WGDT dose analysis calculation (12400604-UR(B)-001, Rev. 0), Section 6.1, the Kr-85 activity in the WGDT after 40 years is calculated to be 7.68E+03 Ci by integrating the Kr-85 concentration over 40 years without decay. This activity was multiplied by the scaling factor 7.07 to get the current WGDT activity of 5.43E+04 Ci, which is what I'm using in my calculation.

My verifier has asked me how we know that the 1.11 uCi/cc is an annualized concentration. USAR 14.5.3.1 states that the postulated amount of activity (in the WGDT) is taken to be one Reactor Coolant System equilibrium cycle inventory, except for Kr-85, which is the activity at the end of 40 years. Is one RCS equilibrium cycle equivalent to one year of operation?

For Kr-85, in 12400604-UR(B)-001, p.20/45 (see attachment), the specific activity from one equilibrium cycle was simply multiplied by 40 years to arrive at a 40 year activity in the WGDT. Can you help us in locating some additional information such as the definition of "equilibrium cycle" so we can be sure the approach we are using is correct? We went back to Appendix D of the USAR, specifically Table D.4-1, where the RCS fission product activities are given, and the table points to Reference 8, which is: Westinghouse Calculation Note CN-REA-07-36, "Prairie Island Core Activity Inventory and Reactor Coolant Activity," Revision 0. We could not locate this in the PI document server. This document may well clarify what an equilibrium cycle is or what duration it covers.

Thanks for your help in this matter.

Jason D. Olson, P.E.* | Nuclear Engineer, Energy
Black & Veatch Corporation | 11401 Lamar Ave, Overland Park, KS 66211
+1 913-458-7666 P | olsonjd@BV.com
*Licensed in KS