PrairieIslandISFSIPEm Resource

From: Nguyen, John-Chau
Sent: Monday, September 08, 2014 10:13 AM
To: PrairieIslandISFSIHearing Resource
Subject: pi isfsi 2nd rai responses
Attachments: ML14234A463 - Prairie Island Independent Spent Fuel Storage Installation, Supplement to License Renewal Application - Response to Second Request for Additional Information (TAC L24592)..pdf

Thank you.

John-Chau Nguyen
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Hearing Identifier: Prairie_Island_ISFSI_Public
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~~ENCLOSURE 4 CONTAINS PROPRIETARY INFORMATION~~
~~WITHHOLD FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390~~

Xcel Energy®

Prairie Island Nuclear Generating Plant
1717 Wakonade Drive East
Welch, MN 55089

July 31, 2014

L-PI-14-069
10 CFR 72.42

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Director, Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

Prairie Island Independent Spent Fuel Storage Installation
Docket No. 72-10
Materials License No. SNM-2506

Supplement to License Renewal Application – Response to Second Request for Additional Information (TAC No. L24592)

References:

1. Letter from M.A. Schimmel (NSPM) to Document Control Desk (NRC), "Prairie Island Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application," L-PI-11-074, dated October 20, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11304A068).

2. Letter from J.E. Lynch (NSPM) to Document Control Desk (NRC), "Supplement to License Renewal Application – Response to Request for Additional Information (TAC No. L24592)," L-PI-13-073, dated July 26, 2013 (ADAMS Accession No. ML13210A272).

3. Letter from P. Longmire, PhD (NRC) to J.E. Lynch (NSPM), "Second Request for Additional Information for Review of the License Renewal Application for the Prairie Island Independent Spent Fuel Storage Installation – SNM-2506 (TAC No. L24592)," dated May 27, 2014 (ADAMS Accession No. ML14147A527).

Pursuant to 10 CFR 72.42, Northern States Power Company, a Minnesota corporation doing business as Xcel Energy (hereafter "NSPM"), submitted in Reference 1 an application to renew the site-specific license for the Prairie Island Independent Spent Fuel Storage Installation (ISFSI), Special Nuclear Materials (SNM) license SNM-2506. This License Renewal Application (LRA) requested that SNM-2506 be extended an additional 40 years beyond the current expiration date of October 31, 2013.

NMSS26

In Reference 2, NSPM provided responses to a Request for Additional Information (RAI) dated January 31, 2013 (ADAMS Accession No. ML13035A083) and a revised RAI dated June 5, 2013 (ADAMS Accession No. ML13163A291). In Reference 3, the Nuclear Regulatory Commission (NRC) provided a second RAI to support the staff's technical review of the LRA. A public meeting was held with the NRC technical staff on June 16, 2014 to obtain clarification of the staff's questions and discuss NSPM's proposed responses. This letter provides NSPM's response to the RAIs in Reference 3.

Enclosure 1 to this letter contains the oath or affirmation statement required pursuant to 10 CFR 72.16.

Enclosure 2 to this letter provides NSPM's responses to the subject RAI. NSPM submits this supplemental information in accordance with 10 CFR 72.42.

Enclosure 3 to this letter contains an affidavit from AREVA Inc., that has been executed to support withholding Enclosure 4 from public disclosure. This affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). NSPM respectfully requests that the information in Enclosure 4 which is proprietary to AREVA Inc., be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4), as authorized by 10 CFR 9.17(a)4.

Correspondence with respect to the proprietary aspects of AREVA Inc. information or the supporting AREVA Inc. affidavit in Enclosure 3 should be addressed to Jeffery Isakson, Vice President, AREVA Inc., 7135 Minstrel Way, Columbia, MD 21045.

Enclosure 4 to this letter contains AREVA Inc. calculation TN40HT-0513, Revision 1, "Evaluation of the potential for the buildup of flammable gases in the TN-40 dry cask system." This calculation supports statements made in Enclosure 2, in responses to RAIs 10 and 11. This calculation contains proprietary information from AREVA, Inc.

Enclosure 5 is a non-proprietary version of the calculation provided in Enclosure 4.

NSPM has determined that the supplemental information provided in this letter meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(11) and, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared. In addition, pursuant to 10 CFR 51.60(b)(2), no environmental report need be prepared. The proposed changes do not require any changes to the Prairie Island ISFSI Environmental Report.

If there are any questions or if additional information is needed, please contact Gene Eckholt, Projects Licensing Manager, at 651-267-1742.

Summary of Commitments

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

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

Enclosures (5)

cc: Administrator, Region III, USNRC (letter only)
SFST Project Manager, Prairie Island, USNRC (8 copies with Enclosure)
NRR Project Manager, Prairie Island, USNRC (letter only)
Resident Inspector, Prairie Island, USNRC (letter only)
State of Minnesota (letter only)

ENCLOSURE 1

Oath or Affirmation Pursuant to 10 CFR 72.16

1 Page Follows

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND INDEPENDENT SPENT FUEL STORAGE INSTALLATION
DOCKET NO. 72-10

LICENSE RENEWAL APPLICATION FOR
MATERIALS LICENSE No. SNM-2506

SUPPLEMENT TO LICENSE RENEWAL APPLICATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The Northern States Power Company, a Minnesota corporation, d/b/a Xcel Energy (hereafter "NSPM") provides additional information that supports the application to renew the Materials License for the Prairie Island Independent Spent Fuel Storage Installation.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY - MINNESOTA

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

State of Minnesota

County of Goodhue

On this 31 day of July, 2014 before me a notary public acting in said County, personally appeared Kevin Davison, 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 NSPM, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true.

[signature]

CATHERINE S MCEATHRON
Notary Public
Minnesota
My Comm. Expires
Jan 31, 2017

ENCLOSURE 2

Responses to Second Request for Additional Information

License Renewal Application
Prairie Island Independent Spent Fuel Storage Installation
SNM-2506

Introduction

This Enclosure provides supplemental information from the Northern States Power Company, a Minnesota corporation (hereafter "NSPM") doing business as Xcel Energy, in support of the License Renewal Application (LRA) for the Prairie Island Independent Spent Fuel Storage Installation (ISFSI). The Prairie Island ISFSI LRA was submitted October 20, 2011 (Reference 1, ADAMS Accession No. ML11304A068), and requested an extension of the site-specific license for an additional 40 years.

In a letter dated July 26, 2013 (Reference 2), NSPM responded to a request for additional information (RAI) from the Nuclear Regulatory Commission (NRC). After further review, the NRC provided a second RAI in Reference 3. A public meeting was held June 16, 2014 to clarify the staff's questions and to discuss NSPM's proposed responses.

This Enclosure provides NSPM's responses to each of the RAI questions provided in Reference 3. Each of the RAI questions is quoted in italics and each question is then followed by the NSPM response in normal font.

Reference documents are identified at the end of this Enclosure.

RAI Responses

RAI-1:
Provide a table, or other format, that identifies each instance in the safety analysis report (SAR) that refers to a limited storage system period. A justification of their categorization should be provided.

The RAI GI-1 response (ML13210A272) stated that areas of the SAR which refer to a limited storage system period (i.e., 25 years, 20 years, storage period, service life, etc.) are grouped into three categories. Only one example was provided in each category. A complete listing and description of all the instances and a justification of their categorization should be provided in order to confirm that the adequacy of the design over the renewal life span has been correctly addressed.

This information is required to evaluate compliance with 10 CFR 72.42(b).

NSPM Response:

NSPM included new and revised sections of the SAR in the LRA (Reference 1), Enclosure 3, Appendix C, "Safety Analysis Report Supplement and Changes." This appendix identified pertinent changes to the SAR to describe aging management activities needed to provide reasonable assurance that the intended functions of structures, systems, and components (SSCs) are maintained during the period of extended operations. These aging management activities include Time-Limited Aging Analyses (TLAAs) and the ISFSI Inspection and Monitoring Activities Program, which is the Aging Management Program (AMP) credited for the ISFSI.

In the response to RAI GI-1 (Reference 2), NSPM acknowledged that additional changes to the SAR would be needed to effectively implement a license renewal. These changes typically involve updating statements that refer to a limited storage system period, such as 25 years, 20 years, or more generic terms such as storage period and service life.

To clearly explain how each of these SAR statements will be updated for the renewed license, markups of the affected SAR pages are provided in Attachment A to this Enclosure. Attachment A also includes a table that summarizes each proposed change and provides a justification for the changes shown in the markups. Note that since actual SAR markups are provided and the changes are not merely categorized into the three groups discussed in the response to RAI GI-1, the justification discussions in Attachment A do not specifically address "categorization" as requested in the RAI.

The SAR markups in Attachment A are provided for information only and no approval is requested. NSPM plans to include these changes in the next scheduled update to the SAR after receipt of the renewed materials license for the Prairie Island ISFSI.

RAI-2:

Provide a revised Aging Management Program (AMP) addressing the following aging effects/mechanisms of the concrete pad, for both above-grade (accessible and inaccessible) and below-grade (underground inaccessible areas), as applicable. Otherwise, provide detailed justifications for why these aging effects/mechanisms do not require an AMP:

- *Cracking or loss of material (spalling, scaling) due to chemical attack (above-grade, below-grade);*
- *Cracking and loss of strength due to cement aggregate reactions (above-grade inaccessible, below-grade);*
- *Cracking, loss of material, and loss of bond due to corrosion of embedded steel (above grade, below-grade);*
- *Increase in porosity/permeability and loss of strength due to leaching of Ca(OH)2 (above-grade inaccessible); and*
- *Cracking due to settlement (above-grade, below-grade).*

Section A2.6 (NSPM, 2011), "Acceptance Criteria," states that the acceptance criteria for all visual inspections of the concrete pads will be consistent with, or more restrictive than, those contained in ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures." ACI 349.3R details degradation mechanisms applicable to the reinforced concrete pad, which were excluded from AMP results in NSPM (2011). A complete AMP should address these aging mechanisms, or a detailed justification is required for their exclusion. The justification should provide a site-specific technical basis (e.g., engineering analysis, operational experience data), which demonstrates that these aging mechanisms will not adversely affect the ability of the pad to perform its intended important-to-safety (ITS) function during the period of extended operation. If the applicant intends on relying on the degradation of accessible areas as a precursor for degradation in inaccessible areas (both above and below grade), the justification should demonstrate that such an approach will be sufficient to prevent a loss of ITS function. Note that per ACI 349.3R, testing activities may be used to quantify the environment to which the below-grade or inaccessible structure is exposed. These tests could include a program for analysis of soils and groundwater chemistry, as well as an evaluation of their propensity to cause concrete degradation or steel reinforcement corrosion. Similar guidance and acceptance criteria are given in ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

This information is needed to determine compliance with 10 CFR 72.42(a)(2), 122(b)(1) and 72.122(f).

NSPM Response:

As described in Section 3.1.2 of the ISFSI LRA (Reference 1), the ISFSI AMP was developed using the terminology, aging management tools, and reports provided by the Electric Power Research Institute (EPRI). This is consistent with the Aging Management Program for Prairie Island Nuclear Generating Plant (PINGP) and avoids confusion for engineers and other site personnel that would be created if different terminology and basis documents were used. This could be particularly confusing for the concrete pads which are fundamentally the same as concrete structures in the plant.

For concrete pads, as stated in the response to RAI 3-1 (Reference 2), NSPM utilized EPRI Report 1002950, "Aging Effects for Structures and Structural Components (Structural Tools)," Revision 1, August 2003 (Reference 4). The terminology used to describe aging effects and aging mechanisms in Table 3.4.1 of the LRA (Reference 1), "AMR Results for Concrete Pads," is consistent with the EPRI report. Section 5 of that report discusses the identification of aging effects of concrete structures and components. Table 5-2 of the EPRI report contains a summary of the potential aging effects of concrete structures and identifies criteria that must be met (or present) in order for the applicable aging effect/mechanism to occur.

For each aging effect/mechanism listed in the RAI, Table RAI 2-1 below lists the mechanism applicability criteria from Table 5-2 of EPRI Report 1002950. The far right column of the table provides the site specific justification why the subject aging mechanism/effect is, or is not, applicable to the Prairie Island (PI) ISFSI concrete pads.

It is noted that the terminology used in the RAI for some of the aging effects does not agree with the broader terminology used in the EPRI report. For example, the RAI lists "cracking due to chemical attack," while the EPRI report discusses this as part of the loss of material aging effect due to aggressive chemicals. This difference in terminology between the RAI aging effects/mechanisms and that used in the EPRI report is highlighted in the footnotes of Table RAI 2-1.

It is also noted that the RAI refers to degradation mechanisms identified in ACI 349.3R. NSPM did state in Section A2.6.2 of the LRA that the acceptance criteria for all visual inspections of the concrete pads are consistent with, or more restrictive than, those contained in ACI 349.3R; however, ACI 349.3R was not used to perform the aging management review (AMR). The AMR was developed using the EPRI tools and reports mentioned previously.

With regard to potential degradation in inaccessible areas, NSPM relies upon inspections/monitoring of accessible areas as a precursor for degradation in inaccessible areas (both under a cask and below grade). This is further explained in the response to RAI-5 for justification of the inspection frequency of these areas. In addition, NSPM is revising the AMP listed in Appendix A of the LRA to include ground water monitoring, as discussed in the response to RAI-4.

Table RAI 2-1

Excerpts from Table 5-2 of EPRI Report 1002950

RAI listed Aging Mechanism	RAI listed Aging Effect	Mechanism Applicability Criteria from Table 5-2 of EPRI Report 1002950	Applicability to the PI ISFSI Concrete Pads Site Specific Justification
Chemical attack	Cracking[(1)]	Plant site is located near heavy industry area (applicable for concrete above grade) And Acidic solutions with pH < 5.5 (applicable for concrete above and below grade) Or Chloride solutions > 500 ppm (applicable for concrete above and below grade) Or Sulfate solutions > 1500 ppm (applicable for concrete above and below grade)	The concrete pads are not located near any heavy industrial sites with exposure to aggressive chemical environments (acidic solutions pH <5.5, chloride solutions >500 ppm, or sulfate solutions > 1500 ppm), including groundwater chemistry (refer to the Response to RAI A-3 in Reference 2). Therefore, cracking due to chemical attack is not a potential aging effect that requires an Aging Management Program.

Table RAI 2-1

Excerpts from Table 5-2 of EPRI Report 1002950

RAI listed Aging Mechanism	RAI listed Aging Effect	Mechanism Applicability Criteria from Table 5-2 of EPRI Report 1002950	Applicability to the PI ISFSI Concrete Pads Site Specific Justification
Chemical attack	Loss of material	Plant site is located near heavy industry area (applicable for concrete above grade) And Acidic solutions with pH < 5.5 (applicable for concrete above and below grade) Or Chloride solutions > 500 ppm (applicable for concrete above and below grade) Or Sulfate solutions > 1500 ppm (applicable for concrete above and below grade)	The concrete pads are not located near any heavy industrial sites with exposure to aggressive chemical environments (acidic solutions pH<5.5, chloride solutions >500 ppm or sulfate solutions > 1500 ppm), including groundwater chemistry (refer to the Response to RAI A-3 in Reference 2). Therefore, loss of material due to chemical attack is not a potential aging effect that requires an Aging Management Program.
Cement aggregate reactions	Cracking	Improper or contaminated admixtures – salt-contaminated aggregates, seawater or deicing salt.	The concrete pads have the potential for exposure to deicing salt. Therefore, cracking due to cement aggregate reactions is a potential aging mechanism and is already included in the ISFSI Inspection and Monitoring Activities Program, i.e., an AMP. Refer to Tables 3.4-1 and A2.1-1 of the LRA.

Table RAI 2-1

Excerpts from Table 5-2 of EPRI Report 1002950

RAI listed Aging Mechanism	RAI listed Aging Effect	Mechanism Applicability Criteria from Table 5-2 of EPRI Report 1002950	Applicability to the PI ISFSI Concrete Pads Site Specific Justification
Cement aggregate reactions	Loss of strength(6)	EPRI Report 1002950 does not list loss of strength due to reaction with aggregate as an aging effect/mechanism. However, it is reasonable to apply the mechanism applicability criteria for the reaction with aggregate associated with the cracking effects to the loss of strength effect. Improper or contaminated admixtures – salt-contaminated aggregates, seawater or deicing salt.	The concrete pads have the potential for exposure to deicing salt. Therefore, loss of strength (as evidenced by cracking) due to cement aggregate reactions is a potential aging mechanism. Cracking due to cement aggregate reactions is already included in the ISFSI Inspection and Monitoring Activities Program, i.e., an AMP. Refer to Tables 3.4-1 and A2.1-1 of the LRA.
Corrosion of embedded steel	Cracking(2)	Concrete that is not of good quality, well consolidated, and properly cured Or Embedded steel is exposed to an aggressive environment.	The concrete pads are of good quality, well consolidated, and properly cured. The pads were designed and placed in accordance with ACI 318-89 or 349-90, as applicable. As discussed above the pads are not exposed to an aggressive environment. Therefore, cracking due to corrosion of embedded steel is not a potential aging effect that requires an Aging Management Program.

Table RAI 2-1

Excerpts from Table 5-2 of EPRI Report 1002950

RAI listed Aging Mechanism	RAI listed Aging Effect	Mechanism Applicability Criteria from Table 5-2 of EPRI Report 1002950	Applicability to the PI ISFSI Concrete Pads Site Specific Justification
Corrosion of embedded steel	Loss of material	Concrete that is not of good quality, well consolidated, and properly cured Or Embedded steel is exposed to an aggressive environment.	The concrete pads are of good quality, well consolidated, and properly cured. The pads were designed and placed in accordance with ACI 318-89 or 349-90, as applicable. As discussed above the pads are not exposed to an aggressive environment. Therefore, loss of material due to corrosion of embedded steel is not a potential aging effect that requires an Aging Management Program.
Corrosion of embedded steel	Loss of bond(3)	Concrete that is not of good quality, well consolidated, and properly cured Or Embedded steel is exposed to an aggressive environment.	The concrete pads are of good quality, well consolidated, and properly cured. The pads were designed and placed in accordance with ACI 318-89 or 349-90, as applicable. As discussed above the pads are not exposed to an aggressive environment. Therefore, loss of bond due to corrosion of embedded steel is not a potential aging effect that requires an Aging Management Program.

Table RAI 2-1

Excerpts from Table 5-2 of EPRI Report 1002950

RAI listed Aging Mechanism	RAI listed Aging Effect	Mechanism Applicability Criteria from Table 5-2 of EPRI Report 1002950	Applicability to the PI ISFSI Concrete Pads Site Specific Justification
Leaching $Ca(OH)_2$	Increase in porosity / permeability [4]	Concrete structures that are exposed to flowing liquid, ponding, or hydraulic pressure And Defects in the concrete such as cracks, voids, or low strength are necessary to permit movement of water through the concrete.	The concrete pads have the potential for exposure to flowing liquid, ponding, or hydraulic pressure and potential defects in the concrete such as cracks, voids, or low strength may be present to permit movement of water through the concrete. Therefore, change in material properties (including increase in porosity / permeability) due to leaching Ca(OH)2 is a potential aging effect and is already included in the ISFSI Inspection and Monitoring Activities Program, i.e., an AMP. Refer to Tables 3.4-1 and A2.1-1 of the LRA.
Leaching $Ca(OH)_2$	Loss of strength[5]	Concrete structures that are exposed to flowing liquid, ponding, or hydraulic pressure And Defects in the concrete such as cracks, voids, or low strength are necessary to permit movement of water through the concrete.	The concrete pads have the potential for exposure to flowing liquid, ponding, or hydraulic pressure and potential defects in the concrete such as cracks, voids, or low strength may be present to permit movement of water through the concrete. Therefore, change in material properties (including loss of strength) due to leaching $Ca(OH)_2$ is a potential aging effect and is already included in the ISFSI Inspection and Monitoring Activities Program, i.e., an AMP. Refer to Tables 3.4-1 and A2.1-1 of the LRA.

Table RAI 2-1

Excerpts from Table 5-2 of EPRI Report 1002950

RAI listed Aging Mechanism	RAI listed Aging Effect	Mechanism Applicability Criteria from Table 5-2 of EPRI Report 1002950	Applicability to the PI ISFSI Concrete Pads Site Specific Justification
Settlement	Cracking	Structures founded on soft soil And/Or Significant changes in underground water conditions over a long period of time (i.e., lowering of the ground water table).	The concrete pads have the potential to be founded on soft soil or the potential for significant changes in underground water conditions over a long period of time (i.e., lowering of the ground water table). Therefore, cracking due to settlement is a potential aging effect and is already included in the ISFSI Inspection and Monitoring Activities Program, i.e., an AMP. Refer to Tables 3.4-1 and A2.1-1 of the LRA.

(1) While EPRI Report 1002950 does not list cracking as a separate aging effect associated with the aggressive chemicals, it does discuss it in Section 5.3.1.4 as a part of the loss of material aging effect. Therefore, the applicability criterion listed in this table corresponds to loss of material due aggressive chemicals.

(2) While EPRI Report 1002950 does not list cracking as a separate aging effect associated with the corrosion of embedded steel, it does discuss it in Section 5.3.1.5 as a part of the loss of material aging effect. Therefore, the applicability criterion listed in this table corresponds to loss of material due to corrosion of embedded steel below.

(3) While EPRI Report 1002950 does not list loss of bond as a separate aging effect associated with the corrosion of embedded steel, it does discuss it in Section 5.3.1.5 as a part of the loss of material aging effect. Therefore, the applicability criteria listed in this table correspond to loss of material due to corrosion of embedded steel above.

(4) While EPRI Report 1002950 does not list increase in porosity / permeability as a separate aging effect associated with leaching $Ca(OH)_2$, it does discuss it in Section 5.3.3.1 as a part of the change in material properties aging effect. Therefore, the applicability criterion listed in this table corresponds to change in material properties due to leaching $Ca(OH)_2$.

(5) While EPRI Report 1002950 does not list loss of strength as a separate aging effect associated with leaching $Ca(OH)_2$, it does discuss it in Section 5.3.3.1 as a part of the change in material properties aging effect. Therefore, the applicability criterion listed in this table corresponds to change in material properties due to leaching $Ca(OH)_2$.

(6) EPRI Report 1002950 does not list loss of strength due to reactions with aggregate as an aging effect/mechanism. Section 5.3.2.2 of the report states that it is only when the expansive reaction products become extensive and cause cracking of concrete that aggregate reactivity is considered a deleterious reaction. In addition, NUREG-1801 does not list loss of strength due to reaction with aggregates. NUREG-1801 only lists cracking due to expansion as the aging effect associated with the reaction with aggregates mechanism. Based on this information, NSPM has responded to this portion of the RAI on the premise that the loss of strength due to cement aggregate reaction referred to in ACI 349.3R is a consequence of the cracking aging effect.

RAI-3:

Specify which materials properties are covered by the aging effect "Change in Materials Properties" when referring to the aging mechanism "Leaching of Ca(OH)2" in the concrete pad. Provide technical justification detailing how the proposed Aging Management Activity (i.e., visual examination) will be able to properly identify any 'changes in materials properties' adversely affecting the ability of the pad to perform its intended ITS function during the license period of extended operation.

Table 3.4.1 (NSPM, 2011), "AMR Results for Concrete Pads," identifies leaching of Ca(OH)2 as an aging mechanism of the concrete pad. The corresponding aging effect is named "Change of Materials Properties," yet NSPM (2011) does not provide details on what specific materials properties are to be measured or evaluated. These properties need to be properly defined in the license renewal application and the proposed AMP must include sufficient detail to ensure its adequacy and applicability. An applicable AMP should use a method/technique that ensures detection of the aging effect before there is a loss of the intended safety function. Additionally, the AMP must provide valid acceptance criteria. As an example, if the monitored "change in materials properties" is a reduction in modulus of elasticity, the applicant should provide a valid justification for the adequacy of the proposed method/technique (visual, volumetric, etc.) to characterize such material property, as well as valid acceptance criteria or reference to an applicable consensus standard (e.g., ACI, ASME, ASCE code). Note that ACI 349.3R provides acceptance criteria based solely on visual inspections for specific materials properties and not all materials properties.

This information is needed to determine compliance with 10 CFR 72.42(a)(2), 122(b)(1) and 72.122(f).

NSPM Response:

As explained in the response to RAI-2, the ISFSI AMP was developed using the terminology, aging management tools, and reports provided by the Electric Power Research Institute (EPRI). For concrete pads, NSPM utilized EPRI Report 1002950, "Aging Effects for Structures and Structural Components (Structural Tools)," Revision 1, August 2003. The terminology used to describe aging effects and aging mechanisms in Table 3.4-1 of the LRA (Reference 1), "AMR Results for Concrete Pads," is consistent with the EPRI report.

Section 5.3.3 of the EPRI report describes the potential "Change in Material Properties" aging effect due to various aging mechanisms in concrete structures as follows:

> Change in material properties is evidenced in concrete structures and structural members as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength. These are a result of one or more of the following aging mechanisms: leaching of calcium hydroxide, aggressive chemicals, elevated temperature, irradiation, and creep.

Section 5.3.3.1 of the EPRI report discusses the leaching of Calcium Hydroxide aging mechanism and the potential material properties it could affect as follows:

> Water, either from rain or melting snow, that contains small amounts of calcium ions can readily dissolve calcium compounds in concrete when it passes through cracks, inadequately prepared construction joints, or areas inadequately consolidated during placement. The most readily soluble calcium compound is calcium hydroxide (lime). The aggressiveness or affinity of water to leach calcium hydroxide depends on its dissolved salt content and its temperature (calcium hydroxide is more soluble in cold water). Since leaching occurs when water passes through the concrete, structures that are subject to flowing liquid, ponding, or hydraulic pressure are more susceptible to degradation by leaching than those structures that water merely passes over. When calcium hydroxide is leached away, other cement constituents become exposed to chemical decomposition, eventually leaving behind silica and alumina gels with little or no strength. Leaching over a long period of time increases the porosity and permeability of concrete, making it more susceptible to other forms of aggressive attack and reducing the strength of concrete. Leaching also lowers the pH of concrete and threatens the integrity of the exterior protective oxide film of reinforcing material.

From the above information, the specific material properties covered by the aging effect "Change in Material Properties" when referring to the aging mechanism "Leaching of $Ca(OH)_2$" in the concrete pads are:

- Increase in porosity and permeability
- Reduced strength
- Lower pH

The aging management of the potential change in material properties due to leaching is accomplished by managing the aging mechanism, i.e., by performing visual inspections for the evidence of leaching. By inspecting for deposits of calcium hydroxide (lime), any potential degradation may be managed prior to the leaching mechanism affecting the material properties of the concrete. Thus, the specific material properties listed above do not need to be measured. The use of visual inspections for the presence of leaching as a way of managing this mechanism/effect is consistent with Section 5.2 "Acceptance After Review" evaluation criteria contained in ACI 349.3R. Section 5.2.1, "Concrete surfaces" of ACI 349.3R states:

> The observed condition of the structure shall be compared to the second tier criteria below to determine if the structure is acceptable, requires further evaluation, or repair.
>
> a) Appearance of leaching and chemical attack

The criteria used to determine when a condition is to be entered into the Corrective Action Program (CAP), or when repairs are to be initiated are consistent with the above criteria. Specifically, the periodic structures inspection procedure for performing the visual inspections of the ISFSI concrete pads lists the following acceptance criteria:

> <u>Calcium streaks and deposits</u>
>
> Calcium deposits are typically due to seepage through cracks. If cracks are more than 0.2 mm wide, the procedure refers to crack repair and evaluation guidance. If less than this, seepage shall be monitored to determine if calcium deposits are slowly filling in and sealing the crack, a process known as autogenous healing. If seepage appears to be increasing, a Work Request shall be initiated to repair the crack.

Based on the above acceptance criteria, evidence of leaching, e.g., observed calcium deposits, is monitored via the ISFSI Aging Management Program, with potential degradation reported via the Corrective Action Program. Evidence of leaching is repaired if it appears to be increasing.

Further discussions of the acceptance criteria for the concrete pads are included in the response to RAI-6. Regarding the inclusion of acceptance criteria in the AMP, the ISFSI Aging Management Program is structured the same as the Aging Management Program for the plant. As discussed in the response to RAI-2, this consistency avoids confusion for engineers and other site personnel that would be created if different approaches were used for the ISFSI and PINGP AMPs. Therefore, the inspection procedures specify that if any indication of leaching is observed, an Action Request within the Corrective Action Program is initiated. Further actions would then be evaluated in accordance with the Corrective Action Program, as discussed in the response to RAI-6. No change to the AMP is proposed.

RAI-4:

Revise the license renewal application (LRA) to include a water chemistry program as part of the AMP for the concrete pad. The program should provide results representative of water in the near proximity to the pad. Otherwise, provide an engineering justification for why it is not required for the management of the following aging mechanisms/effects:

- *Cracking or loss of material (spalling, scaling) due to chemical attack; and*
- *Cracking, loss of material, and loss of bond due to corrosion of embedded steel.*

The scope of the ISFSI Inspection and Monitoring Activities Program (Section A.2.1, NSPM, 2011) does not include a water chemistry program. However, response to RAI A-3 (NSPM, 2013) states that NSPM periodically samples water from on-site wells and the Mississippi River for chloride, sulfate, and pH. The applicant used results from samples taken in October 2012 and June 2013 to justify not needing to manage below-grade aging effects due to chemical attack and corrosion of embedded steel. However, periodic evaluation of the water quality in near proximity to the pad is required to ensure that an aggressive soil/ground water environment will not be present throughout the 40 years of extended operation. As stated in ACI 349.3R, chemical attack may occur from exposure to aggressive groundwater, acidic rain/condensation, seawater/salt-spray, exposure to any acids, caustics or other aggressive chemicals (including pesticides for weed and rodent control). The water chemistry program should include sufficient detail about the 10 program elements of an adequate AMP, as detailed in NUREG-1927, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance."

This information is needed to determine compliance with 10 CFR 72.42(a)(2), 122(b)(1), (e) and (f)

NSPM Response:

NSPM has revised the Aging Management Program to include ground water monitoring, as shown in Attachment B to this Enclosure. This revision to the ISFSI Aging Management Program contained in Appendix A to the LRA (Reference 1) addresses the 10 program elements of an AMP detailed in NUREG-1927 (Reference 5), as follows:

1. Scope. Section A2.1.2 of the AMP was revised to include monitoring of ground water chemistry.

2. Preventive Actions. Section A2.2.2 was revised to explain that monitoring of ground water chemistry will be performed to confirm the absence of an aggressive chemical environment for the ISFSI concrete pads.

3. Parameters Monitored or Inspected. Section A2.3.2 was revised to state that ground water chemistry will be monitored for chloride, sulfate, and pH. The intended function monitored by water chemistry analysis is: Provides structural/functional support (SS).

4. Detection of Aging Effects. Section A2.4.2 was revised to explain that monitoring of ground water chemistry is a mitigation activity and does not detect aging effects. PINGP site well water and river water will be sampled every six months to confirm that the concrete pads are not exposed to an aggressive chemical environment.

5. Monitoring and Trending. The concrete pads are the components that would be affected by any aggressive chemicals found in ground water, and monitoring and trending of the condition of the concrete pads is already addressed in Section A2.5.2 of the AMP. As noted in element 4, ground water chemistry does not detect aging effects and no changes are made to this section of the AMP.

6. Acceptance Criteria. Section A2.6.2 was revised to include the following acceptance criteria for ground water chemistry samples:

Chloride	≤	500 ppm
Sulfate	≤	1500 ppm
pH	≥	5.5

   These acceptance criteria are used in the ground water chemistry monitoring program for the PINGP Aging Management Program, and are consistent with the criteria identified in NUREG-1801, Generic Aging Lessons Learned (Reference 6). If any of these acceptance criteria are not met, an Action Request will be initiated in the Corrective Action Program to be evaluated.

7. Corrective Actions. The Corrective Action discussion in Section A2.7.2 of the AMP describes NSPM's Corrective Action Program. CAP requirements are established in accordance with the NSPM Quality Assurance Topical Report and with 10 CFR 50, Appendix B. This discussion applies to all activities described in the AMP and no changes are made to this section of the AMP.

8. Confirmation Process. The confirmation process discussion in Section A2.8.2 of the AMP describes that the NSPM CAP includes features to ensure effective corrective actions. This discussion applies to all activities described in the AMP and no changes are made to this section of the AMP.

9. Administrative Controls. The discussion in Section A2.9.2 of the AMP describes NSPM's formal review and approval processes for aging management activities. This discussion applies to all activities described in the AMP and no changes are made to this section of the AMP.

10. Operating Experience. The discussion in Section A2.10.2 of the AMP describes experience with inspections of ISFSI structures, including the concrete pads associated with ground water monitoring. No changes are made to this section of the AMP.

RAI-5:
Revise inspection frequencies to intervals consistent with ACI 349.3R or provide detailed justifications for any deviations from this criterion. For below-grade areas (underground), justify the use of opportunistic inspections for assessing the condition of the concrete and embedded steel, instead of periodic examination of excavated areas subject to the severest condition.

Section A2.6 (NSPM, 2011), "Acceptance Criteria," states that the acceptance criteria for all visual inspections of the concrete pad will be consistent with, or more restrictive than, those contained in ACI 349.3R. ACI 349.3R states that all safety-related structures should be visually inspected at intervals not to exceed 10 years. Specifically, Table 6.1 in ACI 349.3R, "Frequency of Inspection," states that above-grade (directly and indirectly exposed to a natural environment) and below-grade (underground) structures are to be inspected every five and 10 years, respectively. However, Section A2.4 in NSPM (2011), "Detection of Aging Effects", states that the concrete surface underneath the cask will be inspected every 20 years, and below-grade structures will be inspected only when excavated, exposed or modified.

This information is required to determine compliance with 10 CFR 72.42.; and 72.122(f)and (i); 72.128(a).

NSPM Response:

As explained previously in the response to RAI-2, NSPM did state in Section A2.6.2 of the LRA (Reference 1) that the acceptance criteria for all visual inspections of the concrete pads are consistent with, or more restrictive than, those contained in ACI 349.3R. However, Section A2.6.2 of the LRA is the "Acceptance Criteria" element of the AMP, i.e., NUREG-1927 element 6. It is not the element that addresses frequency of inspections. Section A2.4.2 of the LRA program discusses NUREG-1927 element 4, "Detection of Aging Effects," which includes the frequencies of inspections; Section A2.4.2 does not refer to ACI 349.3R.

The ISFSI concrete pads are fundamentally no different than any other safety related concrete structures on the PINGP site and thus are not susceptible to any unique aging mechanisms or effects. Thus, there is no reason or need to manage the aging effects of the ISFSI pad differently than the other concrete structures on site. Therefore, the frequencies for inspection of the ISFSI concrete pads are the same as the frequencies previously approved for the PINGP Aging Management Program.

The accessible areas of pads will be inspected every five years, as discussed in Section A2.4.2 of the LRA. The inaccessible areas (e.g., below-grade or under a cask) will be inspected if excavated, exposed, or modified, as discussed in Section A2.3.2 of the LRA. These inspection frequencies are consistent with those used for other PINGP concrete structures, and are also consistent with those identified in NUREG-1801 (Reference 6). Section XI.S6, element 4, of NUREG-1801 states the following [emphasis added to relevant sentences]:

> 4. Detection of Aging Effects:
> Structures are monitored under this program using periodic visual inspection of each structure/aging effect combination by a qualified inspector to ensure that aging degradation will be detected and quantified before there is loss of intended function. Visual inspection of high strength (actual measured yield strength ≥ 150 ksi or 1,034 MPa) structural bolting greater than 1 inch (25 mm) in diameter is supplemented with volumetric or surface examinations to detect cracking. Other structural bolting (ASTM A-

325, ASTM F1852, and ASTM A490 bolts) and anchor bolts are monitored for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts. Accessible sliding surfaces are monitored for indication of significant loss of material due to wear or corrosion, debris, or dirt. Visual inspection of elastomeric vibration isolation elements should be supplemented by feel to detect hardening if the vibration isolation function is suspect. The inspection frequency depends on safety significance and the condition of the structure as specified in NRC RG 1.160, Rev. 2. **In general, all structures and ground water quality are monitored on a frequency not to exceed 5 years.** Some structures of lower safety significance, and subjected to benign environmental conditions, may be monitored at an interval exceeding five years; however, they should be identified and listed, together with their operating experience. The program includes provisions for more frequent inspections of structures and components categorized as (a)(1) in accordance with 10 CFR 50.65. Inspector qualifications should be consistent with industry guidelines and standards and guidelines for implementing the requirements of 10 CFR 50.65. Qualifications of inspection and evaluation personnel specified in ACI 349.3R are acceptable for license renewal.

The structures monitoring program addresses detection of aging affects for inaccessible, below-grade concrete structural elements. For plants with non-aggressive ground water/soil (pH > 5.5, chlorides < 500 ppm, or sulfates <1500 ppm), the program recommends: **(a) evaluating the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas and (b) examining representative samples of the exposed portions of the below grade concrete, when excavated for any reason.**

For plants with aggressive ground water/soil (pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm) and/or where the concrete structural elements have experienced degradation, a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

The guidance in NUREG-1801 is based upon industry-wide operating experience of concrete structural elements and thus, provides a sound justification for the proposed inspection frequency. As stated above, the accessible areas of pads will be inspected every five years. This inspection would include areas near the edge of the pad as well as areas near the casks. The inspections of these accessible areas would be used to identify conditions that could indicate the presence of, or result in degradation to the inaccessible areas. In addition, the PI Structures Monitoring Program calls for inspections whenever an inaccessible area becomes accessible for any reason, e.g., if a cask is moved or the soil adjacent to the pad is excavated. Therefore, the concrete pad inspection frequency listed in Appendix A of the LRA is consistent with the NRC endorsed guidance in NUREG-1801.

As noted previously, the frequency of inspections within an Aging Management Program is included in NUREG-1927 element 4, and would therefore be expected to be addressed in Section A2.4.2 of the LRA. NSPM discussed the inspections of opportunity of the concrete pads in Section A2.3.2, Parameters Monitored. To provide additional clarity of the inspection frequency of the concrete pad, NSPM will revise Section A2.4.2 to include inspections of opportunity if inaccessible areas of the pads become accessible, e.g., if a cask is moved or if excavation exposes a below grade portion. See Attachment B of this Enclosure.

RAI-6:
Regarding the NSPM Corrective Action Program (CAP) and the concrete pad:

- *Clarify the criteria applied to determine which inspection results will require either:*
  1. an Action Request;
  2. a modification to the existing AMP; and/or
  3. an official notification to the NRC.
- *Provide details on how the CAP will capture and evaluate operating experience (OE) from other ISFSIs using similar dry cask storage systems where the concrete pad is important to safety. Clarify the CAP criteria applied to determine which external OE will trigger any of the action items listed above.*
- *Provide details on how the CAP will ensure proper monitoring and trending once an aging effect has been identified but not corrected following an earlier inspection.*

Section A2.6 (NSPM, 2011), "Acceptance Criteria," states that the acceptance criteria for all visual inspections of the concrete pad will be consistent with, or more restrictive than, those contained in ACI 349.3R. ACI 349.3R provides three-tier acceptance criteria: (1) acceptance without further evaluation, (2) acceptance after review, and (3) acceptance requiring further evaluation. The staff needs clarification on the CAP criteria used to determine which results under a Tier 2 or Tier 3 acceptance will require either (1) an Action Request, (2) a modification to the existing AMP (e.g., inspection frequency), and/or (3) an official notification to the NRC.

The applicant should clarify any differences in the CAP criteria based on OE obtained at other ISFSIs using similar TN cask systems where the concrete pad is important to safety. The staff also needs details on how the NSPM CAP will update baseline data for the concrete pad based on results from previous inspections, to ensure proper monitoring and trending once an aging effect has been identified.

This information is required to determine compliance with 10 CFR 72.42(a), and 72.172.

NSPM Response:

<u>Clarification of Criteria</u>
As discussed in Section A2.6.2 of the LRA, NSPM utilizes acceptance criteria to determine when a condition is entered into NSPM's Corrective Action Program for evaluation. With regard to the inspections of the concrete pads, the acceptance criteria for all visual inspections are consistent with, or more restrictive than, those contained in ACI 349.3R. The applicable criteria for determining when an evaluation would be initiated are contained in Section 5.2.1 of ACI 349.3R, i.e., the second-tier criteria.

Specifically, the current inspection procedure calls for the condition to be entered into the Corrective Action Program, i.e., through initiation of an Action Request, and requires the condition to be evaluated or repaired if any of the following acceptance criteria are not met:

A. Cracks
   A Work Request shall be initiated to repair dormant cracks between 0.4 and 1.0 mm wide or rationale provided in the inspection report for not requesting repairs. An Action Request shall be initiated to evaluate and correct cracks wider than:

   - 0.4 mm if the cracks are active, or
   - 1.0 mm if the cracks are passive

B. Calcium streaks and deposits
Calcium deposits are typically due to seepage through cracks. If cracks are more than 0.2 mm wide, proceed in accordance with the previous paragraph. If less than this, seepage shall be monitored to determine if calcium deposits are slowly filling in and sealing the crack, a process known as autogenous healing. If seepage appears to be increasing, a Work Request shall be initiated to repair the crack.

C. Surface scaling
If scaling is deeper than 1/4", a Work Request shall be initiated to repair the affected areas. If it exceeds 15.0 square feet, an Action Request shall be initiated to evaluate and correct the condition.

If scaling is more than 3/4" in depth and 8" in diameter, an Action Request shall be initiated to evaluate and correct the condition.

D. Spalling
A Work Request shall be initiated to repair spalls deeper than 3/8" or greater than 2" in diameter. If corroded rebar are exposed or the affected surface area exceeds 1.5 square feet, an Action Request shall be initiated to evaluate and correct the condition.

E. Rust stains
Rust stains can be due to rebar corrosion or corrosion of abandoned concrete anchors, loose scraps of embedded wire, or exposed tips of rebar chairs. The former is a potential problem, and the latter is not. Serious rebar corrosion usually results in spalling. An initial effort shall be made to determine the source of stains including, if necessary, initiation of a Work Request to excavate the area in question. The excavation shall be repaired and, if deeper than half the thickness of the concrete member or corroded rebar is exposed, an Action Request shall be initiated.

F. Failure of old concrete patches
A Work Request shall be initiated to excavate the patched area until sound concrete is reached. The excavation shall be repaired and, if deeper than half the thickness of the concrete member or corroded rebar is exposed, an Action Request shall be initiated.

In addition to evaluating an identified condition to determine if it is acceptable or if a repair is required, the Corrective Action Program also contains provisions to (note this is not a complete list of provisions):

- Determine if the condition is reportable to the NRC, e.g., results in the loss of an intended function
- Perform equipment evaluations, apparent cause evaluations, and root cause evaluations
- Perform functionality assessments
- Address the extent of condition
- Determine actions to prevent recurrence
- Identify Operating Experience Actions
- Trend conditions

Thus, it is through the evaluations conducted as part of the Corrective Action Program that the determination would be made if an Aging Management Program, Monitoring Program, or inspection procedure would be revised. There is a wide range of conditions and locations of observed aging effects that would be entered into the Corrective Action Program, and without knowing the specific circumstances, it is not practical to list "criteria applied to determine which inspection results will require" a modification to the existing AMP.

Operating Experience Program

In addition to the Corrective Action Program, NSPM has a formal Operating Experience (OE) Program. The OE program provides guidance and requirements for sharing, evaluating, translating into station processes and using OE information across the NSPM fleet. The primary objectives of the OE Program are to promote the identification and transfer of lessons learned from internal events to other plants and the industry, as well as to ensure that lessons learned from industry events are incorporated into plant processes and behaviors. The site OE coordinator assembles, prescreens, and distributes OE packages for screening information from a wide variety of sources including, but not limited to:

- Daily downloads from the Institute of Nuclear Power Operations (INPO)
- INPO event reports, and analysis documents
- NRC documents, e.g., Information Notices, Regulatory Issues Summaries, Interim Staff Guidance
- Vendor bulletins
- External industry documents
- INPO, NRC and other industry guidance recommendations, or other program changes or enhancements
- Information obtained by membership/participation in industry owner's groups, working groups and task forces

The program contains provisions for automatically initiating an Action Request within the Corrective Action Program for certain types of documents. After going through a screening process, other documents may be distributed for information only, or a formal Operating Experience Evaluation (OEE) may be performed. The instructions for performing an OEE require that an Action Request be initiated within the Corrective Action Program if it is found that a condition adverse to quality exists. The instructions also include consideration for generating actions that change "license renewal aging management program(s), such as revising implementing procedures or generating new implementing procedures or AMPs." Thus, it is through NSPM's formal OE program that OE from other ISFSIs using similar dry cask storage systems where the concrete pad is important to safety would be evaluated, to determine whether it should be evaluated within the Corrective Action Program and/or result in a change to the Aging Management Program.

Monitoring and Trending

In addition to the trending conducted within the Corrective Action Program, the Structures Monitoring Program requires that the program coordinator evaluate the results of the inspections for adverse trends including an evaluation of whether the frequency of the inspections should be increased. The periodic structures inspection procedure contains requirements to generate an inspection report that includes a section on historical information and trends. This section is to contain relevant maintenance information on the structure collected while preparing for the inspection. At a minimum, it is to identify the status of Work Requests and Action Requests issued during the previous inspection of the structure. The section is to include a discussion of the significance of past and present inspection findings. In particular, this section addresses whether the findings represent an adverse trend or random deficiency indicative of normal structural aging.

RAI-7:

Include the following information in the AMP for the berm:

- *A definition of "absence of aging effects" as it is used as a criteria for satisfactory berm performance as outline in ML12065A073.*
- *The technical basis for the berm inspection frequency criteria as outline in the same response. Provide basis on how operating experience (LRA Section A2.4.2, last sentence page A-6) supports the 5 year inspection frequency.*
- *Identify in the LRA, Table A2.1-1: (i) earthen berm material properties that will change due to desiccation and (ii) visible signs of the effects of the earthen berm material properties change that could be detected by visual inspection.*

Table A2.1-1 indicates that the aging due to a number of degradation mechanisms causes a change in unnamed materials properties. Changes in the materials properties of the component may affect its ability to perform its safety function. In order for a visual examination to determine if a change in properties has occurred, this change must be manifest in a visual affect. Clarification of these broad statements is needed in order to determine what materials properties changes affect the visual appearance of the component and whether visual examination is adequate to determine the changes in these properties.

This information is needed to meet the requirement of 10 CFR 72.42(a)(2).

NSPM Response:

Absence of Aging Effects

As discussed in LRA Section A2.6.2, NSPM utilizes acceptance criteria to determine when a condition is entered into NSPM's Corrective Action Program for evaluation. To ensure that the condition evaluation is performed and any subsequent correction actions may be taken before there is a loss of intended function, very conservative criteria are selected. With regard to the inspections of the berm, the acceptance criterion listed in Section A2.6.2 is the "absence of any of the aging effects listed in Table A2.1-1." Specifically the inspector would look for indications of:

- Slope instability (indication of loss of form aging effect):
  The inspection procedure calls for the inspector to look for indications of sand boils, seepage, slippage of the embankment toe, and dropping of the embankment crown due to more than surface erosion.

- Settlement (indication of loss of form aging effect):
  The inspection procedure relies upon the training and qualification of the inspectors (i.e., civil or structural degree and one or more years of structural inspection experience) to make the determination if settlement has occurred.

- Surface erosion (indication of loss of material and change in material properties aging effects):
  The inspection procedure calls for the inspector to look for indications of rutting, raveling, loss of riprap, and other irregularities which over time have the potential to change embankment height and slope.

The wide range of conditions that may be observed during inspections of the berm are appropriately addressed in the inspection procedure, consistent with other earthen structures included in the Structures Monitoring Program for the PINGP site.

Inspection Frequency
The technical basis for the berm inspection frequency of every five years is that it is consistent with the other earthen structure(s) in the PINGP Structures Monitoring Program, e.g., the Intake Canal, and with the frequency for earthen embankment structures called for in Section XI.S7 of NUREG-1801. In addition, the operational experience of the berm surrounding the ISFSI has not indicated that more frequent inspections are required.

Earthen Berm Material Properties and Dessication
As referred to in Section 3.1.2 of the LRA, NSPM utilized EPRI analysis tools and reports to identify potential aging effects for the berm. The report used for the berm is EPRI Report 1002950, "Aging Effects for Structures and Structural Components (Structural Tools)," Revision 1, August 2003 (Reference 4). To provide direct traceability to the source documents used for the Aging Management Review, Table A2.1-1 of the LRA lists/categorizes the aging effects and mechanisms consistent with how they are listed/categorized in the EPRI report. Section 8.3.3 of the EPRI report describes the potential "Change in Material Properties" due to various aging mechanisms in earthen structures as follows:

> The primary change to material properties for earthen structures is desiccation. Desiccation may occur when soil is exposed to the air for extended periods of time. Water not ionically combined with the soil is drawn toward the surface where it evaporates [4]. Highly plastic preloaded clay is especially susceptible owing to its potential for shrinking and loss in pliability. This, in turn, may cause the exposed surface of the clay to become brittle and flake off or delaminate. Desiccation, coupled with surface flow, wind, or wave action may accelerate these effects of erosion.
>
> Desiccation is not an applicable aging mechanism for submerged structures, since the soil is continuously exposed to water. Examples include areas like the bottom liner of emergency cooling ponds or submerged levees. Therefore, no provisions are required to manage desiccation for continuously submerged portions of earthen structures regardless of the material used in construction.
>
> The effects of desiccation may be controlled through proper material selection and embankment slope during design and construction. In addition, the presence of vegetation can minimize this effect since plants tend to hold moisture in the soil. Periodic inspections may be used to determine if material has been lost due to desiccation.
>
> Earthen structures, specifically canals, dams, emergency cooling ponds, levees, and submerged levees, have specific designs. Each utility should review their current design to ensure that the aging effects of desiccation are addressed. Desiccation is not applicable for submerged levees. Loss of form is an applicable aging effect for earthen structures subjected to desiccation, pending individual plant design review.
>
> [Reference 4 cited above: K. Terzaghi and R. B. Peck, *Soil Mechanics in Engineering Practice*, Second Edition,John Wiley and Sons, Inc., 1967.]

From the above information, the specific material property covered by the aging effect "Change in Material Properties" when referring to the aging mechanism "desiccation" in the berm is the self-adhesion property of the soil. The loss of adhesion may accelerate the effects of erosion. Thus, the aging management of the potential change in material properties due to desiccation is accomplished by managing the potential loss of material, i.e., by performing visual inspections for loss of material due to erosion.

Based on the above, NSPM considers that aging effects are appropriately addressed in the inspection procedures (consistent with the PINGP site Structures Monitoring Program), that the inspection frequency is consistent with NUREG-1801, and that visual examinations can detect indications of desiccation, and no changes to the AMP are included.

RAI-8:

Provide a detailed technical basis for the statement in the AMP that the acceptance criteria for all visual examinations of an in-service cask are the absence of any signs of aging, as indicated in LRA Section Sec A2.6.2 page A-8.

The current acceptance criterion is absence of aging effects as indicated in LRA Section A2.6.2, which is a subjective and vague criterion. The response to RAI A-2 indicates that a calculation was done to show: 1) based on referenced corrosion rates that the maximum expected corrosion of the bottom of the cask should be less than an inch in 100 years. Furthermore, it was calculated that a loss of one inch of the original 7.35 inch bottom plate had no effect on the ability of the cask to perform its safety functions. Hence, a quantitative or actionable operational criterion such as identified in the calculations is required.

This information is needed to meet the requirement of 10 CFR 72.42(a)(2).

NSPM Response:

As discussed in LRA Section A2.6.2, NSPM utilizes acceptance criteria to determine when a condition is entered into NSPM's Corrective Action Program for evaluation. To ensure that the condition evaluation is performed and any subsequent corrective actions may be taken before there is a loss of intended function, very conservative criteria are selected. With regard to the inspections of the cask, the acceptance criterion listed in Section A2.6.2 is the "absence of any of the aging effects listed in Table A2.1-1", i.e., loss of material due to various corrosion mechanisms. Thus, if the inspector observes any indication of corrosion (indication of loss of material aging effect) the condition would be entered into the Corrective Action Program.

To provide additional clarification, NSPM will revise Section A2.6.2 of Appendix A of the LRA to explain that the "absence of any aging effects" means "no observable indications of corrosion." In addition, Section A2.4, "Detection of Aging Effects," will be revised to clarify that visual inspections of the casks will be performed with the unaided eye under general lighting conditions; mirrors, flashlights, and magnifiers may be used as an aid to visual inspections but are not required." These changes to Appendix A of the LRA are shown in Attachment B.

The analysis of the one-inch loss of material that was described in the response to RAI A-2 in Reference 2, was performed to address the expected loss of material from the cask and to evaluate the adequacy of the inspection frequency. This analysis was not intended to determine a quantitative or actionable operational criterion. As stated above, NSPM inspection procedures utilize very conservative criteria when determining if the condition should be entered into the Corrective Action Program. Within the Corrective Action Program, the engineer performing the condition evaluation would utilize all available information (possibly including the calculation) to determine whether the cask is still able to perform its intended functions and also to determine if the inspection frequency needs to be revised.

To presuppose an "actionable operational criterion" based solely on the analysis described in the response to RAI A-2 in Reference 2 may restrict the evaluator's use of other information and lead to non-conservative decisions or actions. Also, a quantitative criterion at which action would be taken could lead to the perception that anything less than that value would be acceptable. This may not be desirable considering the different locations and configurations where corrosion could be observed. Based on these considerations, NSPM does not consider that a "quantitative or actionable operational criterion" should be provided and no changes to the AMP are included.

RAI-9:

Provide conclusive evidence supporting the statement that there was no "observable corrosion or loss of material" in the Response to RAI A-2, page 14. Clarify the use of the observations from the lead cask examination in the operational experience section of the AMP to support the conclusion.

- *Define the meaning of "measurable loss of material" used in discussing the examination of the lead cask (LRA, page A-12).*
- *Place labels indicating salient features on all photographs in the Response to RAI Enclosure 2, Attachments 2, 3, 4 (Inspection Results) that support the statement in the observational experience section of the AMP that there was "no measurable loss of material".*
- *Provide evidence that there were no pits exceeding the acceptance criterion given in the response to RAI M7 under the cask base areas exhibiting corrosion and corrosion product stains indicated in the AMP section under "lead cask inspection."*

Table A2.1-1 indicates that general and pitting corrosion are aging mechanisms for the carbon steel cask. The photographs in the CAP reports were insufficient to support the conclusion that there was no observable loss of material on the base metal of Cask 01. While some photographs used for support are self-explanatory, many photographs need labels to determine the reason the photograph supports the conclusions reached about the degradation.

This information is needed to meet the requirement of 10 CFR 72.42(a)(2).

NSPM Response:

Regarding the reporting of the observations from the lead cask inspections referred to in the RAI, NSPM would like to make two points of clarification:

1) The response to RAI A-2 in Reference 2, does not state that there was no "observable corrosion." The statement on page 14 of 26 of the response to RAI-2 states that "there was no observable loss of material." The response to RAI A-2 states that some corrosion and corrosion product stains were observed. These statements are based on those contained in Section 4.1 of the "ISFSI License Renewal Baseline Inspection Report." (A copy of the inspection report was previously provided to the Staff via Attachment 1 to Enclosure 1 of Reference 7.) The specific statement from the report is:

   > Approximately 25% of the protective coating on the bottom of Cask 01 was observed to have disbondment (loss of adhesion to substrate). In those areas the bare metal did not have any observable loss of material (no depth). Most of the bare metal was clean, however some corrosion and corrosion product stains were observed, mainly in areas where the coating was cracking.

2) When discussing the Lead Cask Inspections on page A-12 of the LRA (Reference 1), NSPM misquoted the Baseline Inspection Report by stating "in those areas, the base metal did not have measureable loss of material." The statement should have said "the base metal did not have observable loss of material (no depth)." NSPM will revise page A-12 of the LRA to correctly reflect the inspection report, as shown in Attachment B.

The RAI requested that NSPM provide conclusive evidence supporting the statements in the response to RAI-2 of Reference 2. The inspections of the lead cask were performed per the guidance in Appendix E of NUREG-1927 which calls for "visual inspection." In addition, visual inspection is the typical inspection technique used to monitor the loss of material aging effect of cask components, as described in Elements 2, 3, and 4 of the ISFSI Inspection and Monitoring Program described in Appendix A of the LRA (Reference 1). With visual inspections, the conclusive evidence of the results is the documented observation report provided by the inspector. In the case of the Prairie Island lead cask inspection, that documentation is the ISFSI License Renewal Baseline Inspection Report, which was provided to the Staff via Attachment 1 to Enclosure 1 of Reference 7.

The RAI requests that NSPM define the meaning of "measureable loss of material." As explained above, the correct phrase should have been "no observable loss of material." In the inspection report, the inspector provided clarification of this statement by including the parenthetical "(no depth)." Thus, the meaning of the statement is that the inspector did not observe any depth to the "corrosion and corrosion product stains" he observed.

The RAI requests NSPM to provide labels indicating the salient features on the photographs of the baseline inspections provided in Attachments 2, 3, and 4 of Enclosure 2 of Reference 2. Copies of the photographs with salient features labeled were provided to the Staff and are available as ADAMS Accession numbers ML13273A827, ML13273A826, ML13273A825, and ML13273A824. It should be noted that these photographs were taken and included in the CAP documentation to provide visual information of the general condition of the components inspected, and not to provide "conclusive evidence" of the observations of the inspector that there was no observable loss of material.

RAI-10:

Provide an AMP for monitoring the degradation of the cask neutron shield. Specifically, the AMP should demonstrate its effectiveness with respect to elements (3), (4), (5), (6), (7), and (8) of an AMP as defined in NUREG-1927, Section 3.6.

In RAI A-1 transmitted to the applicant (ML13035A083), the staff requested the applicant to provide an aging management program that is capable of detecting degradation of the cask neutron shield. In its response, the applicant stated that the TLDs at the PINGP ISFSI will provide dose monitoring to assure that the requirements of 10 CFR 72.104, 72.106, and 10 CFR 72.128(a)(2) are met. In addition, the applicant stated that the PINGP ISFSI Inspection and Monitoring Activities Program reviews the 2-meter dose rate measurements and verifies the absence of an increasing trend on any individual cask. The applicant also stated that any age-related degradation of the shielding material occurs slowly and quarterly monitoring of the TLDs provides sufficient time to detect degradation of shielding materials.

The staff reviewed the applicant's responses to the RAI and determined that the response did not adequately address the concerns for the following reasons:

1. *The TLDs may not be able to detect degradation of the neutron shield for individual casks because it is almost impossible to use the TLD measurement data to detect increases in radiation from individual casks.*

2. *The neutron dose rate measurements at 2-meters from the cask surface presented in the response to the RAI do not seem to be adequate for detecting degradation of the neutron shield as it can be observed that the measurement data provide neither correlation to the decay of the source terms nor any consistency between measurements. Furthermore, it is unclear if these measurements were taken with the same equipment at the same locations and distance away from the casks all the time.*

Although neutron dose rate measurements outside the cask may be used as a means to detect neutron shield degradation, the measurements must be carefully designed. First, the detector must be carefully selected and calibrated as the neutrons from the fuel will be moderated into a wide range of energies after going through the neutron shield. When the neutron shield degrades, the neutrons passing through it will shift to higher energies because the neutron shield functions both as a moderator and an absorber. The detector should be able to detect the shift of the neutron energy. Secondly, the locations of the measurements must be carefully selected to ensure the measured data is consistent and not disproportionately influenced by the radiation from other casks and background. Inaccurate data could result if radiation from surrounding casks as well as background is not considered when taking measurements. Thirdly, the measurements should have sufficient resolution to assure that degradation of any part of the neutron shield can be detected. Unless the degradation is uniform, radiation measurements at one location would only tell if degradation occurred at that location. Fourthly, the data should be analyzed to cull the time dependency of the source terms and irregular measurements for trending analysis. Finally, a time-limited aging analysis may be necessary to determine the degradation of the neutron shield due to heat and irradiation in order to detect cracks and/or unexpected degradation of the neutron shield.

The staff believes that due to either inconsistencies in the measurement technique, equipment, locations, variations in background, or a combination of all of these factors, the data (LRA Figure A.2.10 -1.2, and RAI A-1 response) presented is insufficient to detect degradation of the *neutron shield. The applicant needs to revise its neutron shield degradation monitoring program and demonstrate that the revised program is sufficient and effective for this purpose.*

This information is needed for the staff to meet the requirement of 10 CFR 72.42(a)(2).

NSPM Response:

Clarification and Aging Management Review
Monitoring of the neutron shield material specified in the Prairie Island ISFSI Aging Management Program is intended to detect degradation before there is a loss of intended function. With regards to radiation shielding, NUREG-1927 defines *Radiation shielding* as barriers to radiation that are designed to meet the requirements of 10 CFR 72.104(a), 10 CFR 72.106(b), and 10 CFR 72.128(a)(2). Since the safety analysis in the Prairie Island ISFSI SAR does not credit the neutron shield during accidents, the requirements of 10 CFR 72.106(b) are not applicable. Similarly, the applicable requirements of 10 CFR 72.128(a)(2) are that the system must be designed with suitable shielding for radioactive protection under normal conditions, i.e., to satisfy the requirements of 10 CFR 72.104(a). Thus, the intended function of the neutron shield material is to ensure that the dose rate to a real individual beyond the controlled area boundary from the ISFSI is less than that determined in the safety analysis that demonstrates compliance with the limits in 10 CFR 72.104(a).

Prior to developing an Aging Management Program, NUREG-1927 calls for an aging management review to be performed to identify aging effects that could lead to a loss of intended function. NSPM performed the required review and summarized those aging effects/mechanisms that could lead to a loss of intended function in the tables in Section 3 of the LRA. With regard to the neutron shielding material, Table 3.2-1 shows that NSPM did not identify any aging effects/mechanisms that could lead to a loss of intended function. The review did identify various aging effects such as embrittlement, loss of elasticity, cracking, and radiolytic decomposition. However, the polymeric compounds used for neutron shielding are in solid form and in all cases are encased by a metallic structure. Since these compounds are encased by a rigid structure, these aging effects will not lead to loss of form and thus have no impact on the ability of the neutron shielding material to perform its intended function.

With regard to the concern that cracking of the neutron shielding material could lead to the formation of a neutron streaming path, NSPM concludes that potential cracking will not lead to a loss of intended function, based on the operating experience with the TN-40 and TN-40HT cask designs. SAR Figures 1.3-3 and TN40HT-72-3 show that the radial resins are encased in aluminum boxes 1/8 inch thick. Thus, there are 1/4 inch "paths" through the resin every six inches around the cask. However, during surveys after cask loading and during periodic surveys on the concrete pads, NSPM has not identified neutron streaming through these "paths." Since it is unlikely that potential cracks in the resins could be larger than 1/4 inch in width because of the confinement of the resin within the aluminum boxes, NSPM has concluded that cracking of the resin will not lead to a loss of intended function.

With regard to radiolytic decomposition releasing hydrogen and thus affecting the resins' ability to moderate fast neutrons, the response to RAI 11 includes a calculation of the amount of hydrogen that could be generated from radiolytic decomposition. The calculation shows that the amount of hydrogen that could be generated is only 1.4E-4% by weight of the hydrogen available to moderate fast neutrons. Thus, radiolytic decomposition of the resins will not lead to a loss of intended function.

With regard to thermal stability of the polyester and polyethylene shielding materials, the service temperatures of these materials are 285 °F and 191 °F, respectively. This is lower than the limit of 300 °F provided in SAR Table A3.3-3, "Maximum Temperatures for Hot Normal/Off-Normal Conditions." Therefore, no thermal degradation is expected.

While NSPM's Aging Management Review of the polymeric neutron shielding material did not identify any aging effects that could lead to a loss of intended function, the following aging effect and associated mechanism will be conservatively managed to ensure the continued effectiveness of the neutron shielding material:

- Cracking due to material property changes from radiation exposure

To reflect the management of this aging effect, Section 3.2.3, "Aging Effects Requiring Management," of the LRA (Enclosure 2 to Reference 1) is revised as shown on Attachment D to this enclosure. Attachment D also includes a revision to Table 3.2-1, "AMR Results for Casks," of the LRA. In addition, the ISFSI Inspection and Monitoring Program will be revised as shown in Attachment B, to reflect the inclusion of this aging effect and to clarify that the neutron surveys are part of this program.

<u>Concerns Identified in RAI</u>
The RAI identified two concerns with the response to RAI A-1 that NSPM provided in Reference 2. The following provides additional information and explanation to address these concerns.

<u>Concern #1</u>
The Staff did not find NSPM's response to RAI A-1 adequate to address the concerns because: "*The TLDs may not be able to detect degradation of the neutron shield for individual casks because it is almost impossible to use the TLD measurement data to detect increases in radiation from individual casks.*"

NSPM understands that thermoluminescent dosimeters (TLDs) are primarily used to monitor gamma radiation and are not an effective way to detect degradation of neutron shield material. The response to RAI A-1 included radiation surveys to monitor aging effects of both neutron and gamma shielding materials within the scope of the AMP. Thus, the response included a discussion of how the TLDs are utilized as part of the monitoring program to ensure compliance with 10 CFR 72.104(a) consistent with the guidance in Interim Staff Guidance (ISG)-13. NSPM also understands that TLDs may not be effective in detecting gamma shielding degradation of an individual cask. However, they are effective in demonstrating that the ultimate intended function, i.e., compliance with 10 CFR 72.104(a), is fulfilled.

<u>Concern #2</u>
The Staff also did not find NSPM's response to RAI A-1 adequate to address the concerns because: "*The neutron dose rate measurements at 2-meters from the cask surface presented in the response to the RAI do not seem to be adequate for detecting degradation of the neutron shield as it can be observed that the measurement data provide neither correlation to the decay of the source terms nor any consistency between measurements. Furthermore, it is unclear if these measurements were taken with the same equipment at the same locations and distance away from the casks all the time.*"

NSPM fully understands the difficulty in measuring neutron dose rates due to the variations in neutron energy spectrums between those used to calibrate the neutron meter and those encountered in the field. In addition, NSPM understands the need for consistency between measurements and challenges in interpreting the results. NSPM took these factors into account when developing the neutron shielding monitoring portion of the AMP described in Appendix A of the LRA.

For field measurements, NSPM utilizes the Eberline Rem-Ball meter to monitor neutron dose rates. The Rem-Ball is a moderator-type neutron dose rate meter, i.e. it is surrounded by a poly material that moderates fast neutrons. Thus it requires calibration to known neutron energy fields or neutron field correction factors. The calibration of this meter is based on the neutron energy spectrum associated with heavy-water moderated Cf-252 because it provides a neutron spectrum close to that typically encountered in reactor environments. Thus, the Rem-Ball meter is not calibrated to the neutron energy spectrums that would be encountered outside the neutron shielding of a cask. NSPM also has a REM-500 meter which is a Tissue Equivalent Proportional Counter with dose readings that take into account the energy of the neutrons being counted. This meter includes micro-processing capabilities such that it provides a more accurate neutron dose in fields with an energy spectrum different than that used for its calibration. Unfortunately, this meter is not as robust as the Rem-Ball meter and is not used for routine field measurements.

Recognizing that the neutron field characteristics surrounding the casks are different than the bases for the calibration of the Rem-Ball, NSPM has performed evaluations comparing dose measurements of the Rem-Ball to those from the REM-500. The most recent evaluation was performed in 2013 on a TN-40HT cask. The results of the evaluation showed that along the side of the cask, the Rem-Ball reads 1.5 times higher than the more accurate REM-500. Thus, the "measured" neutron dose rates from the quarterly surveys are conservatively higher than the actual dose rates. This is one of the reasons that NSPM chose to use the absence of an increasing dose rate trend for the acceptance criteria in the AMP rather than comparison to a particular value based on the safety analysis.

The neutron surveys are conducted by taking readings in the same locations every calendar quarter. The survey measurements are taken approximately one meter off the ground directly out from each cask (i.e., perpendicular to the edge of the concrete pads) and at distances of two meters from the cask, 30 centimeters from the cask, and on contact. After reviewing the data, the readings at two meters from the cask were chosen to be the measurements used for trending purposes within the AMP for the following reasons: 1) Due to the large surface area of the cask, there is very little change (if any) in the recorded measured dose rates between the contact, 30-centimeter, and two-meter readings. This is not unexpected since at these distances the cask acts more like a planar source than a point or line source. 2) Within the context of the AMP, these measurements are being used to confirm that the neutron shielding material is still performing its shielding intended function. Readings two meters from the cask provide a better representation of the condition of the shielding material over a larger area than a contact reading would.

Due to the configuration of the casks on the ISFSI pads, it is not practical to shield the impact of neighboring casks on the measurements being taken. However, the impact of neighboring casks would be conservative in that doses from neighboring casks would result in a higher measured reading for an individual cask. This is another reason why NSPM chose to use the absence of an increasing dose rate trend for the acceptance criteria in the AMP rather than comparison to a particular value based on the safety analysis.

Trending neutron dose rate readings from the same location on each cask is consistent with the aging management principle of sampling representative locations and casks rather than 100% of surfaces on all casks, as further discussed in NUREG-1927, AMP Element 4 and Appendix E. While no attempt is made to ensure that the fuel loadings within a cask are truly symmetrical, it is reasonable to assume that the actual loads result in a fairly uniform radial source term. SAR Figure A7.2-2 shows the axial gamma and neutron power profiles, i.e., the axial source term, used in the safety analysis. This figure shows that the highest axial source terms are approximately one meter from the bottom of the cask. Since the most likely aging effect of the shielding material is due to radiation damage, it is reasonable to expect that the most likely location for degradation to occur is approximately one meter from the ground, and that any such degradation would be uniform radially. Therefore, trending neutron radiation readings from every cask, at an elevation of one meter, and at a single radial location two meters from the outer shell provides an adequate sample size to ensure timely detection of aging effects that could lead to a loss of intended function.

A review of the radiation survey data provided in Attachment 1 of Enclosure 2 of Reference 2, shows that many of the radiation protection technicians record dose rates to the nearest whole mrem/hr value. While this is an acceptable practice for documenting radiation surveys, it does contribute to the appearance of large variations between readings when the magnitude of the readings is only a few mrem/hr. The staff expressed a concern that the data provided in the response to RAI A-1 in Reference 2 did not appear to correlate to the decay of the source terms and did not indicate consistency between measurements.

As mentioned above, the apparent inconsistency between measurements is attributed in part to the recording of readings to the nearest whole mrem/hr by different technicians and the normal variation in measured readings relative to the magnitude of the dose rates. The magnitude of the dose rate also contributes to the apparent lack of correlation to the decay of the source term. For example, using the decay constant for the neutron source term provided in SAR Section A7A.7.1 of 0.0358 year^{-1}, and assuming an initial dose rate reading of 5 mrem/hr, the calculated dose rate after 10 years of decay would be 3.5 mrem/hr and after 15 years it would be 3 mrem/hr. This 1.5 to 2 mrem/hr change in dose rate due to the exponential decay of the neutron source term would be difficult to see, considering the variability in measuring such low dose rates. However, if the assumed initial dose rates were closer to the value determined from the safety analysis, i.e., 12 mrem/hr from SAR Figure A7A.5-5, then the calculated dose rate after 10 years of decay would be 8.4 mrem/hr and after 15 years it would be 7 mrem/hr. This larger change in dose rates would be discernable when plotted versus time even taking into account the variability in measuring neutron dose rates.

While there are several challenges in determining a true neutron dose rate from an individual cask for comparison to a value determined from the safety analysis, (e.g., calibration to the energy spectrum, dose from nearby casks, and variability in taking measured readings), they become less significant if the data is periodically reviewed for an increasing trend. As previously discussed, the Rem-Ball meter used for surveys measures high compared to the actual dose. Additional measured dose from adjacent casks would also cause a reading higher than actual. Although these considerations could affect individual dose measurements, they would not significantly affect the trend in measured dose rates. In addition, the variability in measuring the dose rates over time would not affect the ability to determine if there is an increasing trend that would challenge a dose rate determined from the safety analysis. While it may be difficult to correlate the data in the response to RAI A-1 in Reference 2 to the decay of the neutron source term, the data does show that there are no increasing trends that would challenge a dose rate determined from the safety analysis, e.g., 7 mrem/hr after 15 years of decay.

Summary

While NSPM's Aging Management Review of the neutron shielding identified no aging effects that could lead to a loss of intended function, and thus no Aging Management Program is required, NSPM elected to include the quarterly surveys of the neutron dose rate within the scope of the program described in Appendix A of the LRA. These surveys determined a measured neutron dose rate at the same location from each individual cask in storage. These measured dose rates are conservatively higher than the true dose rates due to the type of meters being used and due to the presence of nearby casks. The data for each cask is trended and (after considering factors such as the addition of casks nearby, the variability in measurement and recording techniques), a determination is made if there is an increasing trend. Considering the conservative nature of the measured values and the margin to the safety analysis, NSPM concludes that an increasing trend would be detected prior to the loss of intended function and no changes to the AMP described in Appendix A of the LRA are required.

RAI-11:

Provide TLAA calculations to support the contention that there will be no buildup of flammable hydrogen based on radiolytic degradation of the polymer neutron shielding material from doses expected over the total storage duration (initial plus renewal). If the analysis indicates a buildup of flammable hydrogen then provide a description and an AMP for the safety relief valve.

The response to RAI A-4 in the 2nd, 3rd and last paragraphs was not sufficient. No calculation of the potential hydrogen buildup in the initial storage period was conducted as part of the initial licensing basis. In addition, the response indicated that the dose was too low, no source of ignition was present, and that the steel enclosures were vented with a relief valve. The larger dose from the longer storage period may generate hydrogen, and as acknowledged in the RAI A-4 response, due to the small volumes involved, a flammable concentration may be reached. The response further states that even if there is a build-up of hydrogen there are vent valves to alleviate the situation. Since hydrogen flammability is a safety concern, the relief valves, which prevent a harmful buildup of hydrogen, becomes important to safety if the TLAA calculation indicates that a flammable buildup of hydrogen is possible.

This information is needed to meet the requirement of 10 CFR 72.42(a)(2) and 72.120(d).

NSPM Response:

NSPM has performed a calculation to evaluate the potential for the buildup of flammable gases within the neutron absorbing material enclosure. A copy of the calculation is contained in Enclosure 4 to this letter. The calculation determined the amount of flammable gas that could be generated by determining:

a) The radiolytic G-value (molecules of flammable gas per 100 ev of radiation energy absorbed) using the methodology described in NUREG/CR-6673, "Hydrogen Generation in TRU Waste Transportation Packages,"
b) The amount of energy absorbed in the resins (both the polypropylene top shield and the polyester radial shield) due to gamma and neutron radiation during 60 years of storage,
c) The mass of resins within the enclosures, and
d) The amount of potential flammable gases, which is determined by combining the G-value, energy absorbed, and mass of the resins.

The calculation determined that there could be 4.25E-2 moles of flammable gases generated in the top polypropylene resin and 4.29E-1 moles in the side polyester resin.

It is noted that the calculation in Enclosure 4 includes equation 3-3 to determine flammable gas buildup. This equation includes an "energy deposit rate" which is multiplied by a summation of annual values. Thus, the units of the "energy deposit rate" need to be "per year." Although the "per year" units are not specifically stated in the calculation, NSPM has confirmed with AREVA Inc. that these are the units used in the calculation.

The calculation then utilized information from a Sandia National Laboratory report on the hydrogen "packing capacity" of the resin (see Attachment C to this Enclosure) to determine the maximum amount of hydrogen that may be dissolved in the resin. The calculation determined that the amount of hydrogen that can be dissolved in the resin is significantly larger than the total amount of hydrogen generated during 60 years of storage. Since hydrogen represents the majority of the flammable gases generated due to radiolytic degradation, a negligible amount of flammable gases will be released from the resin.

Although the Sandia report is based on non-irradiated resin materials, AREVA has determined that the calculation in Enclosure 4 remains valid for 60 years of irradiated service. AREVA based this determination on an evaluation performed by AREVA TN Europe in 2001 of polyester resin samples taken after approximately 15 years of service. These samples conformed to hydrogen and boron content requirements and showed no degradation of the material. Based on this evaluation, AREVA does not expect any significant degradation or modification of the characteristics of an irradiated resin as compared to a non-irradiated resin. Thus, the maximum amount of hydrogen that can be dissolved in the resin determined in the calculation in Enclosure 4, based on the Sandia National Laboratory report, can be used to compare to the total amount of hydrogen generated during the 60 years of service.

NSPM will include a discussion of potential flammable gas generation in the ISFSI SAR, as part of license renewal implementation.

RAI-12:
Provide an AMP for the high burnup fuel behavior addressing the required 10 points indicated in NUREG-1927 and include it in the revised Appendix A of the LRA which will be incorporated by reference in the license.

The AMP should be based on the response to RAI 3-2 (NSPM, 2013). Since the response relies on the DOE Cask Demonstration Project, this AMP should be developed to be consistent with the DOE Cask Demonstration test plan (EPRI, 2014), and ISG-24 (NRC, 2013). The site plans may be cited if appropriate and specific section reference citations are provided.

This information is needed to meet the requirement of 10 CFR 72.42(a)(2).

NSPM Response:

NSPM will revise the Aging Management Program contained in Appendix A of the LRA (Reference 1) to include a High Burnup Fuel Monitoring Program, as shown in Section A3.0 of Attachment B.

The High Burnup Fuel Monitoring Program relies upon the joint Electric Power Research Institute (ERPI) and Department of Energy's (DOE) "High Burnup Dry Storage Cask Research and Development Project" (HDRP) to monitor the condition of high burnup spent fuel assemblies in dry storage as a surrogate program for the high burnup fuel being stored at the Prairie Island ISFSI. An alternative program that meets the guidelines of ISG-24, "The Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity for Continued Storage of High Burnup Fuel Beyond 20 Years," (Reference 8) may also be used. Although the program is a confirmatory program, it uses each attribute of an effective AMP as described in NUREG-1927 for the renewal of a site-specific Part 72 license to the extent possible.

References

1. Letter from M.A. Schimmel (NSPM) to Document Control Desk (NRC), "Prairie Island Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application," L-PI-11-074, dated October 20, 2011 (ADAMS Accession No. ML11304A068).

2. Letter from J.E. Lynch (NSPM) to Document Control Desk (NRC), "Supplement to License Renewal Application – Response to Request for Additional Information (TAC No. L24592)," L-PI-13-073, dated July 26, 2013 (ADAMS Accession No. ML13210A272).

3. Letter from P. Longmire, PhD (NRC) to J.E. Lynch (NSPM), "Second Request for Additional Information for Review of the License Renewal Application for the Prairie Island Independent Spent Fuel Storage Installation – SNM-2506 (TAC No. L24592)," dated May 27, 2014 (ADAMS Accession No. ML14147A527).

4. EPRI Report 1002950, "Aging Effects for Structures and Structural Components (Structural Tools)", Revision 1, August 2003.

5. NUREG-1927, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance," NRC.

6. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Rev. 2, NRC.

7. Letter from M.A. Schimmel (NSPM) to Document Control Desk (NRC), "Responses to Requests for Supplemental Information – Prairie Island Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application (TAC No. L24592)," L-PI-12-008, dated February 29, 2012 (ADAMS Accession No. ML12065A073).

8. Interim Staff Guidance (ISG)-24, "The Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity for Continued Storage of High Burnup Fuel Beyond 20 Years," Revision 0, July 2014.

Attachment A

Safety Analysis Report Markups
And Justifications

As described in the response to RAI-1, this attachment describes updates to statements in the Prairie Island Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report (SAR) that refer to a limited storage period, such as 25 years, 20 years, or more generic terms such as storage period and service life. Table RAI-1 provides a summary and justification for each update.

To clarify each change, marked-up pages of the SAR follow Table RAI-1. These marked-up pages are provided for information and no approval is requested.

These changes are in addition to the SAR changes provided in the License Renewal Application (ADAMS Accession No. ML11304A068), Enclosure 3, Attachment C.

Table RAI-1
SAR Changes and Justifications

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
1.2-1	40 year term of PINGP operating license	Deleted sentence	The sentence refers to the term of the operating license for the PINGP reactors and is not related to the operating term of the ISFSI. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
3.2-10	Outer surface temperature above 32F throughout service life	Deleted sentence and added discussion about the vertical storage configuration	The TN-40 (TN-40HT) cask design is a vertical system and thus snow and ice loadings do not need to be considered in the design, other than the potential loading on top of the cask. As stated in the second paragraph of the section, a 50 psf loading is considered on the weather cover, e.g. the top of the cask. Thus, the SAR does not need to credit (or mention) melting of snow and ice.
3.2-15	Cavity pressure at end of life*	Revise the criterion that the cavity pressure must remain above one atmosphere at the end of life to after 20 years in service.	While the SAR currently contains a design requirement that the cask cavity pressure must always remain above one atmosphere,* this design criterion is not credited or relied upon to support an intended function nor is the reduction in internal pressure due to cooling of the gases considered an aging effect. Therefore, the analysis supporting this design criterion does not meet the definition of a TLAA. However, NSPM recognizes that this is still a design criterion that must be satisfied and if needed the casks would be re-pressurized every 20 years. Thus, the wording of the proposed SAR change is to reflect that the analysis is valid for 20 years of service.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
3.3-1	ISFSI license period of 20 years	Delete reference to 20 years	The sentence is simply referring to the duration of the license and does not involve a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
3.3-1	Storage of spent fuel for at least 25 years	Delete reference to 25 years	The sentence is simply asserting that the cask was designed for at least a 25 year service life but does not involve a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
3.3-2	Monitoring system leak rate for 25 year storage period	Clarify that the calculation of the monitoring system's performance is based on a 25 year period and that the system is capable of being re-pressurized if needed.	The interseal pressure monitoring system is not an SSC within the scope of license renewal and thus the calculation is not a TLAA. The calculation was used to define the allowable helium test leakage rate which is now controlled by the ISFSI Technical Specifications. The system's design allows it to be re-pressurized if needed. Therefore, the SAR mark-ups provide clarification of the system's capability including the ability to be re-pressurized.
3.3-4	Cavity pressure decrease is calculated for 20 year period*	Clarify that the calculation of the cavity pressure decrease is based on a 20 year period.	The cavity pressure decrease due to cavity gas temperature reduction is calculated for 20 years.* The mark-up clarifies that this is for the initial 20 year storage period. This discussion of cavity pressure is provided to support the pressure monitoring system, which is not an SSC within the scope of license renewal and thus the calculation is not a TLAA.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
3.3-5	twenty years	No changes needed, therefore no markup is provided	The interseal pressure monitoring system is not an SSC within the scope of license renewal and thus the calculation is not a TLAA. The calculation was used to define the allowable helium test leakage rate which is now controlled by the ISFSI Technical Specifications. The system's design allows it to be re-pressurized if needed. As written the SAR is simply describing the calculation that was performed and no changes are needed.
3.3-14	No degradation of neutron shielding over 25 year storage life	Changed reference from 25 years storage life to through the period of extended operations	The sentence is simply asserting that the SSC was designed for at least a 25 year service life but does not involve a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
3.3-22	TN-40 storage life of 20 years makes it conservative to use CFSM curves for 40 years	Clarify that the calculation is historical and that the cladding temperature limits are now based on ISG-11 Rev, 3 limits.	Amendment 7 of the NSPM ISFSI Technical Specification included a requirement to establish a helium environment within the cask before the temperature limits in ISG-11 Rev. 3 were exceeded. The proposed SAR mark-ups provide clarification that the calculation within the SAR is historical and the results are more restrictive than the limits in ISG-11.
3.4-1	Heat generation limits protect cladding integrity for 40 years	Delete reference to 40 years storage life	The sentence is simply asserting that the SSC was designed for at least 40 years of storage, but within this section it is not being used as a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
Table 3.4-1	Design Life for TN-40 casks	Delete reference to a minimum design life of 25 years	The entry is simply asserting that the SSC has a minimum design life of 25 years but does not involve a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
4.3-1	Cask transport vehicle design for 100 trips over approximately 25 year period	Delete reference to 25 year period	The sentence is simply asserting the approximate time period over which the cask transport vehicle would make the one hundred one-way trips. In addition, the cask transport vehicle is not a SSC within the scope of license renewal. Thus, no TLAA is involved. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
4.6-1	Decommission-ing at end of service lifetime	No change	Statement makes a general reference to when decommissioning would occur.
4.6-1	Cask material activation assumptions over 20 years	Changed to "60 years"	Changed to reflect that a new activation analysis assumed a constant neutron flux. Note that a new sentence is being added (page 4.6-2) to summarize new specific activation calculation for 60 years of storage. In addition, the activation analysis is not relied upon for the determination of the adequacy of the ISFSI design.
Table 4.6-2	-----	Clarified that table is based on 20 years	Revised the table to clarify that it summarizes the results for 20 years of storage.
Table 4.6-3	-----	Added new results	The table is being revised to reflect the results of a specific activation calculation performed for 60 years of storage.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
Table 5.1-2	Major Maintenance once per 20 years	Delete reference to 20 years	The sentence is simply referring to the duration of the license and does not involve a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
Table 7.2-3	20 - yr	No change, therefore no markup is provided	The table is summarizing fission product activities of the fuel for various time periods after discharge from the reactor. The information is not relied upon for the determination of the adequacy of the ISFSI.
Table 7.2-7	20 year	No change, therefore no markup is provided	The table is summarizing fission gas and volatile nuclides inventory of the fuel for various time periods after discharge from the reactor. The information is not relied upon for the determination of the adequacy of the ISFSI.
A3.2-18	Cavity pressure at end of life*	Revise the criteria that the cavity pressure must remain above one atmosphere at the end of life to after 25 years in service.	While the SAR currently contains a design requirement that the cask cavity pressure must always remain above one atmosphere,* this design criterion is not credited or relied upon to support an intended function nor is the reduction in internal pressure due to cooling of the gases considered an aging effect. Therefore, the analysis supporting this design criterion does not meet the definition of a TLAA. However, NSPM recognizes that this is still a design criterion that must be satisfied and if needed the casks would be re-pressurized every 25 years. Thus, the wording of the proposed SAR change is to reflect that the analysis is valid for 25 years of service.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
A3.3-1	Storage of spent fuel for at least 25 years	Delete reference to 25 years	The sentence is simply asserting that the SSC was designed for at least a 25 year service life but does not involve a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
A3.3-2	25 year storage period	Clarify that the calculation of the monitoring system's performance is based on a 25 year period and that the system is capable of being re-pressurized if needed.	The interseal pressure monitoring system is not an SSC within the scope of license renewal and thus the calculation is not a TLAA. The calculation was used to define the allowable helium test leakage rate which is now controlled by the ISFSI Technical Specifications. The system's design allows it to be re-pressurized if needed. Therefore, the SAR mark-ups provide clarification of the system's capability including the ability to be re-pressurized.
A3.3-36	Cavity pressure at end of life and after 25 years of storage*	Revise the criteria that the cavity pressure must remain above one atmosphere at the end of life to after 25 years in service.	While the SAR currently contains a design requirement that the cask cavity pressure must always remain above one atmosphere,* this design criterion is not credited or relied upon to support an intended function nor is the reduction in internal pressure due to cooling of the gases considered an aging effect. Therefore, the analysis supporting this design criterion does not meet the definition of a TLAA. However, NSPM recognizes that this is still a design criterion that must be satisfied and if needed the casks would be re-pressurized every 25 years. Thus, the wording of the proposed SAR change is to reflect that the analysis is valid for 25 years of service.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
A3.4-1	Cladding integrity for 25 years	Delete reference to 25 years of storage	The sentence is simply asserting that the SSC was designed for at least 25 years of storage, but within this section it is not being used as a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
Table A3.4-1	Design Life of 25 years	Delete reference to a minimum design life of 25 years	The entry is simply asserting that the SSC had a minimum design life of 25 years but this does not involve a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
A4.2-9	Material durability over 25 year lifetime	Delete reference to storage lifetime of 25 years or more	The sentence is simply asserting that the SSC was designed for at least 25 years of storage, but within this section it is not being used as a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
A4.2-9	Neutron flux after 25 years	See Appendix C of LRA	This Calculation was identified as involving a TLAA. Thus, markups were provided in Appendix C of the LRA.
A4.2-11	Storage lifetime of 25 years or more	Delete reference to storage lifetime of 25 years or more	The sentence is simply asserting that the SSC was designed for at least 25 years of storage, but within this section it is not being used as a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
A4.6-1	Decommissioning at the end of cask service life	No change	Statement makes a general reference to when decommissioning would occur.
A4.6-1	Activation of cask materials after 20 years	Add "and 60" and include reference to Table 4.6-3	Statement is simply describing the analyses performed for the TN-40 cask design.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
A4.6-2	Cask design life of 25 years	Revised sentence	Revised sentence to reflect that the original calculation was performed for 40 years of storage. Note that a new sentence is being added to summarize a new specific activation calculation for 60 years of storage. In addition activation analysis is not relied upon for the determination of the adequacy of the ISFSI design.
A4.6-2	Analysis for storage for 40 years	Revised sentence	Revised sentence to reflect that the original calculation was performed for 40 years of storage. Note that a new sentence is being added to summarize a new specific activation calculation for 60 years of storage. In addition, the activation analysis is not relied upon for the determination of the adequacy of the ISFSI design.
Table A4.6-1	-----	Clarified that table is based on 40 years	Revised the table to clarify that it summarizes the results for 40 years of storage.
Table A4.6-2	-----	Added new results	The table is being revised to reflect the results of a specific activation calculation performed for 60 years of storage.
A4B.1-10	year	See Appendix C of LRA	This calculation was identified as involving a TLAA. Thus, markups were provided in Appendix C of the LRA.
A7A.7-2	40 years	No change, therefore no markup is provided	Range of cooling for fuel components is based on the rate casks are placed in service at the ISFSI.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
A7A.8-5	Overpressure system pressure calculations over the 25 year life of the cask	Clarify that the calculation of the monitoring system's performance is based on a 25 year period and that the system is capable of being re-pressurized if needed.	The interseal pressure monitoring system in not a SSC within the scope of license renewal and thus the calculation is not a TLAA. The calculation was used to define the allowable helium test leakage rate which is now controlled by the ISFSI Technical Specifications. The system's design allows it to be re-pressurized if needed. Therefore, the SAR mark-ups provide clarification on the system's capability including the ability to be re-pressurized.
A7A.8-5	Monitoring system pressure and cavity leakage during storage period	Clarify that the calculation of the monitoring system's performance is based on a 25 year period and that the system is capable of being re-pressurized if needed.	The interseal pressure monitoring system in not a SSC within the scope of license renewal and thus the calculation is not a TLAA. The calculation was used to define the allowable helium test leakage rate which is now controlled by the ISFSI Technical Specifications. The system's design allows it to be re-pressurized if needed. Therefore, the SAR mark-ups provide clarification on the system's capability including the ability to be re-pressurized.
A7A.8-6	Storage period of 25 years	Delete reference to storage lifetime of 25 years or more	The sentence is simply asserting that the SSC was designed for at least 25 years of storage, but within this section it is not being used as a time-limited assumption in an analysis. Therefore, it is not relied upon for the determination of the adequacy of the ISFSI design.
A7A.8-12	Over 40 years	No change, therefore no markup is provided	Tables identify the time to equalize overpressure system pressure with a postulated Latent Seal failure. This is not relied upon for determination of the adequacy of the ISFSI design.

SAR Page No.	Age-Related Phrase	Summary of Change	Justification
A8.2-8	year	no change, therefore no markup is provided	The interseal pressure monitoring system in not a SSC within the scope of license renewal and thus the calculation is not a TLAA. The calculation was used to define the allowable helium test leakage rate which is now controlled by the ISFSI Technical Specifications. The system's design allows it to be re-pressurized if needed. Therefore, the SAR mark-ups provide clarification on the system's capability including the ability to be re-pressurized.
A8.2-9	year	no change, therefore no markup is provided	The interseal pressure monitoring system in not a SSC within the scope of license renewal and thus the calculation is not a TLAA. The calculation was used to define the allowable helium test leakage rate which is now controlled by the ISFSI Technical Specifications. The system's design allows it to be re-pressurized if needed. Therefore, the SAR mark-ups provide clarification on the system's capability including the ability to be re-pressurized.
A9.7-8	lifetime	No change, therefore no markup is provided	General reference to lifetime of spent fuel storage. No time-limited assumption involved.

* It is noted that NSPM submitted a License Application Request (LAR) on May 23, 2014 (ADAMS Accession No. ML14143A202) to delete the design criterion that cavity pressure will remain above ambient on the coldest day at the "end of life." Approval of the cavity pressure LAR is not required for License Renewal and the LAR is not mentioned in the attached SAR markup. Also, the attached SAR markups do not reflect changes that would result from the cavity pressure LAR.

Marked up SAR pages follow.

1.2 GENERAL DESCRIPTION OF LOCATION

The Prairie Island Nuclear Generating Plant site encompasses about 578 acres and is located within the city limits of Red Wing, Minnesota, in Goodhue County. Northern States Power Company, Minnesota (NSPM*) owns most of the land in the site in fee. The U.S. Army Corps of Engineers controls the land that is not owned by NSPM. The Corps has entered into an agreement with NSPM to prevent residential construction on this land for the life of the power plant.

The Prairie Island site is located on a low island terrace associated with the Mississippi flood plain. It is surrounded by the Vermillion River on the west and by the Mississippi River on the east. The site has been evaluated under the criteria of 10CFR100 prior to issuance of an Operating License for each unit (References 3 and 4). ~~The term of each license is 40 years from the date of issuance.~~

NSP began commercial operation of Prairie Island Nuclear Generating Plant Units 1 and 2, on December 16, 1973, and December 21, 1974, respectively. Westinghouse Electric Corporation designed and supplied the nuclear steam supply system for each unit. Each reactor was originally rated at 1,650 MWt, which is equivalent to approximately 575 MWe (gross), and was uprated in 2010 through a measurement uncertainty recapture power uprate to 1,677 MWt, which is equivalent to approximately 584 MWe (gross) (Reference 8). A complete description of the power plants is contained in the Prairie Island USAR. 01249569

Figure 1.2-1 shows the location of the ISFSI and cask transporter access road in relation to other facilities on the Prairie Island site. The protected area fence surrounding the ISFSI is within the Prairie Island site boundary and exclusion area. The controlled area, which is required by 10CFR72.106 to be established around the ISFSI, corresponds to the site exclusion area boundary. Earthen berms surrounding the ISFSI provide radiological shielding.

* Northern States Power Company was incorporated in Minnesota as a wholly owned subsidiary of Xcel Energy, Inc. effective August 18, 2008.

3.2.4 SNOW AND ICE LOADINGS

The TN-40 is a vertical storage cask system and thus

~~The decay heat of the contained fuel will maintain~~ ~~surface temperature above 32°F throughout its service life,~~ ~~life, with an ambient temperature of -20°F.~~ ~~Therefore, snow or ice will melt when it comes in contact with the cask so that~~ snow and ice loadings need not be considered for the storage cask.

The temperature of the protective cover attached to the top of the cask above the lid could fall below 32°F under certain conditions and a layer of snow or ice might build up. A 50 psf (0.35 psi) snow or ice load as specified in the Prairie Island USAR corresponds to approximately 6 ft of snow or 1 ft of ice. However, this load is insignificant to the TN-40 since the cover is a 0.38 in. thick torospherical steel head which can withstand an external pressure over 20 psi. Therefore, the cover will maintain its intended protective function under these snow or ice loading conditions.

Another possible influence on the TN-40 is a thermal shock when the warm cask is suddenly cooled by cold rain (conservatively assumed at 32°F). A number of such cycles is considered in the thermal fatigue analysis in Section 4.

3.2.5 COMBINED LOAD CRITERIA

3.2.5.1 INTRODUCTION

Sections 3.2.1 through 3.2.4, above, describe the most severe natural phenomena considered in the design of the TN-40. These natural phenomena have been analyzed in those SAR sections where it has been shown that the cask is stable. It will not tip over under any condition or slide on its pad more than about an inch. In addition, the forces and pressures applied to the cask due to these phenomena have been determined.

It should be noted that all of the above phenomena are upper bound, low probability events. In most cases, however, there is a more regular and frequent similar phenomena of lower magnitude. For instance, some small wind load occurs often, but a tornado is unlikely. The forces and pressures determined for the severe phenomena can therefore be conservatively used as upper bound values for all of the similar events.

It has been conservatively assumed that these bounding forces and pressures, with a single exception, can occur at any time and their effects are combined with those due to normal operation. The sole exception is the loading(s) due to the tornado missiles as described in Section 3.2.1.2.2. The missile case is evaluated in combination with others as a low probability event which is postulated only because the consequences of cask penetration might result in severe impact on the immediate environs.

3.2.5.3.3 INTERNAL PRESSURE

The pressure inside the cavity of the storage cask results from several sources. Initially the cavity is pressurized with helium such that the cavity pressure is 2.2 atm at thermal equilibrium. The purpose of pressurizing the cavity above atmospheric pressure is to prevent ~~in-~~leakage of air. The initial pressure is determined on the basis that a 1 atm pressure ~~must exist~~ exists in the cavity on the coldest day ~~at the end of life~~ after 20 years in service. Pressure variations due to daily and seasonal changes in ambient temperature conditions will be small due to the large thermal capacity of the cask.

Fuel clad failure results in the release of fission gas which increases cavity pressure. Under normal storage conditions a 10% fission gas release is assumed due to fuel clad failure. This results in an increase in cavity pressure of 3.6 psi.

Another condition when internal pressure could increase is the cooldown prior to unloading. This could occur at the beginning or end of life. Unloading of fuel at the beginning of life would only be necessary due to excessive leakage past the lid seals or a severe accident, e.g. cask drop. The cask cavity wall temperature at the beginning of life is just below 303°F. Therefore, before returning the cask into a pool, cold water would be pumped into the cavity to reduce the temperature. When the water hits the cavity surface, steam might be produced and the resulting pressure inside the cavity could reach the saturated steam pressure of 71 psia (4.82 atm) corresponding to the cavity wall temperature of 303°F.

Table 3.2-3 presents a summary of internal pressures for the conditions identified. A pressure of 100 psig was chosen as the design internal pressure, since this value exceeds that of all conditions producing an internal pressure. In a response to questions from the NRC Staff, NSP provided justification in Reference 33 for the adequacy of the hydrostatic testing performed on the TN-40 casks during fabrication. In a Safety Assessment dated May 11, 1995 (Reference 31), the NRC concluded that the hydrostatic testing performed on the TN-40 casks was adequate and that TN-40 casks need to be tested with a 25 psig hydrostatic pressure on the inner containment vessel only.

3.2.5.3.4 EXTERNAL PRESSURE

There are several conditions which can result in external pressure on the cask. The external pressure due to flood level is less than 7 psi at the bottom of the cask which corresponds to the 14.7 ft. head of water as discussed in Section 3.2.2.

During fuel loading or unloading the cask is at the bottom of the spent fuel pool, nominally 40 ft. deep. This results in an external hydrostatic pressure of approximately 20 psi.

3.3 SAFETY PROTECTION SYSTEMS

3.3.1 GENERAL

The TN-40 dry storage cask is designed to provide storage of spent fuel ~~for at least 25 years (the ISFSI is licensed for 20 years)~~. The cask cavity pressure is always above ambient during the storage period as a precaution against the in-leakage of air which might be harmful to the fuel. Since the containment vessel consists of a steel cylinder with an integrally-welded bottom closure, the cavity gas can escape only through the lid closure system. In order to ensure cask leak tightness, two systems are employed. A double barrier system for all potential lid leakage paths consisting of covers with multiple seals is utilized. Additionally, pressurization of monitored seal interspaces provides a continuous positive inward and outward pressure gradient which guards against a release of the cavity gas to the environment and the admission of air to the cavity.

3.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

3.3.2.1 CONFINEMENT BARRIERS AND SYSTEMS

A combined cover-seal pressure monitoring system (Figure 3.3-1) always meets or exceeds the requirement of a double barrier closure which guarantees tight, permanent containment. There are two lid penetrations, one for a drain pipe and one for venting and pressurization. When the cask is placed in storage, a pressure greater than that of the cavity is set up in the gaps (interspaces) between the double metallic seals of the lid and the lid penetrations. A decrease in the pressure of the monitoring system would be signalled by a pressure transmitter mounted at the side of the cask (Figure 3.3-1). The system is pressurized through a fill valve mounted near the overpressure tank. Lead shielding will be provided to reduce radiation exposure to the transmitter to acceptable levels.

Connections to the overpressure tank are welded fittings. A quick connect coupling with a diaphragm valve is used to fill the tank.

The Helicoflex metallic face seals of the lid and lid penetrations possess long-term stability and have high corrosion resistance over the entire storage period. These high performance seals are comprised of two metal linings formed around a helically-wound spring. The sealing principle is based on plastically deforming the seal's outer lining. Permanent contact of the lining against the sealing surface is ensured by the outward force exerted by the helically-wound spring. Additionally, all metallic seal seating areas are stainless steel overlay for improved surface control. The overlay technique has been used for Transnuclear's transport casks.

For protection against the environment, a toruspherical protective cover equipped with an elastomer seal is provided above the lid. The lid and cover seals described above are contained in grooves. A high level of sealing over the storage period is assured by utilizing seals in a deformation-controlled design. The deformation of the seals is constant since bolt loads assure that the mating surfaces remain in contact. The seal deformation is set by its original diameter and the depth of the groove.

Metal gasket face seal fittings, diaphragm valves and Helicoflex metallic seals are all capable of limiting leak rates to less than 1 x 10^{-7} atm-cc/sec of helium.

The cask cavity's maximum operating pressure is 2.2 atm. This pressure assures a storage pressure of 1 atm with a minimum ambient temperature of -40°F. The maximum operating pressure of 2.2 atm corresponds to the maximum decay heat load of 0.675 kw per assembly, insolation, a 100°F ambient temperature and the storage of the casks in a 2 x 12 array.

During normal storage, cavity pressure variations due to changing ambient conditions will be small. However, fuel clad failure could result in an increase in cavity pressure due to free gas release of the fuel rods. Based on data from Reference 9, the Exxon assembly contains the most free gas, with 9.04 m^3 at standard temperature and pressure (40 assemblies). The TN-40 cask has a cavity free volume of 6.35 m^3. A 10% release of fission gas would cause an increase in cavity pressure of about 3.8 psi at an average cavity gas temperature of 439°F (Table 3.3-1).

The pressure assuming a 100% fuel failure and the off-normal cavity gas temperature is calculated to be 4.76 atm (70.0 psia).

The initial operating pressure of the monitoring system's overpressure tank is set at 5.5 atm minimum. Over the storage period, the pressure decreases as a result of leakage from the system and as a result of temperature reduction of the gas in the system. Since the level of permeation through the containment vessel is negligible and leakage past the higher pressure of the monitoring system is physically impossible, a decrease in cavity pressure during the storage period occurs only as a result of a reduction in the cavity gas temperature with time. As long as the cavity pressure is greater than ambient pressure and the pressure in the monitoring system is greater than that of the cavity, no in-leakage of air nor out-leakage of cavity gas is possible.

a

The calculations which follow define the monitoring system helium test leakage rate which ensures that no cavity gas can be released to the environment nor air admitted to the casks for ~~the~~ 25 year ~~storage~~ period. If needed during or after the 25 year period, the monitoring system is capable of being repressurized. All seals are considered collectively in the analysis as the monitoring system pressure boundary.

where:

P_0 = *pressure at time* t_0

L_r = *test leak rate (atm-cc/sec)*

V = *monitoring system volume (14,750 cc)*

T = *monitoring system gas temperature (°K)*

t = *time (sec)*

For unchoked flow, the simplified equation in Reference 10, example 21 can be used:

$$L_r = L \frac{\mu}{0.185} \frac{0.99}{P_u - P_d}$$

Substituting into equation (1) and integrating:

$$P = (P_0 - 1) \exp - \{ t L_r / (53.51 V \mu) \} + 1 \qquad (3)$$

where: μ = *viscosity (cp)*

The integration of these equations accounts for the change in the monitoring system leakage rate which occurs during the storage period because of the change in the monitoring system pressure. Although the cask cavity pressure is initially 2.2 atm and remains above ambient during the storage period, it is conservatively assumed in this analysis that both the inner and outer seals of the monitoring system are subject to a constant 1.0 atm downstream pressure.

Since these equations only account for the monitoring system pressure loss due to leakage, a correction of pressure based on a decrease in system gas temperature was employed at the end of each three month time step at which the relation was evaluated.

In order to ensure that the monitoring system pressure is simultaneously greater than the ambient and the cavity gas pressures, both the monitoring system gas and the cavity gas temperatures were established as a function of time. The cavity gas and the monitoring system gas temperatures were calculated for both the beginning and end of storage conditions and they were assumed to decrease linearly during the storage period. The end of storage condition for this calculation was conservatively assumed to be 0.43 kw/assembly and an ambient temperature of 100°F. It was determined that the cavity pressure would decrease from an initial value of 2.2 atm to 1.93 atm ~~in 20 years~~ during the first 20 years of storage due to an average gas temperature reduction. The viscosity of the monitoring system gas was also corrected for temperature change for each successive time increment evaluated.

3.3.2.2.2 RESULTS OF THE THERMAL ANALYSES

Short-Term OFF-NORMAL Storage Conditions

A steady state thermal analysis is performed using the maximum decay heat load of 0.675 kw per assembly (27 kw total), 100°F ambient temperature and an insolation of 135 Btu/hr-ft^2. Figure 3.3-9 shows the temperature distribution predicted by the finite element model. The specific temperature distributions in the hottest cross section of the model, the hottest cross section of the basket, the top 5 in. of the basket, and the cask body are shown in Figures 3.3-10, -11, -12 and -13 respectively. The maximum fuel cladding and average cavity gas temperatures are calculated using the appropriate steady state component temperatures. A summary of the calculated packaging temperatures are listed in Table 3.3-1.

Long-Term NORMAL Storage Conditions

The NORMAL thermal analysis assumes a maximum decay heat of 0.675 kw per assembly, 50°F ambient temperature and a solar heat load of 108 Btu/hr-ft^2.

The steady state temperature distribution in the finite element model is shown in Figure 3.3-14. This is used to calculate the maximum fuel cladding and the average cavity gas temperature. Table 3.3-1 contains a summary of the calculated packaging temperatures.

Evaluation of Packaging Performance

The thermal analysis for normal storage concludes that the TN-40 cask design meets all applicable requirements. The maximum temperatures calculated using conservative assumptions are low. The maximum temperature of any containment structural component is less than 303°F (151°C) which has an insignificant effect on the mechanical properties of the containment materials used. The maximum seal temperature (242°F, 117°C) during normal storage is well below the 570°F long term limit specified for continued seal function. The maximum neutron shield temperature is below 300°F (149°C) and no degradation of the neutron shielding is expected ~~during the 25 year storage life.~~ through the period of extended operations. The long-term maximum fuel cladding temperature is 602°F (317°C) and within allowable fuel temperature limits (Section 3.3.7.1). The minimum temperature of -40°F (-40°C) is also inconsequential to the packaging function.

Buried Cask Thermal Evaluation

The TN-40 cask dissipates heat to the environment by radiation and natural convection. If the packaging is accidentally buried in medium that will not provide the equivalent cooling of natural convection and unrestricted radiation to the environment, component temperatures will increase to a higher steady state condition after long-term burial. Of interest is the containment integrity which is assured as long as the metallic seals remain below 570° and the cavity pressure is less than 100 psig.

The initial temperature limits for the TN-40 design used

The design criteria for the TN-40 dry storage cask require that the maximum fuel cladding temperature of the hottest fuel rod in the cask shall not exceed the temperature limit calculated according to PNL-6189 (Reference 11). This temperature limit has been calculated as a function of fuel age to account for the effect of fuel age on creep deformation and fuel cladding rupture. As the age of fuel increases, its cooling rate rapidly decreases. If the initial fuel temperature is too high at loading, significant creep deformation can occur as a result of the decreasing cooling rates with fuel age. The Commercial Spent Fuel Management Program (CSFM) used the TN-24P packaging as one of its models for developing generic fuel cladding temperature limit curves for 40 year dry storage. The CSFM generic curves are used to establish the fuel cladding temperature limit for 10-year cooled fuel. The ~~TN-40 has a storage life of 20 years and it is conservative to use~~ the CSFM curves developed for 40 year storage.

The temperature limits determined based on the calculation below are more restrictive than those specified in ISG-11 Rev. 3 (Reference 37) and are thus conservative for storage during the period of extended operations.

From Reference 11, the midwall hoop-stress is given by the equation,

$$S_{mhoop}, T_2 = (PD_{mid}/2t)(a)(T_2/T_1)$$

where

S_{mhoop}, T_2 = *the midwall hoop=stress (psi) at temperature of interest* T_2 (°K)

P = *the internal pressure (psi) at the hot-volume average temperature.*

D_{mid} = *the midwall diameter (in.) accounting for cladding corrosion*

t = *the cladding thickness (in.)*

a = *0.95 for PWR fuel assemblies*

3.4 SUMMARY OF STORAGE CASK DESIGN CRITERIA

3.4.1 Cask Design Criteria 01204149

The principal design bases for the TN-40 cask are presented in Table 3.4-1. The TN-40 dry storage cask is designed to store 40 intact 14x14 PWR spent fuel assemblies, with a maximum assembly average burnup of 45,000 MWD/MTU and a minimum cooling time of 10 years.

The maximum total heat generation rate of the stored fuel is limited to 27 kw in order to keep the maximum fuel cladding temperature below the limit necessary to ensure cladding integrity ~~for 40 years storage (Reference 11).~~ The fuel cladding integrity is assured by the limited fuel cladding temperature and maintenance of a nonoxidizing environment in the cask (Reference 23).

The containment vessel (body and lid) is designed and fabricated to the maximum practicable extent as a Class I component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Articles NB-3200. The cask design, fabrication and testing are covered by a Quality Assurance Program which conforms to the criteria in 10CFR72(G).

The cask is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. Poison materials in the fuel basket are employed to maintain $k_{eff} \leq 0.95$ including statistical uncertainties. The TN-40 cask is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornados, lightning, hurricanes and floods. Section 8 describes the cask behavior under these environmental conditions.

3.4.2 Design Basis Limits for Fission Product Barriers (DBLFPBs)

The NRC has defined the design basis limit for a fission product barrier as the controlling numerical value for a parameter established during the license review as presented in the Safety Analysis Report for any parameter(s) used to determine the integrity of the barrier. The list of DBLFPBs for the TN-40 cask is listed in Table 3.4-2. 01204149

27. U.S. Nuclear Regulatory Commission, NUREG 0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, July 1981.

28. IAEA Safety Standards, Regulations for the Safe Transport of Radioactive Material, International Atomic Energy Agency, Vienna, 1985.

29. Gregory, J.J., et. al., "Thermal Measurements in a Series of Large Pool Fires", SAND85-0196, Sandia National Laboratories, Albuquerque, NM, August 1987.

30. Letter, R.O.Anderson (NSP) to U S Nuclear Regulatory Commission, dated March 27, 1995, "Response to Request for Additional Information Regarding NUREG-0612, 'Control of Heavy Loads'".

31. Letter, B.A.Wetzel (NRC) to R.O.Anderson (NSP), transmittal date June 12, 1995, "Safety Evaluation and Safety Assessment Related to Dry Cask Storage At Prairie Island".

32. Safety Evaluation 72-448, TN-40 Cask Weight/Storage Slab Design, May 14, 1996.

33. Safety Evaluation 72-411,TN-40 Dry Cask Hydrostatic Test Compliance, May 9,1995.

34. USNRC, Regulatory Issue Summary 2013-11, Resolution of Licensing Process Expectations for Pressurized Water Reactor Fuel Assemblies Susceptible to Top Nozzle Stress Corrosion Cracking in Dry Cask Spent Fuel Storage and Transportation; September 4, 2013

35. NSP Design Change No 99FH02, Repair Regions D, E and F Spent Fuel Assemblies, Rev 0.

36. 10CFR72.48 Safety Evaluation 573, Rev 0

37. Interim Staff Guidance -11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," November 17, 2003.

01409868

TABLE 3.4-1
DESIGN CRITERIA FOR TN-40 CASKS

Maximum gross weight on crane (with lift beams, without water)	125 tons
Maximum cask height with lid removed	16 ft. 1 in.
~~Minimum design life~~	~~25 years~~
Maximum keff. including bias and uncertainties	$\leq$ 0.95 Normal <0.98 Accident
Payload capacity fuel assemblies	40 intact PWR 14x14
Maximum external dose rate (on storage pad)	200 mrem/hr contact
Spent fuel characteristics	
a) Initial enrichment	3.85%
b) Burnup (max)	45,000 MWD/MTU
c) Cooling time (min)	10 years
d) Decay heat	27 kw (total)
Max clad temperature	340^0C
Cask cavity atmosphere	Helium gas
Maximum internal pressure	100psig
Ambient temperature (Min-Max)	-40^0 to 120^0F
Maximum solar heat load	135 BTU/hr-ft^2
Tornado wind	300mph rotational 60 mph translational
Tornado missiles	4"x12"x144" plank @300 mph; 4,000lb. auto @50 mph
Seismic design earthquake	0.12 g horizontal 0.08 g vertical
Snow and ice	50 psf load

4.3 TRANSPORT SYSTEM

4.3.1 FUNCTION

The function of the transport system is to move the loaded storage casks from the Auxiliary Building rail bay to the concrete pads in the ISFSI. The transport vehicle is capable of being towed over several different types of ground surfaces, including compacted gravel, concrete paving, and asphalt paving. The performance objectives of the transport system are to move the loaded cask in a vertical position in a manner which will preclude damage to the cask body and its internals and to any other safety or security related system or component which could be affected by cask transport. The transport vehicle shall be designed for a minimum of 100 fully-loaded one-way trips ~~over approximately a 25 year period.~~

4.3.2 COMPONENTS

The transport vehicle is designed and fabricated by Ederer, Inc., of Seattle, Washington, based on the general design criteria discussed below. Figures 4.3-1 and 4.3-2 show side and plan views of the transport vehicle.

The Transport vehicle structural frame is fabricated of welded steel plates and shapes sized and connected as required by design stress analysis.

All transport vehicle powered functions are hydraulic. An electro-hydraulic power unit is located on the frame of the transport vehicle. When in operation, the power unit is connected with an outside electrical source at the Auxiliary Building or ISFSI. The power unit consists of a 480V, 3 phase motor coupled to a heavy duty pump. The pump supplies hydraulic fluid to operate the vehicle and cask hoist hydraulic functions. In the event of an electrical power failure, a means is provided at the power unit for lowering the cask. Between power supply locations, the hydraulic operating functions of the hoist are not required and the system is valved closed to prevent hoist movement.

The hoisting mechanism consists of a 'U' shaped steel lift beam with pivot pins at one end connected to the steel structural frame. The other end is raised and lowered by a 12 inch hydraulic cylinder. Lifting links for engagement of the upper cask trunnions are located near the mid-point of each side of the lift beam. The lift links move inward and outward on pins by means of hydraulic cylinders, and can be positioned independently to accommodate off-center positioning of the transport vehicle relative to the cask.

The hydraulic cylinder used for cask lifting is a heavy duty design made of steel. Flow control valves are located on the outlet of the cylinders in order to restrict the flow of oil from the cylinders and control the rate of descent. The operating pressure of the cylinders is designed not to exceed 3,000 psi. The cylinders are equipped with lift cylinder locking valves which assure that the cask will not be accidentally lowered.

4.6 DECOMMISSIONING PLAN

The dry cask design concept to be utilized at the ISFSI features inherent ease and simplicity of decommissioning. At the end of its service lifetime, cask decommissioning could be accomplished by one of several options described below.

The casks, including the spent fuel stored inside, could be shipped to a suitable fuel repository for permanent storage. Depending on licensing requirements existing at the time of shipment off site, placement of the entire cask inside a supplemental shipping container or overpack would be considered.

The spent fuel could be removed from the ISFSI cask and shipped in a licensed shipping container to a suitable fuel repository. If desirable, cask decontamination could be accomplished through the use of conventional high pressure water sprays to further reduce contamination on the cask interior. The sources of contamination on the interior of the cask would be crud from the outside of the fuel pins and the crud left by the spent fuel pool water. The expected low levels of contamination from these sources could be easily removed with a high pressure water spray. After decontamination, the ISFSI cask could either be cut up for scrap or partially scrapped and any remaining contaminated portions shipped as low level radioactive waste to a disposal facility.

For surface decontamination of the ISFSI cask, chemical etching using hydrochloric acid or nitric acid can be applied to remove the contaminated surface of the cask. Alternatively, electropolishing can also be used to achieve the same result.

Cask activation analyses have been performed to quantify specific activity levels of cask materials after years of storage. The following assumptions were made:

1. The cask contains 40 reference PWR assemblies.
2. The neutron flux is assumed constant for ~~20~~ 60 years.
3. The neutron spectrum is the same as in a PWR reactor.

The activation calculation for the first 20 years of storage was ~~is~~ performed using the computer codes ORIGEN2 (Reference 14) with the total neutron fluxes taken from the radial shielding calculation performed with the XSDRN-PM code (see Section 7). The fluxes at the cask centerline, the cavity wall, the neutron shield, and the outer shell are used to irradiate the basket, the body and lid, the neutron shield, and the outer shell and protective cover, respectively. The fluxes, material compositions, and masses or irradiated material are listed in Table 4.6-1. The ORIGEN2 cross section library for PWR's at a burnup of 33,000 MWD/MTU is used. The results listed in Table 4.6-2 indicate that after 20 years irradiation and 30 days decay (to eliminate very short lived radionuclides), the total activity is less than 0.13 Ci.

The specific activity after 60 years of storage was determined by increasing the specific activities calculated for 20 years of storage by a factor of 3.

To evaluate the TN-40 cask and basket for disposal, the specific activity of the isotopes listed in Tables 1 and 2 of 10CFR 61.55 is determined and compared with the limits for Class A wastes in those tables. The actual material volumes of the cask components are used to evaluate their specific activity, rather than diluting the activity over the envelope of the entire cask.

The results of the calculation, shown in Table 4.6-3, show that the TN-40 cask will be far below the specific activity limits for both long and short lived nuclides for Class A waste. Consequently, it is expected that after application of the surface decontamination process as described above, the radiation level due to activation products will be negligible and the cask could be scrapped. A detailed evaluation will be performed at the time of decommissioning to determine the appropriate mode of disposal.

Due to the leak tight design of the storage casks, no residual contamination is expected to be left behind on the concrete base pad. The base pad, fence, and peripheral utility structures will require no decontamination or special handling after the last cask is removed.

The spent fuel pool at Prairie Island Nuclear Generating Plant will remain functional until the ISFSI is decommissioned. This will allow the pool to be utilized to transfer fuel from storage casks to licensed shipping containers for shipment off site if this decommissioning option is chosen.

The volume of waste material produced incidental to ISFSI decommissioning will be limited to that necessary to accomplish surface decontamination of the casks once the spent fuel elements are removed. Furthermore, it is estimated that the cask materials will be only very slightly activated as a result of their long-term exposure to the relatively small neutron flux emanating from the spent fuel, and that the resultant activation level will be well below allowable limits for general release of the casks as noncontrolled material. Hence, it is anticipated that the casks may be decommissioned from nuclear service by surface decontamination alone, which could be performed in the cask decontamination area in the Auxiliary Building.

The costs of decommissioning the ISFSI are expected to represent a small and negligible fraction of the cost of decommissioning the Prairie Island Nuclear Generating Plant.

TABLE 4.6-2

20 year

RESULTS OF ORIGEN2 ACTIVATION CALCULATION

NUCLIDE	**ACTIVITY (Ci)**				
	Basket	Body	N-Shield	Shell	TOTAL
H3	-----	-----	2.12E-10	-----	2.12E-10
C14	-----	2.02E-10	5.12E-10	3.49E-14	5.12E-10
Cr51	3.81E-3	6.35E-4	-----	-----	4.44E-3
Mn54	4.61E-4	8.36E-3	-----	-----	8.82E-3
Fe55	5.06E-3	9.08E-2	-----	1.57E-5	9.59E-2
Fe59	9.36E-5	1.69E-3	-----	-----	1.78E-3
Co58	5.88E-4	6.00E-4	-----	-----	1.19E-3
Co60	8.32E-6	8.31E-6	-----	-----	1.66E-5
Ni63	3.64E-4	3.72E-4	-----	-----	7.36E-4
Ni59	3.17E-6	3.24E-6	-----	-----	6.41E-6
Zn65	-----	-----	1.12E-05	-----	1.12E-5
TOTAL					1.13E-1

<u>NOTE</u>

1. Only nuclides with activity greater than 10^{-5} curie and those nuclides listed in 10CFR61.55 are reported here.

Add the following as a new column.

Specific Activity after 60 yrs of storage
Ci/m^3

6.66E-06
6.51E-11
1.05E-06
4.56E-10
1.98E-13

TABLE 4.6-3

COMPARISON OF TN-40 ACTIVITY WITH CLASS A WASTE LIMITS

after 20 yrs of storage

2.22E-06

Long li… R61.55, Table 1

Zone	Volume m^3	Nuclide	Specific Activity Ci/m^3	Limit Ci/m^3
Basket	1.43	C14	-----	80
		Ni59	~~3.13E-06~~	220
Body	9.27	C14	2.17E-11	80
		Ni59	3.49E-07	
N-Shield	3.37	C14	1.52E-10	
Shell	0.53	C14	6.59E-14	

Add the following as a new column.

Specific Activity after 60 yrs of storage
Ci/m^3
1.75E-05
7.65E-04
2.10E-02
2.69E-06
1.20E-04
3.30E-02
1.88E-10
1.07E-05
8.88E-05

after 20 yrs of storage

Short Lived Isotopes, 10CFR61.55, Table 2, Colum

Zone	Nuclide	Specific Activity Ci/m^3	Limit Ci/m^3
Basket	Co60	5.82E-06	700
	Ni63	2.55E-04	35
	T5*	7.00E-03	700
Body	Co60	8.96E-07	700
	Ni63	4.01E-05	35
	T5*	1.10E-02	700
N-Shield	H3	6.28E-11	40
	T5*	~~8.56E-05~~	700
Shell	T5*	2.96E-05	700

*Nuclides with half lives less than 5 years:
Cr52, Mn54, Fe55, Fe59, Co58, Zn65.

3.56E-06

TABLE 5.1-2

PAGE 3 OF 3

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

OPERATION	NO. OF PERSONNEL	TIME (MIN)	AVG. DISTANCE (ft) FROM CASK
PERIODIC MAINTENANCE			
1. Visual surveillance (NA)	2	15	5
2. Repair surface defects (NA)	2	60	3
3. Instrument testing and calibration	2	180	5
4. Instrument repair (NA)	2	60	3
MAJOR MAINTENANCE ~~**(ONCE IN 20 YEARS)**~~			
1. Replace cask lid seals	3	1950 **	8

* No measurable dose associated with this activity. Therefore, the number of personnel, time and distance are not significant.

\# Parenthetical information corresponds to Table 5.1-1 activity numbers.

** Total time to transfer cask to spent fuel pool, replace lid seals, and return cask to ISFSI pad.

The upper trunnions are designed to meet the requirements of NUREG-0612 (Reference 3) for non-redundant lifting fixture. This is accomplished by evaluating the trunnions to the stress design factors required by ANSI N14.6 (Reference 9), i.e. capable of lifting 6 times and 10 times the cask weight without exceeding the yield and ultimate strengths of the material, respectively. The trunnion loads are shown in Figure A3.2-3 and listed in Table A3.2-3.

The local region of the cask body is conservatively evaluated for a vertical load of 3 g (i.e., 3 times the weight of the cask) which is reacted at the trunnions involved in the handling operation. The factor of 3 provides ample allowance for sudden load application during lifting.

A3.2.5.3.3 INTERNAL PRESSURE

after 25 years in service.

The pressure inside the cavity of the storage cask results from several sources. Initially, the cavity is backfilled with helium to at least 19.5 psia . The purpose of pressurizing the cavity above atmospheric pressure is to prevent in-leakage of air. The initial pressure is determined on the basis that, at minimum, a 1 atm abs pressure ~~must exist~~ in the cavity on the coldest day ~~at the end of life~~. Pressure variations due to daily and seasonal changes in ambient temperature conditions will be small due to the large thermal capacity of the cask. Fuel clad failure results in the release of fission gas which increases cavity pressure. Section A3.3.2.2.6.1 evaluates the increases in pressure due to off-normal and accident scenarios.

exists

Another condition when internal pressure could increase is the cool down prior to unloading. This could occur at the beginning or end of life. Water will be gradually added to the cask during refilling to ensure that the cask pressure limits are not exceeded. See Section A3.3.2.2.5.2 for an evaluation of internal pressure during reflooding.

Table A3.2-2 presents a summary of internal pressures for the conditions identified. A pressure of 22 psig was chosen as the design internal pressure. This value bounds the normal and off-normal operating pressures.

A3.2.5.3.4 EXTERNAL PRESSURE

There are several conditions which can result in external pressure on the cask. The external pressure due to a flood is less than the designed external pressure as discussed in Section A3.2.2.

During fuel loading or unloading the cask is at the bottom of the spent fuel pool, nominally 40 ft. deep. This results in an external hydrostatic pressure of approximately 20 psi.

An explosion on a barge in the vicinity of the Prairie Island plant has been shown to produce an overpressure of less than 2.25 psi at the ISFSI location.

A3.3 SAFETY PROTECTION SYSTEMS

A3.3.1 GENERAL

The TN-40HT cask is designed to provide storage of spent fuel ~~for at least 25 years~~. The cask cavity pressure is always above ambient during the storage period as a precaution against the in-leakage of air which could be harmful to the fuel. Since the containment vessel consists of a steel cylinder with an integrally-welded bottom closure, the cavity gas can escape only through the lid closure system. In order to ensure cask leak tightness, two systems are employed. A double barrier system for all potential lid leakage paths consisting of covers with multiple seals is utilized. Additionally, pressurization of monitored seal interspaces provides a continuous positive pressure gradient which guards against a release of the cavity gas to the environment and the admission of air to the cavity.

A3.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

A3.3.2.1 CONFINEMENT BARRIERS AND SYSTEMS

A combined cover-seal pressure monitoring system (Figure A3.3-1) always meets or exceeds the requirement of a double barrier closure which guarantees tight, permanent confinement. There are two lid penetrations, one for a drain pipe and one for venting and pressurization. When the cask is placed in storage, a pressure greater than that of the cavity is set up in the gaps (interspaces) between the double metallic seals of the lid and the lid penetrations. A decrease in the pressure of the monitoring system would be signaled by a pressure transmitter mounted at the side of the cask (Figure A3.3-1). The system is pressurized through a fill valve mounted near the overpressure tank. Lead shielding will be provided to reduce radiation exposure to the transmitter to acceptable levels.

Connections to the overpressure tank are welded fittings. A quick connect coupling with a diaphragm valve is used to fill the tank.

The Helicoflex metal seals of the lid and lid penetrations possess long-term stability and have high corrosion resistance over the entire storage period. These high performance seals are comprised of two metal linings formed around a helically-wound spring. The sealing principle is based on plastically deforming the seal's outer lining. Permanent contact of the lining against the sealing surface is ensured by the outward force exerted by the helically-wound spring. Additionally, all metal seal seating areas are stainless steel overlay for improved surface control. The overlay technique has been previously used for the TN-40 casks.

For protection against the environment, a torispherical protective cover equipped with an elastomeric seal is provided above the lid. The lid and cover seals described above are contained in grooves. A high level of sealing over the storage period is assured by utilizing seals in a deformation-controlled design. The deformation of the seals is constant since bolt loads assure that the mating surfaces remain in contact. The seal deformation is set by its original diameter and the depth of the groove.

Metal gasket face seal fittings, diaphragm valves, and Helicoflex metal seals are all capable of limiting leak rates to less than 1 x 10^{-7} atm cm^3/sec of helium.

The initial operating pressure of the monitoring system's overpressure tank is set at 5.5 atm abs minimum. Over the storage period, the pressure decreases as a result of leakage from the system and as a result of temperature reduction of the gas in the system. Since the level of permeation through the containment vessel is negligible and leakage past the higher pressure of the monitoring system is physically impossible, a decrease in cavity pressure during the storage period occurs only as a result of a reduction in the cavity gas temperature with time. As long as the cavity pressure is greater than ambient pressure and the pressure in the monitoring system is greater than that of the cavity, no in-leakage of air or out-leakage of cavity gas is possible.

The analyses provided in Appendix A7A define the monitoring system helium test leakage rate which ensures that no cavity gas can be released to the environment nor air admitted to the casks for ~~the~~ a 25 year ~~storage~~ period. All seals are considered collectively in the analysis as the monitoring system pressure boundary.

A3.3.2.2 HEAT TRANSFER DESIGN

The TN-40HT cask is designed to passively reject decay heat under normal conditions of storage and hypothetical accident conditions while maintaining appropriate packaging temperatures and pressures within specified limits. An evaluation of the TN-40HT cask thermal performance is presented in this section. Objectives of the thermal analyses performed for this evaluation include:

- Determination of maximum and minimum temperatures with respect to material limits
- Determination of temperature distributions for analysis of thermal stresses
- Determination of temperatures for containment pressurization

The TN-40HT basket consists of an assembly of 40 stainless steel fuel compartments with aluminum and neutron poison plates sandwiched between them. The compartments are joined by a fusion welding process to 1.75 in. wide stainless steel bars. Above and below the bars are slotted aluminum and neutron poison plates which form an egg-crate structure. Stainless steel basket rails including aluminum inserts are bolted to the basket periphery to provide a conduction path from the basket to the cask cavity wall. This thermal design feature of the basket allows the heat from the fuel assemblies to be conducted along the basket structure to the basket rails and dissipated to the cask cavity wall.

If needed during or after the 25 year period, the monitoring system is capable of being repressurized.

exists

after 25 years in service.

A3.3.2.2.6.2 INTERNAL PRESSURE AT END OF SERVICE LIFE

A minimum helium backfill pressure of 19.5 psia was determined o~~n~~ the basis that a minimum of 1 atm pressure ~~must exist~~ on the coldest day ~~at the end of life~~.

The full-length cask model was run with steady state conditions in the handling building to determine the average cavity gas temperature after completion of the helium backfilling. An ambient temperature of 70 °F is considered for this run. The average gas cavity temperature of 426 °F (886 °R) was retrieved from the model using the methodology described in Section A3.3.2.2.6.1.1. This model did not include the 1.0" gap at each end of the poison and aluminum plates. An evaluation determined that when these gaps are modeled, the cavity gas increased 8°F. Thus, the determination below used an average gas temperature of 434°F (426°F + 8°F). 01368790

after 25 years in service.

The determination of the ~~end-of-life~~ cavity pressure was based on the average gas backfill temperature of 434°F (894°R) at the time of backfill and an average gas temperature of 216°F (676°R) after 25 years of storage an external ambient temperature of -40°F. 01368790

The initial pressure of 19.5psia assures that at the end of 25 years, on the coldest day (–40 °F ambient), the internal pressure of the cask is:

$$P_{cavity} = 19.5 \text{ psia} \times (676°R/894°R) = 14.74 \text{ psi}$$ 01368790

Therefore, the internal pressure of the cask is above the 1 atm minimum.

A3.3.2.2.7 RADIAL HOT GAP BETWEEN THE BASKET RAILS AND THE CASK INNER SHELL

A nominal diametrical cold gap of 0.30 in. is considered between the basket and the cask cavity wall for the TN-40HT cask.

A radial, hot gap of 0.13" at thermal equilibrium is assumed in the ANSYS model for normal storage conditions. To verify this assumption, the hot dimensions of the cask inner diameter and basket outer diameter are calculated at thermal equilibrium as follows.

The outer diameter of the hot basket is:

$$OD_{B,hot} = OD_B + [L_{SS,B} \times \alpha_{SS} (T_{avg,B} - T_{ref}) + L_{Al} \times \alpha_{Al} (T_{avg,Al} - T_{ref})]$$

Where:

$OD_{B,hot}$ = Hot outer diameter of the basket
OD_B = Cold outer diameter of the basket = 72" – 0.30" = 71.70"
$L_{SS,B}$ = Length of basket at 90-270 direction = O_{DB} – 2×0.46" = 70.78"
L_{Al} = Length of aluminum shim = 2×0.46 = 0.92"

A3.4 SUMMARY OF STORAGE CASK DESIGN CRITERIA

The principal design criteria for the TN-40HT cask are presented in Table A3.4-1. The TN-40HT cask is designed to store 40, 14 x 14 PWR spent fuel assemblies with or without fuel inserts. The maximum allowable initial enrichment is 5.0 wt% U-235. The maximum bundle average burnup, maximum decay heat, and minimum cooling time for the fuel assembly are 60 GWd/MTU, 0.80 kW/assembly, and 12 years respectively.

The maximum total heat generation rate of the stored fuel is limited to 32 kW in order to keep the maximum fuel cladding temperature below the limit necessary to ensure cladding integrity ~~for 25 years storage~~. The fuel cladding integrity is assured by the cask and basket design which limits fuel cladding temperature and maintains a non-oxidizing environment in the cask cavity.

The containment vessel (body and lid) is designed and fabricated to the maximum practicable extent as a Class I component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-3200. The alternatives to ASME code requirements are documented in Section A3.5. The cask design, fabrication and testing are covered by a Quality Assurance Program which conforms to the criteria in Subpart G of 10 CFR72.

The cask is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. Poison materials in the fuel basket are employed to maintain the upper subcritical limit of 0.95 minus benchmarking bias and modeling bias. The TN-40HT basket is designed and fabricated to the maximum practicable extent in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200 (Reference 2). The alternatives to ASME code requirements are documented in Section A3.5.

The TN-40HT cask is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornadoes, lightning and floods. Section A8 describes the cask behavior under these accident conditions.

TABLE A3.4-1
DESIGN CRITERIA FOR TN-40HT CASKS

Maximum Gross Weight on Crane (with lift beams, without water)	125 tons
Maximum Cask Height with Lid Removed	16 ft. 1 in.
~~Minimum Design Life~~	~~25 years~~
Upper Subcritical Limit	< 0.95- minus biases
Payload Capacity	≤40 intact 14x14 PWR assemblies (Including inserts)
Spent Fuel Characteristics	
a) Design Basis Initial Enrichment(max)	5.0 wt % U-235
b) Burnup (max)	60 GWD/MTU,
c) Cooling time (min)	12 years
d) Decay Heat	32 kW (total)
Max Clad Temperature	400 °C (752 °F) - Normal 570 °C (1058 °F) -Accident
Cask Cavity Atmosphere	helium gas
Maximum Internal Pressure for Stress Evaluation	100 psig
Minimum/Maximum Ambient Temperature	-40 to 120°F
Daily Averaged Ambient Temperature Over 24 hr. period (min-max)	-40 to 100 °F
Maximum Solar Heat Load (Averaged over 24 hour)	61.488 Btu/Hr-ft^2 (curved surfaces) 122.83 Btu/Hr-ft^2 (horizontal surfaces)
Tornado Wind	360 mph (rotational plus translational)
Tornado Missiles	4000 lb. auto at 50 mph 4" x 12" x 12' wood plank at 300 mph
Cask Drop	18" Drop onto concrete pad or equivalent end drop resulting in 50 g
Seismic Design Earthquake	0.12 g horizontal 0.08 g vertical
Snow and Ice	50 psf load

A4.2.3.4.6 OUTER SHELL

The neutron shield outer shell stress analyses are summarized in Table A4.2-17. The shell stresses are the highest when the cask is vertical and subjected to 3g inertia load and 25 psig internal pressure. Stresses in the shell will be much lower during normal storage of the TN-40HT cask on the ISFSI pad. The outer shell is not analyzed under tornado missile loading, but it could be damaged by either Missile A or Missile B, as defined in Section A3.2.1.2. The effect of any damage to the outer shell is bounded by the cask body structural evaluation where the outer shell is assumed to be completely removed.

A4.2.3.5 MATERIAL DURABILITY

Materials must maintain the ability to perform their safety functions over ~~at least~~ the cask's ~~25 year~~ lifetime under the cask's thermal, radiological, corrosion, and stress environment.

Metallic components

Gamma radiation has no significant effect on metals. The effect of fast neutron irradiation of metals is a function of the integrated fast neutron flux, which is on the order of 10^{14} n/cm^2 inside the TN-40HT cask after 25 years. Studies on fast neutron damage in aluminum, stainless steel, and low alloy steels rarely evaluate damage below 10^{17} n/cm^2 because it is not significant (Reference 14). Extrapolation of the data available down to the 10^{14} range confirms that there will be virtually no neutron damage to any of the TN-40HT cask metallic components.

The effect of the TN-40HT cask temperature environment on the required structural properties is accounted for in the structural evaluation. There is no long term degradation of metals in the TN-40HT cask temperature environment. The effect of creep at temperature is the basis for establishing the seal temperature limits.

The cask exterior carbon steel components are protected from corrosion by the paint (epoxy, acrylic urethane, or equivalent enamel coating). A thermal spray coating may be applied before painting. The interior is protected by a thermal spray coating during loading, and by the helium environment inside the cask during storage. The aluminum, carbon steel, neutron absorber, and stainless steel components are not subject to significant corrosion as discussed in Section A4.2.3.6.

The neutron absorbers (Boral®, borated aluminum, or metal matrix composites) consist of aluminum with boron added in the inert form of boron carbide, aluminum diboride, or titanium diboride. The durability of these materials in the dry storage thermal and radiation environment is similar to that of aluminum.

corrosion to occur. The cask is thoroughly dried before storage by a vacuum drying process. It is then sealed and backfilled with helium.

- The radial neutron shielding materials and the aluminum resin boxes are sealed during all normal operations. The free volume in the sealed region is very small. The resin material is inert after it has cured and does not affect the aluminum boxes or the carbon steel housing.

A4.2.3.6.1 CASK INTERIOR

The TN-40HT cask materials are shown in the Parts List on the Drawings included in Section A1.5. The containment vessel is made from SA-203 Grade E and SA-350 LF3. This low-alloy carbon steel is grit blasted and coated with a Zn/Al metallic spray for corrosion protection.

The aluminum metallic spray coating is subject to the following service environments:

- After fabrication, closed, and shipped under air.
- At fuel loading, borated spent fuel pool water for a short duration.
- Vacuum-dried and helium backfilled for storage ~~lifetime of 25 years or more.~~
- At fuel removal, it may again be exposed to borated spent fuel pool water for a short duration.

The coating is not subject to abrasion except for the one time insertion of the basket.

All sealing surfaces are stainless steel clad by weld overlay. The metallic seals have a stainless steel liner and an aluminum jacket.

The basket is assembled from SA-240 Type 304 stainless steel fuel compartments which are joined to SA-240 Type 304 stainless steel strips by a fusion welding process. Aluminum and neutron absorber plates fit between the fuel compartments. They are not welded or bolted to the stainless steel, but are held in place by the geometry of the compartments and strips. Transition rails made from SA-240 Type 304 stainless steel and aluminum are bolted to the basket.

A4.2.3.6.2 CASK EXTERIOR

The exterior of the cask is carbon steel. The exterior of the cask, with the exception of the trunnion bearing surfaces, is painted using an epoxy, acrylic urethane, or equivalent enamel coating. The paint is selected to be compatible with the pool water and easy to decontaminate. Thermal spray coating prior to painting is optional.

A4.6 DECOMMISSIONING PLAN

The TN-40HT cask design features inherent ease and simplicity of decommissioning. At the end of its service life, cask decommissioning will be preceded by one of the following options:

> Option 1, the TN-40HT cask, including spent fuel in storage, could be shipped to either a monitored retrievable storage system (MRS) or a geological repository for final disposal, or
>
> Option 2, the spent fuel could be removed from the TN-40HT cask (either at the utility or at another off site location) and shipped in a DOE approved cask.

The first option does not require any decommissioning of the TN-40HT cask at Prairie Island Nuclear Generating Plant. No residual contamination is expected to be left behind on the concrete base pad. The base pad, fence, and periphery utility structures will require no decontamination or special handling after the last cask is removed. The ISFSI pad could be demolished with normal construction techniques.

The second option would require decontamination of the TN-40HT cask. The sources of contamination in the interior of the cask would primarily be crud left from the spent fuel pool water or crud from the spent fuel pins. These are expected to be low levels of contamination which could simply be removed with high pressure water spray. After decontamination, the TN-40HT cask could either be cut up for scrap or partially scrapped. For surface decontamination of the TN-40HT cask, electropolishing or chemical etching can be used to remove the contaminated surface of the cask if necessary.

Section 4.6 contains the cask activation analyses for the TN-40 cask to quantify the specific activities of the cask materials after 20 and 60 years of storage. The results of that analysis are shown in Tables 4.6-1 ~~and 4.6-2~~, 4.6-2, and 4.6-3.

Since the TN-40 cask and the TN-40HT cask are very similar in design, the TN-40 activation evaluation can be used to estimate the activity of the TN-40HT cask. Factors are determined that can be applied to results of the TN-40 activation analysis.

The cask bodies are very similar; the TN-40HT has a slightly thinner body and a slightly thicker neutron shield. The TN-40HT basket is heaver than the TN-40 basket with more stainless steel in the basket and the basket rails. The mass factors are shown in the table below:

Zone	TN-40 Mass (kg)	TN-40HT Mass (kg)	TN-40HT Factor
Basket			
SS304	2,770	6,250	2.256
Body, Lid, Rails			
C Steel (SA-105)	56,070	48,563	0.8661
C Steel (SA-203)	12,260	12,555	1.024
SS304	681	3,926	5.765
Neutron Shield			
Resin	4,858	5,573	1.147
Shell, Prot Cover			
C Steel (SA-516)	4,131	4,039	0.9778

To determine the activities after 40 years of storage,

The neutron source term for the high burnup fuel assembly in the TN-40HT cask is 7.59E+08 n/s, (Table A7.2-8). From Table 3.1-2, the neutron source for the TN-40 cask is 2.19E+08 n/s. Therefore, the activation flux in each of the zones (cask centerline, cavity wall, etc.) of the TN-40HT is predicted to be 3.466 times larger (7.59E+08/2.19E+08) than the fluxes determined for the TN-40 (Table 4.6-1). ~~Because the TN-40HT is conservatively analyzed to store fuel for 40 years even though the minimum design life is 25 years;~~ an additional factor of 2 is applied, (TN-40 is 20 years).

Utilizing the nuclide activities reported for the TN-40 in Table 4.6-2 and the mass and activation flux factor shown above, the activities can be estimated for the TN-40HT cask. These values are listed in Table A4.6-1. Note the majority of the activated nuclides come from stainless steel.

To evaluate the TN-40HT cask and basket for disposal, the specific activity of the isotopes listed in Tables 1 and 2 of 10 CFR 61.55 is determined and compared with the limits for Class A waste in those tables.

It is expected that after the application of a surface decontamination method, the radiation levels will be below the acceptable limits of Regulatory Guide 1.86 (Reference 6). The results of the calculation, shown in Table A4.6-2, show that activation of TN-40HT will be far below the specific activity limits for both long and short lived nuclides for Class A waste. A detailed evaluation will be performed at the time of decommissioning to determine the appropriate mode of disposal.

The volume of waste material produced incidental to ISFSI decommissioning is expected to be limited to that resulting from the surface decontamination of the casks if the spent fuel assemblies must be removed.

The actual material volumes of the cask components are used to determine the specific activity values for 40 years of storage. The specific activity after 60 years of storage was obtained by increasing the specific activities calculated for 40 years of storage by a factor of 1.5.

TABLE A4.6-1
RESULTS OF ACTIVATION ANALYSIS

40 yr

Curies per TN-40HTCask

Nuclide	Basket	Body, Lid and Rails	Resin and Al Boxes	Outer Shell & Protective Cover	Total
Cr^{51}	5.959E-02	2.015E-02	-----	-----	7.973E-02
Mn^{54}	7.210E-03	5.379E-02	-----	-----	6.100E-02
Fe^{55}	7.914E-02	5.845E-01	-----	1.064E-04	6.637E-01
Fe^{59}	1.464E-03	1.088E-02	-----	-----	1.234E-02
Co^{58}	9.196E-03	6.222E-03	-----	-----	1.542E-02
Co^{60}	1.301E-04	8.692E-05		-----	2.170E-04
Ni^{63}	5.693E-03	3.855E-03		-----	9.548E-03
Zn^{65}	-----	-----	9.541E-05	-----	9.541E-05
Ni^{59}	7.007E-05	3.915E-05	-----	-----	1.092E-04
H^{3}	-----	-----	1.686E-09	-----	1.686E-09
C^{14}	-----	1.252E-09	4.071E-09	2.365E-13	5.324E-09
TOTAL					8.422E-01

Note: Only the nuclides with activity greater than 10^{-5} curies and those listed in 10 CFR 61.55 are reported here.

TABLE A4.6-2
COMPARISON OF TN-40HT ACTIVITY WITH CLASS A WASTE LIMITS

Add the following as a new column.

After 60 yrs of storage
Ci/m³

6.83E-5
2.19E-10
6.85E-6
1.73E-9
6.88E-13

After 40 yrs of storage

Specific Activity of Long-Lived Isotopes (10CFR61.55 Table 1)

Nuclide	Ci/m^3	Limit (Ci/m^3)	Volume (m^3)	Component
C^{14}	-----	80	1.539	Basket
Ni^{59}	4.551E-5	220		
C^{14}	1.461E-10	80	8.571	Body
Ni^{59}	4.567E-6	220		
C^{14}	1.153E-9	80	3.532	Resin
C^{14}	4.587E-13	80	0.516	Shell

Add the following as a new column.

After 60 yrs of storage
Ci/m³
1.27E-4
5.55E-3
1.53E-1*
1.52E-5
6.75E-4
1.18E-1*
3.10E-4*
7.16E-10
4.05E-5*

After 40 yrs of storage

Specific Activity of Short-Lived Isotopes (10CFR61.55 Table 2)

7.881E-2*

Nuclide	Ci/m^3 "A"	Limit (Ci/m^3) "B"	Volume (m^3)	Component
Co^{60}	8.453E-5	700	1.539	Basket
Ni^{63}	3.698E-3	35		
$T_{1/2}<5$	1.017E-1*	700		
Co^{60}	1.014E-5	700	8.571	Body
Ni^{63}	4.497E-4	35		
$T_{1/2}<5$	~~6.755E-1*~~	700		
$T_{1/2}<5$	2.064E-4*	700	0.516	Shell
H^3	4.773E-10	40	3.532	Resin
$T_{1/2}<5$	2.701E-5*	700		

* - Sum of isotopes with half-life less than 5 years (Cr^{51}, Mn^{54}, Fe^{55}, Fe^{59}, Co^{58}, Zn^{65})

Over the first year, the maximum volume leaked from the OP system is:

$V = 4.489 \times 10^{-5}$ cc/sec × (365 days/year × 24 hrs/day × 3600 sec/hr) = 1416 cc at T_u, P_u

The OP system tank consists of a 6 in. diameter schedule 80 pipe (27 in. long) and two 6 in. diameter schedule 80 end caps. The volume of the tank is 835 in^3. The volume of the OP system is increased to 900 in^3 (14,748 cc) to include the OP system tubing and the space between the metallic seals in the lid and penetrations. Correspondingly, the pressure is reduced by the following in the first year:

$P_{OP\ released} = P_{OP\ Sys,\ Initial} \times \{V_{released} / V_{OP\ Sys}\}$
$P_{OP\ released}$ = 6.349 atm (1416cc / 14748cc) = 0.609 atm

The overpressure system pressure is also corrected for the corresponding drop in temperature over the first year. At the start of the second year, the overpressure system pressure is:

$P_{OP\ start\ of\ second\ year}$ = (6.349 atm -0.609 atm)* (339.6°K/344°K) = 5.666 atm abs

period

These calculations are repeated every year for the 25 year ~~life of the cask~~. Figure A7A.8-2 illustrates the pressure drop from the OP system to the atmosphere. Figure A7A.8-2 also illustrates the pressure drop in the cask cavity due to fuel cooling.

If a leak is to the cask cavity rather than the atmosphere, the pressure drop in the OP system is calculated using a downstream pressure the minimum cavity pressure of 1.37 atm abs (5.43 psig). Figure A7A.8-2 also illustrates the results of this analysis. In this scenario, the corresponding increase in the cask cavity pressure is negligible.

As shown above, the monitoring system pressure is greater than the cask cavity or atmospheric pressure assuming a leak based on the conservative initial acceptance test leak rate of 1×10^{-5} ref cm^3/s. Typically, Helicoflex metal seals result in joints with much lower leak rates than the acceptance criteria. Therefore, no leakage will occur from the cask cavity during the ~~storage period~~.

25 year period

The pressure in the overpressure system will be monitored over the lifetime of the cask. To allow time to diagnose and correct any problems, the OP monitoring system is set to alarm if the overpressure system drops below 2.89 atm abs (27.8 psig). This set point is based on the maximum off-normal cask cavity pressure (32.2 psia from Table A3.3-16) plus 0.7 atm for margin. It ensures that pressure decreases in the overpressure monitoring system are identified well before any potential out leakage from the cask cavity occurs.

If needed during or after the 25 year period, the monitoring system is capable of being repressurized.

A7A.8.5 CONFINEMENT REQUIREMENTS FOR NORMAL CONDITIONS OF STORAGE

The TN-40HT dry storage cask is designed to provide storage of spent fuel ~~for at least 25 years.~~ The cask cavity pressure is always above ambient during the storage period as a precaution against the in-leakage of air which might be harmful to the fuel. Since the containment vessel consists of a steel cylinder with an integrally-welded bottom closure, the cavity gas can escape only through the lid closure system. In order to ensure no release of radioactivity material, two systems are employed. First, all bolted closures are provided with double seals. Second, the interspace between the seals is pressurized to provide a positive pressure gradient. If the inner seals were to leak, helium would flow into the cask cavity and radioactive material would not be released. If the outer seals were to leak, helium would leak from the overpressure system to the exterior, and no radioactive material would be released.

The cask loadings for normal conditions of storage are given in Section A3.2.5. It is shown that the seals are not disturbed by any of the loadings and thus, the cask confinement is maintained.

A7A.8.6 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

A7A.8.6.1 SOURCE TERMS FOR CONFINEMENT CALCULATIONS

Table A7.2-6 lists the activity representing the fission gases, volatiles, and fines contributing more than 0.1% of the activity contained in a design basis fuel, plus Iodine 129.

The releasable source term is first determined. The release fractions (References 8 and 9) applied to the source term are provided below.

Variable	Off-Normal Conditions	Accident Conditions
Fraction of crud that spalls off rods, f_C	0.15	1.0
Fraction of Rods that develop cladding breaches, f_B	0.10	1.0
Fraction of Gases that are released due to a cladding breach, f_G	0.3	0.3
Fraction of Fines that are released due to a cladding breach, f_F	3×10^{-6}*	3×10^{-6}*
Fraction of Volatiles that are released due to a cladding breach, f_V	2×10^{-4}	2×10^{-4}

* Per NUREG-1617, 3×10^{-5} of the fines are released during a cladding breach. Per SAND90-2406, (Reference 10), page IV-7, of the 3×10^{-5} of the fines released recommends that only 10% of the fuel fines ejected remain airborne.

Attachment B

Aging Management Program

Appendix A to LRA
Revision 1

29 pages follow

APPENDIX A

AGING MANAGEMENT PROGRAM

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APPENDIX A: AGING MANAGEMENT PROGRAM

A1.0 Introduction

This appendix is a summary of the activities that manage the effects of aging for the Independent Spent fuel Storage Installation (ISFSI) components that have been identified as being subject to Aging Management Review (AMR). The Aging Management Program (AMP) credited for the management of those aging effects and mechanisms identified for the Prairie Island ISFSI is the ISFSI Inspection and Monitoring Activities Program. This program is a subset of the Prairie Island Nuclear Generating Plant (PINGP) Structures Monitoring Program.

The ISFSI Inspection and Monitoring Activities Program is discussed in Section A2.0. That section provides a description of the ISFSI Inspection and Monitoring Activities Program which includes an introduction, an evaluation in terms of the attributes or elements of an effective Aging Management Program, and a summary. The ten elements, which are part of the ISFSI Inspection and Monitoring Activities Program, are also described. The results of an evaluation of each PINGP program element as compared to each NUREG-1927, "Standard Review Plan for Renewal of Independent Spent Fuel Storage Installation Licenses and Dry Cask Storage System Certificates of Compliance" (Subsection 3.6, Aging Management Program) program element are provided to evidence consistency.

Section 3.0, Aging Management Reviews, provides tables that summarize the results of the AMRs. These tables identify the Aging Management Activity (AMA) credited for managing each aging effect and mechanism for each component or subcomponent listed in the AMR. The AMA manages the aging effects and mechanisms, or the relevant conditions that could lead to those aging effects and mechanisms, applicable to each structure or component and provides reasonable assurance that the integrity of the structure or component will be maintained under current licensing basis conditions during the period of extended operation.

The Aging Management Review of the high burnup fuel spent fuel assemblies in a dry inert environment did not identify any aging effects/mechanisms that could lead to a loss of intended function. However, it is recognized that there has been relatively little operating experience, to date, with dry storage of high burnup fuel. Reference A5.8 provides a listing of a significant amount of scientific analysis examining the long term performance of high burnup spent fuel. These analyses provide a sound foundation for the technical basis that long term storage of high burnup fuel, i.e., greater than 20 years, may be performed safely and in compliance with regulations. However, it is also recognized that scientific analysis is not a complete substitute for confirmatory operating experience. Therefore, Section A3, High Burnup Fuel Monitoring Program, describes a program to confirm that the high burnup fuel assemblies' intended function(s) are maintained during the period of extended operations. Although the program is a confirmatory program, the description below uses each attribute of an effective AMP as described in NUREG-1927 to the extent possible.

A2.0 ISFSI Inspection and Monitoring Program

The Prairie Island ISFSI provides for long-term dry fuel interim storage for spent fuel assemblies until such time that the spent fuel assemblies may be shipped off-site for final disposition. The casks presently utilized at the Prairie Island ISFSI are the Transnuclear TN-40 and TN-40HT (both of which have a 40 fuel assembly capacity) and are designed for outdoor storage. Accordingly, the exterior materials are capable of withstanding the anticipated effects of "weathering" under normal conditions.

The purpose of the ISFSI Inspection and Monitoring Activities Program is to ensure that the structure's or component's intended function(s) is not degraded for the in-service casks, concrete pads or earthen berm.

A description of the ISFSI Aging Management Program is provided below using each attribute of an effective AMP as described in NUREG-1927 for the renewal of a site-specific Part 72 license.

A2.1 Scope of Program

A2.1.1 NUREG-1927 Program Element

NUREG-1927 Program Element 1, Scope of the Program, (Reference A5.1) states "The scope of the program should include the specific structures and components subject to an AMR."

A2.1.2 PINGP Program Element

The ISFSI Inspection and Monitoring Activities Program requires periodic inspection activities that monitor the condition of structures and components within the scope of License Renewal as the method used to manage aging effects.

The aging effects managed by this program are included in Table A2.1-1. The aging effects/mechanisms applicable to each structure and component are dependent upon their associated material/environment combinations, design, and installation. Those structures and components that have been grouped together for aging management review (e.g., Carbon Steel in Atmosphere/Weather) have been evaluated and based upon the materials of construction, design, installation, and environments, will have the same aging effects.

The scope of the ISFSI Inspection and Monitoring Activities Program includes:

1) Visual inspection of the exterior of the in-service casks,
2) Monitoring of the interseal pressure of the in-service casks,
3) Radiation monitoring and associated surveillance activities of the in-service casks,
4) Visual inspection of the concrete pads,
5) Visual inspection of the earthen berm,

6) Visual inspection of an in-service cask bottom prior to the end of the current ISFSI license period,
7) Visual inspection under an in-service cask protective cover (surfaces normally not visible or accessible with the cover in-place) prior to the end of the current ISFSI license period,
8) Visual inspection of the cask bottom in the event an in-service cask is lifted in preparation for movement (inspections of opportunity),
9) Visual inspection under the protective cover (surfaces normally not visible or accessible with the cover in-place) of an in-service cask in the event the cover is removed for maintenance (inspections of opportunity),
10) Visual inspection of the bottom and under the protective cover of the lead cask at least every 20-years, and
11) Monitoring of ground water chemistry.

A2.1.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 1, Scope of the Program.

A2.2 Preventive Actions

A2.2.1 NUREG-1927 Program Element

NUREG-1927 Program Element 2, Preventive Actions, (Reference A5.1) states "Preventive actions should mitigate or prevent the applicable aging effects."

A2.2.2 PINGP Program Element

The ISFSI Inspection and Monitoring Activities Program consists of visual inspections, condition monitoring, and performance monitoring activities to detect degradation of structures and components before the loss of their intended function(s). No preventive or mitigating attributes are associated with these activities.

Aging effects of concrete due to aggressive chemicals were determined not to be applicable to the ISFSI concrete pads due to the lack of exposure to an aggressive chemical environment. To ensure this potential aging mechanism does not become applicable, monitoring of the ground water chemistry will be relied upon as a mitigation program to prevent aging effects from occurring.

A2.2.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 2, Preventive Actions.

A2.3 Parameters Monitored or Inspected

A2.3.1 NUREG-1927 Program Element

NUREG-1927 Program Element 3, Parameters Monitored or Inspected, (Reference A5.1) states "Parameters monitored or inspected should be linked to the effects of aging on the intended functions of the particular structure and component."

A2.3.2 PINGP Program Element

The parameters monitored by the ISFSI Inspection and Monitoring Activities Program are consistent with those identified in industry codes and standards including Electric Power Research Institute (EPRI) Report 1002950, "Aging Effects for Structures and Structural Components (Structural Tools)," EPRI Report 1010639, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," EPRI Technical Report 1007933 "Aging Assessment Field Guide," and American Concrete Institute (ACI) report 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures". The parameters included in the program ensure degraded conditions are identified and corrected by clearly defining degraded condition criteria and associated corrective action requirements to prevent the loss of intended function. Industry and plant specific operating experience (OE) are also reviewed to ensure that parameters inspected focus on conditions identified during these OE reviews. See Table A2.1-1 for a detailed list of aging effects and mechanisms for structures and components inspected or monitored as required by the ISFSI Inspection and Monitoring Activities Program.

The ISFSI Inspection and Monitoring Activities Program also contains provisions to inspect the concrete pads whenever inaccessible (buried) areas are excavated, exposed, or modified.

<u>In-service casks inspections</u>
The condition of the exterior of each in-service cask is inspected visually to ensure the intended functions of the cask exterior are not compromised. Visual inspections will look for signs of damage or deterioration of the cask exterior surfaces. Additionally, the inspections will identify debris accumulating on the cask exterior surfaces. Debris may create the potential for localized conditions to support the corrosion process. The aging effect that is monitored by these inspections is loss of material. The intended functions that are monitored for these subcomponents include:

- Provides heat transfer (HT)
- Maintains a pressure boundary (PB)
- Provides radiation shielding (SH)
- Provides structural/functional support (SS)

<u>Interseal pressure monitoring</u>
The pressure of the helium cover gas is monitored to verify the integrity of the seals of the in-service cask lid and that the intended function is not compromised. The aging effect that is monitored by these inspections is loss of material. The intended function that is monitored for this subcomponent is:

- Maintains a pressure boundary (PB)

Radiation surveys
Radiation surveys (gamma and neutron) as well as continuous monitoring via thermoluminescent dosimeters (TLD) at the ISFSI site boundary are used to verify that the radiation levels remain within the specified limits and that the shielding materials in the in-service casks are intact and are effectively performing their intended function. Degradation in the effectiveness of the shielding material would be detected by a corresponding increase in radiation levels. The aging effects that are monitored by this monitoring are the loss of the gamma shielding material and cracking of the neutron shielding material. The intended function that is monitored for this subcomponent is:

- Provides radiation shielding (SH)

Concrete pad inspections
A visual inspection of the accessible areas of the concrete pads is performed to determine that no deterioration has occurred and that the intended function is not compromised. The aging effects that are monitored by these inspections are change in material properties, cracking and loss of material. The intended function that is monitored for this subcomponent is:

- Provides structural/functional support (SS)

Earthen berm inspections
The earthen berm surrounding the ISFSI is visually inspected to determine that no deterioration has occurred and that the intended function is not compromised. The aging effects that are monitored by these inspections are change in material properties, loss of form and loss of material. The intended function that is monitored for this subcomponent is:

- Provides radiation shielding (SH)

Lead cask inspections
Additionally, a visual inspection of an in-service cask bottom ("lead cask") was performed in June 2011, prior to the end of the current ISFSI license period. This visual inspection was performed to primarily ensure that there was no unanticipated degradation and the intended functions were not compromised. This inspection looked for signs of deterioration in the normally inaccessible area underneath the cask to determine the general condition of the cask bottom. This inspection was considered representative of the total population of the in-service casks based on the service period involved, material, and environment. The aging effect that was monitored by these inspections was loss of material.

A visual inspection under two in-service casks protective covers was also performed in June 2011, prior to the end of the current ISFSI license period. This visual inspection was primarily performed to ensure that there was no unanticipated degradation and the intended functions were not compromised. This inspection looked for signs of deterioration in the normally inaccessible area underneath the protective covers. This inspection was considered representative of the total population of the in-service casks based on the service period involved, materials, and environment. The aging effect that is monitored by these inspections was loss of material.

Ground Water Chemistry
Monitoring of ground water chemistry for Chloride, Sulfate, and pH is used to verify that the concrete pads are not being exposed to an aggressive chemical environment, thus preventing aging effects due to aggressive chemicals from occurring. The intended function monitored by this activity is:

- Provides structural/functional support (SS)

A2.3.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 3, Parameters Monitored or Inspected.

A2.4 Detection of Aging Effects

A2.4.1 NUREG-1927 Program Element

NUREG-1927 Program Element 4, Detection of Aging Effects, (Reference A5.1) states "Detection of aging effects should occur before there is a loss of any structure and component intended function. This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection, and timing of new or one-time inspections to ensure timely detection of aging effects."

A2.4.2 PINGP Program Element

A condition examination is an acceptable method used to identify aging effects and is consistent with methods provided in industry codes and standards.

Additionally, the ISFSI Inspection and Monitoring Activities Program requires inspection personnel to be trained and technically qualified to perform these examinations. The personnel evaluating the structural examination results (concrete pads and earthen berm) are degreed engineers with one or more years of structural inspection experience. The personnel evaluating the cask examination results shall be qualified in accordance with PINGP site-specific requirements.

Quarterly visual inspections of the physical condition of the exterior surfaces of all in-service casks provide a means to detect degradation of these components due to potential loss of material and confirm that the intended functions are not compromised. The visual inspections of the casks will be performed with the unaided eye under general lighting conditions; mirrors, flashlights, and magnifiers may be used as an aid to visual inspections but are not required.

Pressure monitoring of all in-service casks is performed as a continuous process and checked daily for alarms. This provides a means to detect metallic O-ring seal degradation due to potential loss of material and confirm that the intended function is not compromised.

Radiation monitoring at the ISFSI boundary and quarterly radiation surveys (gamma and neutron) of the casks provide a means to detect shielding material degradation of the in-service casks and confirm that the intended function is not compromised.

Visual inspections of the accessible areas of the concrete pads every five years, and inspections of opportunity of inaccessible areas (e.g., if a cask is moved or excavation of a below grade portion), provide a means to detect degradation of these areas due to potential change in material properties, cracking, and loss of material. These inspections confirm that the intended function is not compromised.

Visual inspections of the earthen berm on a five-year frequency provide a means to detect degradation due to potential change in material properties, loss of form, and loss of material. These inspections confirm that the intended function is not compromised.

Visual inspections of the bottom of an in-service cask as an inspection of opportunity and, as a minimum, at 20-year intervals for the lead cask, provide a means to detect degradation of the bottom material due to potential loss of material and confirm that the intended functions are not compromised.

Visual inspections underneath the protective cover of an in-service cask as an inspection of opportunity and, as a minimum, at 20-year intervals for the lead cask, provide a means to detect degradation due to potential loss of material and confirm that the intended functions are not compromised.

Visual inspections of the ISFSI structures and components provide reasonable assurance that any degradation of the in-service casks, concrete pads, or earthen berm is identified and confirm that the structure or component intended function(s) is not compromised.

A review of plant-specific operating experience and industry operating experience for plants with similar materials and site conditions found that aging degradation occurs slowly over time and that an inspection frequency of once every five years was sufficient for the detection of aging effects before any loss of intended function for the concrete pads and earthen berm. This has also been confirmed by this same performance frequency of once every five years for those structures and components within the scope of the Maintenance Rule (10 CFR 50.65) such as the Reactor Containment Vessels, Shield Buildings, Auxiliary Buildings, etc. The ISFSI Inspection and Monitoring Activities Program contains provisions to accelerate the frequency of the examinations based on inspection results.

Monitoring of ground water chemistry is a mitigation activity and does not provide for detection of aging effects. Sampling well water and river water every six months for Chloride, Sulfate and pH provides a means to confirm that the concrete pads are not exposed to an aggressive chemical environment.

A2.4.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 4, Detection of Aging Effect.

A2.5 Monitoring and Trending

A2.5.1 NUREG-1927 Program Element

NUREG-1927 Program Element 5, Monitoring and Trending, (Reference A5.1) states "Monitoring and trending should provide for prediction of the extent of the effects of aging and timely corrective or mitigative actions."

A2.5.2 PINGP Program Element

The ISFSI Inspection and Monitoring Activities Program, as a subset of the PINGP Structures Monitoring Program, requires monitoring the condition of structures and components using current and historical operating experience along with industry operating experience to detect, evaluate, and trend degraded conditions. When degraded conditions are detected and all associated corrective actions are complete, the structures and components are again monitored against established performance goals. The program ensures the original design basis for the structures and components is maintained by effectively managing the applicable aging effects.

Periodic visual inspections determine the potential existence of loss of material for the in-service cask exterior surfaces and accumulation of debris. The inspection frequency is quarterly. Pressure monitoring of each in-service cask to detect potential loss of material is provided as a continuous process and checked daily for alarms. Radiation levels at the ISFSI site are continuously monitored and are evaluated and recorded quarterly to detect the potential for shielding material degradation. Surveys associated with facility entry and/or cask placement are performed as required and supplement the overall radiation monitoring program. The concrete pads are visually inspected at least once every five years for any evidence of change in material properties, cracking, or loss of material. The earthen berm is visually inspected at least once every five years for any evidence of change in material properties, loss of form, and loss of material. A visual inspection of an in-service cask bottom and a visual inspection of the area underneath an in-service cask protective cover were performed in June 2011. Subsequent inspections of normally inaccessible areas of the cask bottoms and area underneath the protective cover will be performed on an inspection of opportunity basis and, as a minimum, at 20-year intervals for the lead cask.

All observations regarding the material condition of the ISFSI are recorded in inspection procedures. The ISFSI Inspection and Monitoring Activities Program includes a process used to evaluate past and current conditions of structures and components and to determine whether they represent an adverse trend or random deficiency indicative of normal aging. If degradation exceeds or appears that it will exceed that expected of a properly maintained structure or component, a corrective action is generated requiring further engineering evaluation. All degraded conditions that result in a corrective action are trended in accordance with the Corrective Action Program.

A2.5.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 5, Monitoring and Trending

A2.6 Acceptance Criteria

A2.6.1 NUREG-1927 Program Element

NUREG-1927 Program Element 6, Acceptance Criteria, (Reference A5.1) states "Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the particular structure and component intended functions are maintained under the existing licensing-basis design conditions during the period of extended operation."

A2.6.2 PINGP Program Element

The ISFSI Inspection and Monitoring Activities Program includes acceptance criteria for when the condition is to be entered into the Corrective Action Program before there is a loss of intended function. The acceptance criteria include sufficient detail to ensure timely detection of any degraded condition, followed by an evaluation in the Corrective Action Program to ensure that the particular structure or component intended function(s) is maintained under the existing licensing basis design conditions. Industry and plant-specific OE are also reviewed to ensure that the ISFSI Inspection and Monitoring Activities Program's acceptance criteria focus on conditions identified during these OE reviews.

The acceptance criteria for all visual inspections of an in-service cask are the absence of any of the aging effects listed in Table A2.1-1, i.e., no observable indications of corrosion.

The acceptance criterion for interseal pressure monitoring is the absence of an alarmed condition. The alarm setpoint is higher than the interseal pressure specified in the Prairie Island ISFSI Technical Specification 3.1.5.

The acceptance criterion for radiation dose monitoring of an in-service cask is the absence of an increasing trend.

The acceptance criteria for all visual inspections of the concrete pads are consistent with, or more restrictive than, those contained in Section 5.2.1 of ACI 349.3R (Reference A5.6), i.e., the second-tier criteria.

The acceptance criteria for all visual inspections of the earthen berm are the absence of any of the aging effects listed in Table A2.1-1.

The acceptance criteria for the ground water chemistry monitoring are Chloride ≤ 500 ppm, Sulfate ≤ 1500 ppm, and a pH ≥ 5.5.

The "Structures Monitoring Program," which invokes the ISFSI Inspection and Monitoring Activities Program, includes a three tiered classification of inspection findings, namely, "Acceptable," "Acceptable with Deficiencies," and "Unacceptable." An "Acceptable" condition is described as a structure or component capable of performing its intended function free of unexpected deficiencies or degradation. The "Acceptable with Deficiencies" condition is described as a structure or component considered capable of performing its intended function, but has accelerated degradation or unexpected deficiencies which, without special attention, could shorten its design life. An "Unacceptable" condition refers to a structure or component that has been damaged or degraded such that it may not be capable of performing its intended function.

A2.6.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 6, Acceptance Criteria.

A2.7 Corrective Actions

A2.7.1 NUREG-1927 Program Element

NUREG-1927 Program Element 7, Corrective Actions, (Reference A5.1) states "*Corrective actions, including root cause determination and prevention of* recurrence, should be timely."

A2.7.2 PINGP Program Element

Northern States Power Company – Minnesota (NSPM) has a single Corrective Action Program that is applied regardless of the safety classification of the structure or component. The Corrective Action Program requirements are established in accordance with the requirements of the NSPM Quality Assurance Topical Report and 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The Corrective Action Program procedures require the initiation of an Action Request for actual or potential problems including failures, malfunctions, discrepancies, deviations, defective material and equipment, nonconformances, and administrative control discrepancies, to ensure that conditions adverse to quality, operability, functionality, and reportability issues are promptly identified, evaluated if necessary, and corrected as appropriate. Guidance on establishing priority and timely resolution of issues is contained within the Corrective Action Program procedure.

All corrective actions for deviating conditions that are adverse to quality are performed in accordance with the requirements of the Quality Assurance Program which complies with the requirements of 10 CFR 50, Appendix B. Any resultant maintenance, repair/replacement activities, or special handling requirements are performed in accordance with approved procedures. Corrective actions provide reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable. Where evaluations are performed without repair or replacement, engineering analysis reasonably assures that the intended function is maintained consistent with the current licensing basis. If the deviating condition is assessed to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude recurrence. Corrective actions identify recurring discrepancies and initiate additional corrective actions including root cause analysis to preclude recurrence.

Degraded conditions identified by the AMP inspections will be entered into the Corrective Action Program. Actions required to resolve inspection findings will be tracked to completion and trended within the Corrective Action Program.

A2.7.3 Comparison to NUREG-1927 Program Element
This PINGP program element is consistent with NUREG-1927, Element 7, Corrective Actions.

A2.8 Confirmation Process

A2.8.1 NUREG-1927 Program Element
NUREG-1927 Program Element 8, Confirmation Process, (Reference A5.1) states "The confirmation process should ensure that preventive actions are adequate and appropriate corrective actions have been completed and are effective."

A2.8.2 PINGP Program Element
The confirmation process is part of the NSPM Corrective Action Program and ensures that the corrective actions taken are adequate and appropriate, have been completed, and are effective. The focus of the confirmation process is on the follow-up actions that must be taken to verify effective implementation of corrective actions. The measure of effectiveness is in terms of correcting the adverse condition and precluding repetition of significant conditions adverse to quality. Procedures include provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause evaluations and prevention of recurrence where appropriate. These procedures provide for tracking, coordinating, monitoring, reviewing, verifying, validating, and approving corrective actions, to ensure effective corrective actions are taken.

The Corrective Action Program is also monitored for potentially adverse trends. The existence of an adverse trend due to recurring or repetitive adverse conditions will result in the initiation of an Action Request. The AMP or AMAs will also uncover unsatisfactory conditions resulting from ineffective corrective action.

A2.8.3 Comparison to NUREG-1927 Program Element
This PINGP program element is consistent with NUREG-1927, Element 8, Confirmation Process.

A2.9 Administrative Controls

A2.9.1 NUREG-1927 Program Element 9, Administrative Controls

NUREG-1927 Program Element 9, Administrative Controls, (Reference A5.1) states "Administrative controls should provide a formal review and approval process."

A2.9.2 PINGP Program Element

The NSPM Quality Assurance Program, associated formal review and approval processes, and administrative controls applicable to the AMP and Aging Management Activities, are implemented in accordance with the requirements of the NSPM Quality Assurance Topical Report and 10 CFR Part 50, Appendix B. The administrative controls that govern AMAs at PINGP are established in accordance with the PINGP Administrative Control Program and associated Fleet Procedures.

A2.9.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 9, Administrative Controls.

A2.10 Operating Experience

A2.10.1 NUREG-1927 Program Element

NUREG-1927 Program Element 10, Operating Experience, (Reference A5.1) states "Operating experience involving the AMP, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of aging will be adequately managed so that the structure and component intended functions will be maintained during the period of extended operation."

A2.10.2 PINGP Program Element

The ISFSI Inspection and Monitoring Activities Program has been effective in maintaining plant structures and components. A review of ISFSI operating history provides evidence that any potential aging effects have been identified, evaluated, and managed effectively, ensuring that structures and components remain capable of performing their intended functions. It can be concluded that there is reasonable assurance that these structures and components will continue to perform their intended functions during the period of extended operation.

<u>Routine Inspections</u>

The Prairie Island ISFSI has been in operation since May of 1995. Visual inspections of the in-service casks, concrete pads, and earthen berm are performed in accordance with existing PINGP procedures. Inspections of the in-service casks to date have identified only minor cases of coating degradation which are corrected by touching-up of the existing coating material. There has been no evidence of loss of material under the degraded coating. No anomalies have been identified for the earthen berm. No anomalies of consequence have been identified for the concrete pads.

Lead Cask Inspections

Additional visual inspections of the normally inaccessible external surfaces of the in-service casks were evaluated for performance during a baseline inspection. The need and scope for these inspections were based on the above OE issues discussed below and the guidance in NUREG-1927 Appendix E – Component-Specific Aging Management (Reference A5.1). NUREG-1927 Appendix E states "A staff-accepted way to verify canister condition at an independent spent fuel storage installation is by remote visual inspection of one or more canisters ("lead canisters"). A lead canister is selected on the basis of longest time in service, or hottest thermal load, and/or other parameters that contribute to degradation" (Reference A5.1). Following this guidance, NSPM selected Cask TN-40 01 (Cask 01) as the lead cask because it had the longest in-service time of 16 years. The baseline inspection included inspection of the bottom of the cask (cask area in direct contact with the concrete pad) and underneath the protective cover. Due to industry OE concerning the area underneath the protective covers (water intrusion and bolt torque issues), this portion of the baseline inspection was expanded to include the inspection of a second in-service cask. As a result, Cask TN-40 13 (Cask 13) was also selected for inspection of the area underneath the protective cover.

The results of the cask bottom inspection revealed that approximately 25% of the protective coating on the bottom of Cask 01 exhibited loss of adhesion (peeling). In areas with loss of adhesion, the base metal did not exhibit any degradation that would affect the cask's intended function. The majority of the base metal was clean, however some corrosion and corrosion product stains were observed, mainly in areas where the epoxy coating itself was cracking. In those areas, the base metal did not have observable loss of material (no depth). Additionally, the concrete under the cask exhibited no visual signs of degradation. Therefore, the evaluation in the Corrective Action Program concluded that no corrective action was necessary.

With the protective cover removed, inspection of the area underneath the cover of Cask 01 was performed. During this inspection, no subcomponents within the scope of License Renewal exhibited any evidence of degradation. The observable area of the lid and lid bolt heads had no indication of corrosion. A coating of rust was found on the cask flange at the protective cover interface. This rust coating originated on the carbon steel protective cover, was deposited on the cask flange, and was easily removed. The removal of this coating revealed no degradation to the stainless steel overlay surface of the cask flange and no corrosion between the lid and flange in the main lid seal area. The neutron shield bolts were removed, inspected, and observed to have no indication of corrosion with the N-5000 lubricant still intact on the threads. The neutron shield protective coating exhibited no signs of corrosion.

The protective cover was found to have uniform corrosion on the flange sealing surface outside (external side) of the O-ring seal. There was minor corrosion around the protective cover bolt holes where the bolt heads had broken the epoxy coating due to friction upon installation. The underside of the protective cover dome had no signs of degradation. The protective cover O-ring seal remained in acceptable condition with the exterior coating on the protective cover exhibiting checking on approximately 15% to 20% of the surface area.

Inspection of the area underneath the protective cover of Cask 13 was also performed with the protective cover removed. During this inspection, no subcomponents within the scope of License Renewal exhibited any evidence of degradation. The observable area of the lid and lid bolt heads had no indication of corrosion. The stainless steel flange overlay had only small stains where rust from the protective cover was deposited. The stains were removed and there was no indication of corrosion on the observable area of the flange and no corrosion was observed between the lid and flange near the main lid seal area. The neutron shield bolts were removed and inspected with no indication of corrosion and also had the N-5000 lubricant still intact on the threads. The neutron shield had two rust stains on the protective coating directly below the access cover with one stain approximately six inches in diameter and the other approximately two inches in diameter. The protective cover was found to have corrosion on the interior. The corrosion appears to have started at the interior face of the access cover opening where the stainless steel overpressure system piping welded to the access plate made contact with the protective cover. The protective cover dome had evidence of corrosion in the area where it connected to the access plate. The access plate itself had corrosion on the entire interior surface excluding the area that was covered by the rubber gasket. However, none of these subcomponents exhibiting corrosion are within the scope of License Renewal.

The inspections performed for Cask 01 and Cask 13 with the protective covers removed were performed on those subcomponents not normally accessible and included the protective covers, access plates, neutron shields, neutron shield bolts, lid bolts, etc. Additionally, the torque values for the lid bolts were verified to address the industry OE discussed above. No degradation of any of the subcomponents within the scope of License Renewal were identified that would affect their intended function(s). Furthermore, the as-found lid bolt torque value met the original 1995 as-left torque value of 930 ft-lbs.

During the baseline inspections of Casks 01 and 13, the accessible areas of the casks were also inspected. The upper trunnions of Cask 01 exhibited some corrosion product stains on the top of the trunnion shaft. There was no indication of corrosion on all other areas inspected on Cask 01 and Cask 13.

A work order to repair the epoxy coating on Cask 01 upper trunnions, and additional cask coating surfaces was initiated. Based on the results of the above inspections, it was concluded that these structures and components remain capable of performing their intended functions throughout the period of extended operation.

Cask Interseal Pressure Monitoring
Trending of the in-service cask interseal helium pressures has revealed no issues with the seals or age related issues with the pressure monitoring system leak-tight integrity on any of the 29 in-service casks. However, there have been instances during extreme cold weather conditions when a low pressure alarm was received requiring the pressure monitoring system to be charged and the fittings tightened. These event-driven issues were a function of extreme temperature conditions and not age-related.

Radiation Surveys
NSPM performs periodic radiation surveys of the in-service casks. Trending of these surveys results shows no evidence that the shielding is degrading. Figure A2.10-1 provides the gamma dose rates at two meters from the three casks that have been in-service the longest. Figure A2.10-2 provides the neutron dose rates at two meters from the three casks that have been in-service the longest.

Corrective Action Program
A review of items in the Corrective Action Program and the "Structures Monitoring Program Quarterly Inspection Reports" was also performed. Minor maintenance items such as cleaning and painting of pull-box covers and transmitter base plates were identified for components which are not within the scope of License Renewal.

As previously discussed, inspections of the in-service casks identified minor cases of coating degradation. Touch-up of the coating material corrected this condition with no evidence of loss of material on the casks. There have been instances during extreme cold weather conditions, as previously discussed (last instance was January of 2010 with an ambient temperature of -20^{0}F), when a low pressure alarm was received on casks requiring the pressure monitoring system to be charged and the fittings tightened. These event-driven issues were a function of extreme temperature conditions and not age-related.

No other issues or findings were noted in the Corrective Action Program database relative to aging of the in-scope ISFSI structures and components.

Program Health Status Reports
The ISFSI Inspection and Monitoring Activities Program is a subset of the PINGP Structures Monitoring Program. A summary of the last two Structures Monitoring Program Inspection Reports that included ISFSI structures is provided below:

- Structures Monitoring Program, Quarterly Inspection Report Third Quarter 2001

  Shallow surface spalls were identified around the base plates of three of the monitor stands adjacent to the casks. These spalls were considered acceptable. It was recommended that monitoring be continued at the specified seven-year frequency.

Four shrinkage cracks were identified on the floor slab of the Equipment Storage Building (not within the scope of License Renewal). This condition was considered acceptable. It was recommended that monitoring be continued at the seven-year frequency. Abraded coatings and surface corrosion were also identified on the pull box frames, door frames and sills in addition to loose or missing nuts and washers at the building columns. A Work Request was initiated to correct these deficient conditions. Corrosion of the interseal pressure transmitter frames and supports, including pull box frames, pull box covers, ground clamps, and Environmental Monitor supports (not within the scope of License Renewal) was also identified. A Work Request was initiated to correct these deficient conditions.

- Structures Monitoring Program, Quarterly Inspection Report Second Quarter 2008

  A Work Request was initiated to excavate to sound material the spalled concrete identified in the 3Q01 inspection and then patch the area to prevent further degradation for the shallow surface spalls identified around the base plates of three of the monitor stands.

  Significant holes were found along the foundation of the Alarm Monitoring Building (not within the scope of License Renewal). A Work Request was initiated to correct these deficient conditions by filling these holes and compacting the affected soil.

  Although the inspections identified above noted minor issues that did not pose any challenges and were adequately monitored by existing PINGP procedures which required a seven-year inspection frequency, this frequency was subsequently changed to a five-year frequency in February of 2011. This change was performed to be consistent with the requirements of the Structures Monitoring Program and the Maintenance Rule and, as a result, increased the ISFSI inspection frequency from a seven year interval to a five year interval.

<u>System Health and Status Reports</u>
The actual status of the ISFSI is evaluated under the Prairie Island ISFSI System Monitoring and Reporting Tool, "Health and Status Report." As of July 2011, overall ISFSI performance was "Green" based on no operability concerns, no open corrective work orders and no overdue preventive maintenance work orders. There have not been any Licensee Event Reports associated with the Prairie Island ISFSI.

No issues or findings were noted relative to the ISFSI structures and components.

<u>NRC Inspection Reports</u>
NRC inspection reports issued during the period of February 28, 2005 through April 29, 2010 were reviewed for the ISFSI site.

No issues or findings were noted relative to the ISFSI structures and components.

Industry OE

EPRI Report 1002882, "Dry Cask Storage Characterization Project - Final Report" (Reference A5.2), indicated the possibility of corrosion of the stainless steel fasteners for the rear breech plate which is located on the bottom of the CASTOR V/21 casks. Although NSPM does not utilize the CASTOR V/21 cask design, the concern was addressed as part of the baseline inspections discussed earlier in this section.

Virginia Electric and Power Company (Dominion) identified in the Surry ISFSI License Renewal Application (Reference A5.3) corrosion of their Transnuclear TN-32 lid bolts and outer metallic lid seals. Dominion stated that the corrosion of the lid bolts and outer metallic seal was the result of external water intrusion in the vicinity of the bolts and seal. It was determined that the Conax connector seal for the electrical connector in the cask protective cover was leaking due to improper installation of the connector. This degradation was a function of improper installation and not age-related. However, as a result of this experience, the vendor, Transnuclear (TN), issued an Information Bulletin (Reference A5.4) on these findings. The TN Information Bulletin informed all TN storage cask users of two issues that occurred at Dominion's Surry Power Station involving the TN-32 Storage Casks.

The first issue concerned the Helicoflex metallic seals utilized in the cask lid. Beginning in December 1999, five low-pressure alarms occurred over a six month period. These alarms were investigated and attributed to loose or leaking pressure switches. The installed Ashcroft pressure switches were replaced with Wasco pressure switches for both Surry and North Anna. Future TN casks use the new Wasco pressure switch. This issue was a design and installation issue and not age-related.

As a result of this issue, Dominion brought five casks back to the fuel pool area from the ISFSI for lid removal. The lid seals were removed and examined both visually and microscopically and revealed that the outer metallic seal contained small thru-wall holes caused by corrosion of the outer aluminum seal jacket. No corrosion was observed on the inner containment seal nor was any leakage detected past the inner seal; therefore, containment of the cask was never compromised. Corrosion was also observed on two of the five casks' lid edges where metallic spray and/or paint did not fully cover the surface. The casks showed evidence of water intrusion and/or high humidity inside the protective cover. In some cases, residue from standing/pooling water under the lid was observed. In the presence of water, the galvanic couple between aluminum and stainless steel is sufficient to cause corrosion. It had been concluded that theTN-32 design with aluminum metallic seals is sensitive to galvanic corrosion occurring if standing water or humid conditions near saturation are experienced under the protective cover.

The TN-32 casks at Surry were a unique design in terms of the protective cover and the overpressure (OP) system. The OP system utilized pressure switches attached directly to the OP tank with electrical wires emerging from the top of the protective cover through a Conax fitting. Water entered the protective cover through the Conax fitting at the apex of the dome, due to the Conax connectors not being properly installed on the casks. This issue was a design and installation issue and not age-related. A new protective cover and OP system was retrofitted to the existing casks consistent with other TN metal storage casks with tubing to the OP tank through a bolted and gasketed cover plate located on the protective cover (similar to the TN-40 and TN-40HT cask design). Thus, the potential leak path through the Conax connection at the top of the cover was eliminated.

TN stated that the Surry site location may have exacerbated the corrosion issue due to the brackish environment and the presence of chlorides in the water from precipitation or humidity which would accelerate a galvanic reaction. Although the Prairie Island ISFSI site is not located in an area that is exposed to a brackish environment, NSPM has conservatively included loss of material due to galvanic corrosion for aluminum as a potential aging mechanism in the ISFSI Atmosphere/Weather environment.

The second issue discussed in the TN bulletin was identified upon returning the Surry casks to the fuel pool area to remove the lid. It was discovered that some lid bolts on three casks did not have the original torque value applied prior to placement of the casks at the ISFSI. Lid bolts could be removed by hand on two casks. However, in all cases there was no evidence that the lid metallic O-rings lost their seal due to the reduced bolt torque. A majority of the hand-tight bolts were identified at locations that are tightened early in the "star" torquing sequence. Evaluations by TN confirmed that the lid seals would remain compressed and containment would be maintained.

Consensus was that a change in bolt torquing sequence methodology should be taken to mitigate the possibility of thermal expansion causing the bolting problems. TN stated that it was common practice for the final torque on the lid bolts to be applied after thermal equilibrium of the cask was obtained. This would translate into using an intermediate lid bolt torque value during the draining and vacuum drying operations. A minimum of two passes should be utilized in the star pattern and additional passes made as necessary until there is no further movement of the bolts. Additionally, lubricant should be applied to the bolts and special attention paid to the calibration of the bolt torquing equipment. TN recommended the use of Neolube or Loc-Tite N-5000 as the lubricant. Additional information on this subject may be found in the TN Information Bulletin.

Similar operating experience was identified with the TN-68 casks utilized at the Peach Bottom Atomic Power Station (Reference A5.5). This information was evaluated in NSPM's review of the operating experience.

In response to the bolting issues, the vendor recommended a bolt torquing sequence methodology and application of Loc-Tite N-5000, as stated above.

These recommendations have been addressed at PINGP and are incorporated in the applicable existing PINGP maintenance procedures.

Precedent License Renewal Applications OE
A review of precedent ISFSI license renewal applications was performed to evaluate any relevant operating experience. ISFSIs included in this review were Calvert Cliffs Nuclear Power Plant, H. B. Robinson Steam Electric Station, and Surry Power Station. The results of these reviews concluded that the Prairie Island ISFSI Inspection and Monitoring Activities Program is effective in monitoring and detecting degradation and taking effective corrective actions as needed to preclude loss of intended function.

Conclusion
The OE, reviews, and monitoring described above confirm that any potential aging effects will be identified, evaluated, and managed effectively, ensuring that these structures and components remain capable of performing their intended functions.

A2.10.3 Comparison to NUREG-1927 Program Element

This PINGP program element is consistent with NUREG-1927, Element 10, Operating Experience.

A3.0 HIGH BURNUP FUEL MONITORING PROGRAM

The Prairie Island ISFSI provides for long-term dry fuel interim storage for high burnup spent fuel assemblies, i.e., fuel assemblies with discharge burnups greater than 45 GWD/MTU, until such time that the spent fuel assemblies may be shipped off-site for final disposition. The cask system presently utilized at the Prairie Island ISFSI for the storage of high burnup spent fuel is the Transnuclear TN-40HT which has a 40 fuel assembly capacity and is designed for outdoor storage.

The Aging Management Review of the high burnup fuel spent fuel assemblies in a dry inert environment did not identify any aging effects/mechanisms that could lead to a loss of intended function. However, it is recognized that there has been relatively little operating experience, to date, with dry storage of high burnup fuel. Reference A5.8 provides a listing of a significant amount of scientific analysis examining the long term performance of high burnup spent fuel. These analyses provide a sound foundation for the technical basis that long term storage of high burnup fuel, i.e., greater than 20 years, may be performed safely and in compliance with regulations. However, it is also recognized that scientific analysis is not a complete substitute for confirmatory operating experience. Therefore, the purpose of the High Burnup Fuel Monitoring Program is to confirm that the high burnup fuel assemblies' intended function(s) are maintained during the period of extended operations.

A description of the High Burnup Fuel Monitoring Program is provided below. Although the program is a confirmatory program, the description below uses each attribute of an effective AMP as described in NUREG-1927 for the renewal of a site-specific Part 72 license to the extent possible.

A3.1 AMP Element 1: Scope of the Program

Fuel Stored in a TN-40HT Cask is limited to an assembly average burnup of 60 GWd/MTU (note that the nominal burnup value is lower to account for uncertainties). The cladding materials for the Prairie Island high burnup fuel are Zircaloy-4 and Zirlo™, and the fuel is stored in a dry helium environment. High burnup fuel was first placed into dry storage in a TN-40 HT cask on April 4, 2013.

The High Burnup Fuel Monitoring Program relies upon the joint Electric Power Research Institute (EPRI) and Department of Energy (DOE) "High Burnup Dry Storage Cask Research and Development Project" (HDRP) (Reference A5.9) or an alternative program meeting the guidance in Interim Staff Guidance (ISG) 24, Reference A5.10, as a surrogate program to monitor the condition of high burnup spent fuel assemblies in dry storage.

The HDRP is a program designed to collect data from a spent nuclear fuel storage system containing high burnup fuel in a dry helium environment. The program entails loading and storing a TN-32 bolted lid cask (the Research Project Cask) at Dominion Virginia Power's North Anna Power Station with intact high burnup spent nuclear fuel (with nominal burnups ranging between 53 GWd/MTU and 58 GWd/MTU). The fuel assemblies to be used in the program include four different kinds of cladding (Zircaloy-4, low-tin Zircaloy-4, Zirlo™, and M5™). The Research Project Cask is to be licensed to the temperature limits contained in ISG-11, Reference A5.7, and loaded such that the fuel cladding temperature is as close to the limit as practicable. Aging effects will be determined for material/environment combinations per ISG-24 Rev. 0 or the "High Burnup Dry Storage Cask Research and Development Project" (HDRP).

A3.2 AMP Element 2: Preventive Actions

The High Burnup Fuel Monitoring Program consists of condition monitoring to confirm there is no degradation of a high burnup fuel assembly that would result in a loss of intended function(s). Other than the initial design limits placed on loading operations, no preventive or mitigating attributes are associated with these activities.

During the initial loading operations of the TN-40HT casks, the design and ISFSI Technical Specifications (TS) require that the fuel be stored in a dry inert environment. TS 3.1.1, "Cask Cavity Vacuum Drying," demonstrates that the cask cavity is dry by maintaining a cavity absolute pressure less than or equal to 10 mbar for a 30 minute period with the cask isolated from the vacuum pump. TS 3.1.2, "Cask Helium Backfill Pressure," requires that the cask then be backfilled with helium. These two TS requirements ensure that the high burnup fuel is stored in an inert environment thus preventing cladding degradation due to oxidation mechanisms. TS 3.1.2 also requires that the helium environment be established within 34 hours of commencing cask draining. This time requirement ensures that the peak cladding temperature remains below 752°F (i.e., the temperature specified in ISG-11), thus mitigating degradation due to cladding creep.

A3.3 AMP Element 3: Parameters Monitored/ Inspected

Either the surveillance demonstration program as described in the HDRP or an alternative program should meet the guidance of ISG-24, Rev. 0.

A3.4 AMP Element 4: Detection of Aging Effects

Either the surveillance demonstration program as described in the HDRP or an alternative program should meet the guidance of ISG-24, Rev. 0.

A3.5 AMP Element 5: Monitoring & Trending

As information/data from a fuel performance surveillance demonstration program becomes available, NSPM will monitor, evaluate, and trend the information via its Operating Experience Program and/or the Corrective Action Program to determine what actions should be taken to manage fuel and cladding performance, if any.

Similarly, NSPM will use its Operating Experience Program and/or Corrective Action Program to determine what actions should be taken if it receives information/ data from other sources than the demonstration program on fuel performance.

Formal evaluations of the aggregate feedback from the HDRP and other sources of information will be performed at the specific points in time during the period of extended operation delineated in the table below. These evaluations will include an assessment of the continued ability of the high burnup fuel assemblies to continue to perform their intended function(s) at each point.

Toll Gate	Year *	Assessment
1	2028	Evaluate information obtained from the HDRP loading and initial period of storage along with other available sources of information. If the HDRP NDE (i.e., cask gas sampling, temperature data) has not been obtained at this point and no other information is available then NSPM has to provide evidence to the NRC that no more than 1% of the HBF has failed.

Toll Gate	Year *	Assessment
2.	2038	Evaluate, if available, information obtained from the destructive (DE) and non-destructive (NDE) examination of the fuel placed into storage in the HDRP along with other available sources of information. If the aggregate of this information confirms the ability of the high burnup fuel assemblies to continue to perform intended function(s) for the remainder of the period of extended operations, subsequent assessments may be cancelled. If the HDRP DE of the fuel has not been examined at this point and no other information is available then NSPM has to provide evidence to the NRC by opening a cask or single effects surrogate experiments that the fuel performance acceptance criteria 1-4 in element 6 continue to be met.
3	2048	Evaluate any other new information.

* Assessments are due by April 4 of the year identified in the table

The above assessments are not, by definition, stopping points. No particular action, unless noted in this AMP, other than performing an assessment is required to continue cask operation. To proceed, an assessment of aggregated available operating experience (both domestic and international), including data from monitoring and inspection programs, NRC-generated communications, and other information will be performed. The evaluation will include an assessment of the ability of the high burnup fuel assemblies to continue to perform their intended function(s).

A3.6 AMP Element 6: Acceptance Criteria

- The HDRP or any other demonstration used to provide fuel performance data should meet the acceptance criteria guidance of ISG-24 Rev 0.
- If any of the following fuel performance criteria are exceeded in the HDRP or alternative program, a corrective action is required[1]:
    1. Cladding Creep: total creep strain extrapolated to the total approved storage duration based on the best fit to the data, accounting for initial condition uncertainty shall be less than 1%
    2. Hydrogen – maximum hydrogen content of the cover gas over the approved storage period shall be extrapolated from the gas measurements to be less than 5%
    3. Drying – The moisture content in the cask , accounting for measurement uncertainty, shall indicate no greater than one liter of residual water after the drying process is complete

[1] While it is not a fuel performance criteria, the spatial distribution and time history of the temperature must be known to evaluate the relationship between the performance of the rods in the HDRP and the HBF rod behavior expected in the TN-40HT cask.

4. Fuel rod breach – fission gas analysis shall not indicate more than 1% of the fuel rod cladding breaches

A3.7 AMP Element 7: Corrective Actions

The NSPM Corrective Action Program commensurate with 10 CFR 50 Appendix B will be followed.

In addition, at each of the assessments in AMP Section 5, the impact of the aggregate feedback will be assessed and actions taken when warranted. These evaluations will address any lessons learned and take appropriate corrective actions, including:

- Perform repairs or replacements
- Modify this confirmatory program in a timely manner
- Adjust age-related degradation monitoring and inspection programs (e.g., scope, frequency)
- Actions to prevent reoccurrence
- An evaluation of the DCSS to perform it's safety and retrievability functions
- Evaluation of the effect of the corrective actions on this component to other safety components.

A3.8 AMP Element 8: Confirmation Process

The confirmation process is part of the NSPM Corrective Action Program and ensures that the corrective actions taken are adequate and appropriate, have been completed, and are effective. The focus of the confirmation process is on the follow-up actions that must be taken to verify effective implementation of corrective actions. The measure of effectiveness is in terms of correcting the adverse condition and precluding repetition of significant conditions adverse to quality. Procedures include provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause evaluations and prevention of recurrence where appropriate. These procedures provide for tracking, coordinating, monitoring, reviewing, verifying, validating, and approving corrective actions, to ensure effective corrective actions are taken.

A3.9 AMP Element 9: Administrative Controls

The NSPM Quality Assurance Program, associated formal review and approval processes, and administrative controls applicable to this program and Aging Management Activities, are implemented in accordance with the requirements of the NSPM Quality Assurance Topical Report and 10 CFR Part 50, Appendix B. The administrative controls that govern AMAs at PINGP are established in accordance with the PINGP Administrative Control Program and associated Fleet Procedures.

A3.10 AMP Element 10: Operating Experience

Surrogate surveillance demonstration programs with storage conditions and fuel types similar to those in the dry storage system that satisfies the ISG-24 acceptance criteria are a viable method to obtain operating experience. NSPM intends to rely on the information from the HDRP with similar types of HBU fuel.

The HDRP is viable as a surrogate surveillance program. Additional data/research to assess fuel performance from both domestic and international sources that are relevant to the fuel in the NSPM casks will also be used.

A4.0 Summary

The review of operating experience identified a number of incidents related to dry fuel storage. Although many of these were event-driven and most were not age-related, for those that did involve credible aging effects and mechanisms, evaluations were conducted to assess potential susceptibility. These evaluations indicated that the aging effects and mechanisms that were identified at the Prairie Island ISFSI are bounded by the Aging Management Reviews that were performed for those structures and components identified as within the scope of License Renewal.

Operating experience to date has not indicated any degradation that would affect the structures or component intended function(s). Inspections, monitoring, and surveillances continue to be conducted that would identify deficiencies. The Corrective Action Program is in place to track and correct deficiencies in a timely manner. Corrective actions have been effectively implemented when inspection and monitoring results have indicated degradation. Continued implementation of the ISFSI Inspection and Monitoring Activities Program and the High Burnup Fuel Monitoring Program provide reasonable assurance that the aging effects will be managed such that the intended functions will be maintained during the period of extended operation.

A5.0 References (Appendix A, Aging Management Program)

A5.1 NUREG-1927, *Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance*, March 2011.

A5.2 EPRI Report 1002882, *Dry Cask Storage Characterization Project, Final Report*, September 2002.

A5.3 Letter from D.A. Christian, Virginia Electric and Power Company to D.A. Cool (NRC), *Surry Independent Spent Fuel Storage Installation License Renewal Application*, dated April 29, 2002, ADAMS Accession Number ML021290068.

A5.4 Transnuclear Information Bulletin, April 2001.

A5.5 Letter from G. L. Stathes, Exelon Generation Company to Director Spent Fuel Project Office (NRC), *Submittal of Independent Spent Fuel Storage Installation (ISFSI) Cask Event Report*, dated December 01, 2010, ADAMS Accession Number ML110060275.

A5.6 American Concrete Institute, ACI 349.3R-96, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*, January 1996.

A5.7 NRC Interim Staff Guidance 11, *Cladding Considerations for the Transportation and Storage of Spent Fuel*, Revision 3, November 17, 2003.

A5.8 Letter from R. McCullum (NEI) to M. Lombard (NRC), dated March 22, 2013, *Industry Analysis and Confirmatory Information Gathering Program to Support the Long-Term Storage of High Burnup Fuel (HBF)*, (ADAMS Accession No. ML13084A045).

A5.9 High Burnup Dry Storage Cask Research and Development Project Final Test Plan, February 27, 2014, DOE Contract No.: DE-NE-0000593.

A5.10 NRC Interim Staff Guidance 24, *The Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity for Continued Storage of High Burnup Fuel Beyond 20 Years*, Revision 0, July 11, 2014.

TABLE A2.1-1
Managed Aging Effects

Material	Environment	Aging Effect	Aging Mechanism
Aluminum	Atmosphere/Weather	Loss of Material	Crevice Corrosion
Aluminum	Atmosphere/Weather	Loss of Material	Galvanic Corrosion
Aluminum	Atmosphere/Weather	Loss of Material	Pitting Corrosion
Carbon Steel	Atmosphere/Weather	Loss of Material	Crevice Corrosion
Carbon Steel	Atmosphere/Weather	Loss of Material	Galvanic Corrosion
Carbon Steel	Atmosphere/Weather	Loss of Material	General Corrosion
Carbon Steel	Atmosphere/Weather	Loss of Material	Pitting Corrosion
Polypropylene	Air/Gas	Cracking	Material property changes from radiation exposure
Borated Polyester	Air/Gas	Cracking5	Material property changes from radiation exposure
Stainless steel	Atmosphere/Weather	Loss of Material	Crevice Corrosion
Stainless steel	Atmosphere/Weather	Loss of Material	Pitting Corrosion
Concrete	Atmosphere/Weather	Change in Material Properties	Leaching of $Ca(OH)_2$
Concrete	Atmosphere/Weather	Cracking	Freeze-Thaw
Concrete	Atmosphere/Weather	Cracking	Reaction with Aggregates
Concrete	Atmosphere/Weather	Loss of Material	Freeze-Thaw
Concrete	Soil	Change in Material Properties	Leaching of $Ca(OH)_2$
Concrete	Soil	Cracking	Reaction with Aggregates
Concrete	Soil	Cracking	Settlement
Earthen Structures	Atmosphere/Weather	Change in Material Properties	Desiccation
Earthen Structures	Atmosphere/Weather	Loss of Form	Settlement
Earthen Structures	Atmosphere/Weather	Loss of Form	Frost Action
Earthen Structures	Atmosphere/Weather	Loss of Material	Erosion (Wind/Rain Impact)

Figure A2.10-1
Two Meter Gamma Dose Rates

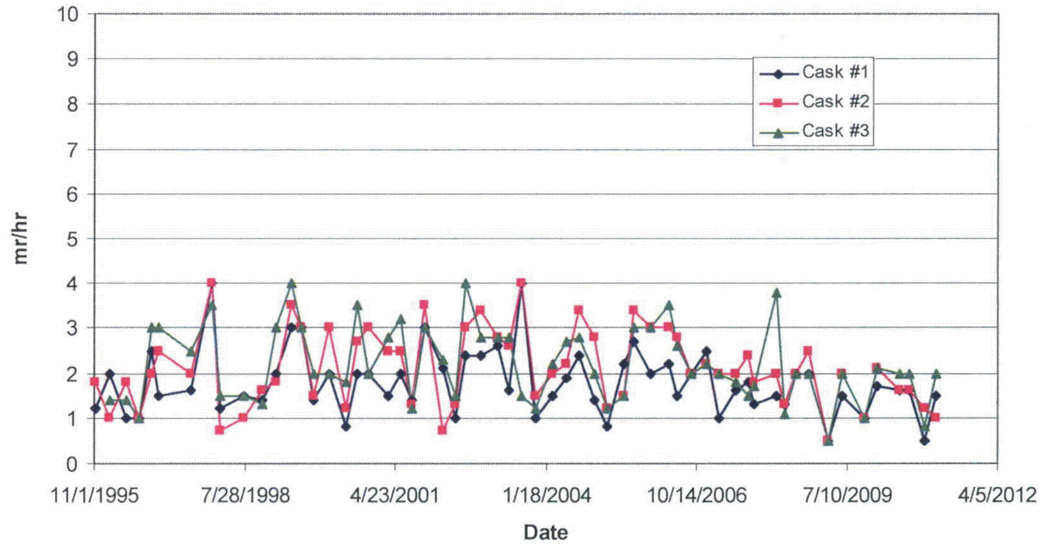
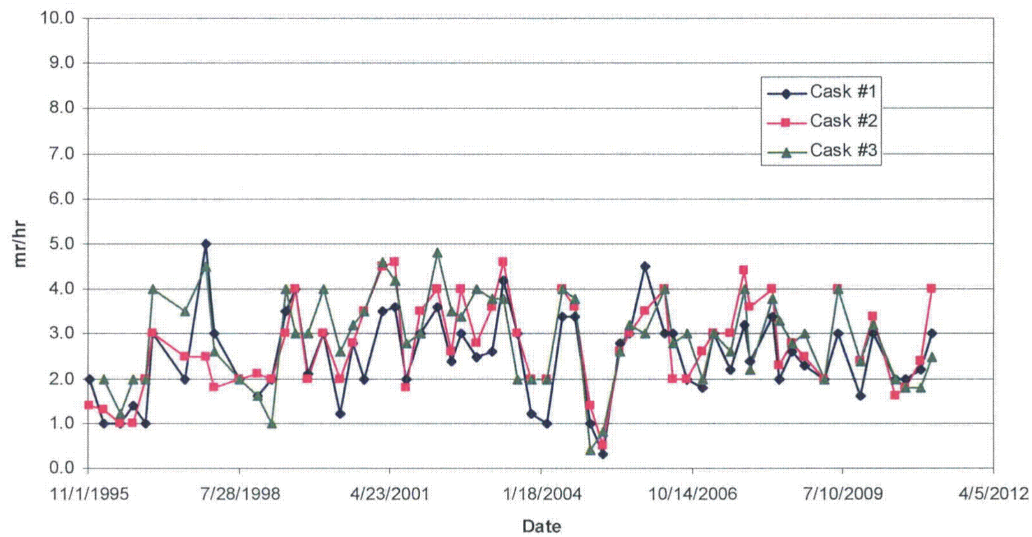


Figure A2.10-2
Two Meter Neutron Dose Rates



Attachment C

Sandia National Laboratory Report
SAND2008-1163
Technical Reference on Hydrogen Compatibility of Materials
Nonmetals: Polymers (code 8100)

9 pages follow

Technical Reference on Hydrogen Compatibility of Materials

Nonmetals:
Polymers (code 8100)

Prepared by:

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This report may be updated and revised periodically in response to the needs of the technical community; up-to-date versions can be requested from the editors at the address given below or downloaded at http://www.ca.sandia.gov/matlsTechRef/. The content of this report will also be incorporated into a Sandia National Laboratory report (SAND2008-1163); the most recent version can be obtained from the link above. The success of this reference depends upon feedback from the technical community; please forward your comments, suggestions, criticisms and relevant public-domain data to:

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Sandia National Laboratories

Technical Reference on Hydrogen Compatibility of Materials

Nonmetals:

Polymers (code 8100)

1. General

Polymers are a diverse category of materials characterized by chains of covalently-bonded atoms with repeating structural units. The materials can be processed in numerous ways with almost infinite variation. The properties of polymers are determined by a number of factors including crystallinity, density, chain orientation, cross-linking, purity, phase distribution, etc.

We are unaware of hydrogen compatibility studies for common polymer materials that might be expected in gaseous hydrogen service, thus we have eliminated the sections on mechanical properties and microstructural considerations. Gas permeation through polymer materials, however, has been extensively studied; therefore we provide a non-exhaustive summary of hydrogen transport data in common polymer materials.

Relatively large amounts of hydrogen are often soluble in polymer materials; therefore, exposure to high-pressure hydrogen may cause damage (blistering or swelling) of the polymer materials. This is manifest in high-pressure applications due to depressurization of a system (or rapid temperature changes) as hydrogen expands in free volume and at interfaces within the polymers. .

1.1 Composition and microstructure

Polymers are generally characterized by the composition and molecular structure of the material. Nomenclature often evolves from common usage and generally does not incorporate structural details. We use ASTM D1418 and D1600 for guidance on naming. Table 1.1.1 includes the abbreviations used in this document.

2. Permeability, Diffusivity and Solubility

Hydrogen transport in polymers has been extensively studied, particularly for high-vacuum systems. Similar to studies of metals, studies of the hydrogen permeation in polymers have generally been performed at low pressure. Permeability, diffusivity and solubility are often assumed to be independent of pressure for metals and data generated at low-pressure are extrapolated to describe high-pressure systems. This extrapolation implies that hydrogen transport and solubility properties are independent of concentration (i.e., Fickian diffusion). While concentration-dependent transport properties (non-Fickian diffusion) are often observed in polymers, we are unaware of any studies on polymers that suggest hydrogen transport and solubility are dependent on concentration. Thus, until studies show otherwise, we assume that hydrogen permeability, diffusivity and solubility in polymers are independent of pressure. Unlike metals, hydrogen transport in polymer materials is sufficiently rapid that the permeation rates can generally be measured at or near ambient temperature.

The permeability (Φ) is determined from Fick's first law for diffusion, and represents a steady-state property of the material (assuming diffusion is independent of pressure). It is defined in the same way as for metals, such that

$$\Phi = DS \tag{1}$$

where D is the diffusivity and S is the solubility. Hydrogen transport in polymers differs from metals in one important aspect: hydrogen does not dissociate prior to dissolution in the material, thus the concentration of hydrogen dissolved in the polymer (c) is proportional to the fugacity (f, which equals the pressure in the limit of an ideal gas):

$$S = {}^{c}\!/_{f} \tag{2}$$

while in metals c is proportional to $\sqrt{f}$. In materials where hydrogen does not dissociate, such as polymers, it should be clear from equations 1 and 2 that the units of permeability are

$$[\Phi] = [\mathit{diffusivity}]\frac{[\mathit{concentration}]}{[\mathit{pressure}]} = \frac{\text{m}^2}{\text{s}}\frac{\text{mole H}_2/\text{m}^3}{\text{MPa}} = \frac{\text{mole H}_2}{\text{m}\cdot\text{s}\cdot\text{MPa}} \tag{3}$$

Other forms of these units are, of course, possible and they can be a significant source of confusion. The units in equation 3 are commonly accepted for high-pressure hydrogen since they do not require definition of a reference state.

In tables 2.1 through 2.4, the hydrogen transport properties for a number of polymeric materials are summarized. A secondary resource [1] is used for these values and no effort was made to verify the primary references; the interested reader is also referred to Ref. [2], which contains a lists of primary sources by material. A selection of the hydrogen transport data from Ref. [1] is summarized here. Table 2.1 provides hydrogen transport properties for several common categories of plastics at approximately room temperature. Table 2.2 provides the transport properties for several commercial elastomers near room temperature, while Table 2.3 provides properties for a number of elastomers (rubbers) from a range of classes at room temperature and, when available, at elevated temperature.

Permeability, diffusivity and solubility follow a classic exponential form:

$$A = A_0 \exp\left(\frac{-E_A}{RT}\right) \tag{4}$$

where A_0 and E_A are material-dependent constants, R is are the universal gas constant (8.31447 J mol^{-1} K^{-1}) and T is temperature in Kelvin. Table 2.4 provides the constants from equation 4 that summarize the temperature dependence of these properties for several of the materials from the previous tables. The temperature dependence of hydrogen transport and solubility for the materials in Table 2.4 is plotted in Figure 2.1 (permeability), Figure 2.2 (diffusivity) and Figure 2.3 (solubility); these properties are linear when plotted on a log scale as a function of $1/T$ as shown in these figures.

3. References

1. S Pauly. Permeability and Diffusion Data. in: J Brandrup, EH Immergut and EA Grulke, editors. Polymer Handbook, fourth edition. New York: John Wiley and Sons (1999).
2. SA Stern, B Krishnakumar and SM Nadakatti. Permeability of Polymers to Gases and Vapors. in: JE Mark, editor. Physical Properties of Polymers Handbook. Woodbury NY: American Institute of Physics (1996).

Table 1.1.1. Standard abbreviations (from ASTM D1418 and D1600) for common polymeric materials reported in this document.

Abbreviation	Term (AISI/ASTM)
	Plastics
LDPE	Low density polyethylene plastics
PMMA	Poly(methyl methacrylate)
PP	Polypropylene
PS	Polystyrene
PTFE	Polytetrafluorethylene
PVC	Poly(vinyl chloride)
PVF	Poly(vinyl fluoride)
	Rubbers (Elastomeric Polymers)
CIIR	Chloro-isobutene-isoprene rubber
CR	Chloroprene rubber
IIR	Isobutene-isoprene rubber
NR	Natural rubber
NBR	Acrylonitrile-butadiene rubber
NIR	Acrylonitrile-isoprene rubber
SBR	Styrene-butadiene rubber
CSM	Chloro-sulfonyl-polyethylene
EPDM	Terpolymer of ethylene, propylene, and a diene
FKM	One type of fluoro rubber
VMQ	Silicone rubber (vinyl and methyl substituents)

Table 2.1. Hydrogen transport properties for some common plastics from Ref. [1]; abbreviations from Table 1.1.1. Values in parenthesis are not given in Ref. [1], and were calculated using equation 1.

Material	Temperature (K)	$\Phi \times 10^9$ $\left(\frac{\text{mol } H_2}{\text{m} \cdot \text{s} \cdot \text{MPa}}\right)$	$D \times 10^{12}$ $\left(\frac{\text{m}^2}{\text{s}}\right)$	S $\left(\frac{\text{mol } H_2}{\text{m}^3 \cdot \text{MPa}}\right)$
LDPE 0.914 g/cm^3	298	3.3	47.4	70.5
PP 0.907 g/cm^3, 50% crystallinity	293	13.8	210	(65.9)
PS biaxial structure	298	7.58	—	—
PMMA	308	1.24	—	—
PVC unplasticized	298	0.58	50	12
	300	0.80	48	(16.7)
PTFE	298	3.3	—	—
	298	3.23	14.7	220
PVF	308	0.18	—	—
PVF (Kynar)	308	0.180	33.6	(5.36)

Table 2.2. Hydrogen transport properties for some common commercial elastomers from Ref. [1].

Material	Temperature (K)	$\Phi \times 10^9$ $\left(\frac{\text{mol } H_2}{\text{m} \cdot \text{s} \cdot \text{MPa}}\right)$	$D \times 10^{12}$ $\left(\frac{\text{m}^2}{\text{s}}\right)$	S $\left(\frac{\text{mol } H_2}{\text{m}^3 \cdot \text{MPa}}\right)$
Hypalon 40 (CSM)	308	3.68	265	14.5
Kraton FG	308	22.4	1160	19.2
Viton GF (FKM)	308	7.32	345	21.0

Table 2.3. Hydrogen transport properties for some common elastomeric polymers from Ref. [1]; abbreviations from Table 1.1.1.

Material		Temperature (K)	$\Phi \times 10^9$ $\left(\frac{\text{mol } H_2}{\text{m} \cdot \text{s} \cdot \text{MPa}}\right)$	$D \times 10^{12}$ $\left(\frac{\text{m}^2}{\text{s}}\right)$	S $\left(\frac{\text{mol } H_2}{\text{m}^3 \cdot \text{MPa}}\right)$
Poly(butadiene) (BR)		298	14.1	960	14.5
CIIR	shore 70‡	293 353	0.857 11.6	—	—
CR	shore 42‡	293 353	2.23 27.3	—	—
Neoprene (CR) Neoprene G (CR)		308 298	9.24 4.55	933 380	9.9 12.8
Poly(isobutene-*co*-isoprene) (IIR) 98/2		298	2.42	152	15.8
Poly(butadiene-*co*-acrylonitrile) (NBR) 80/20 (Perbunan†) 61/39 (Hycar†)		 298 298	 8.43 2.39	 643 243	 13.2 9.7
NBR	shore 70‡	293 353	1.45 12.1	—	—
	shore 60‡	293 353	1.67 21.1	—	—
	shore 50‡	293 353	2.04 17.1	—	—
Poly(isoprene-*co*- acrylonitrile) (NIR) 74/26		298	2.49	247	10.1
NR	shore 66‡	293 353	4.09 38.4	—	—
SBR	shore 52‡	293 353	2.98 21.7	—	—
CSM	shore 70‡	293 353	1.31 9.28	—	—
EPDM	shore 68‡	293 353	6.96 38.4	—	—
FKM	shore 70‡	293 353	1.51 18.6	—	—
VMQ	shore 50‡	293 353	100 260	—	—

† trade names
‡ shore durometer (hardness)

Table 2.4. Relationships for temperature dependence of hydrogen transport properties for some common polymers from Ref. [1]; abbreviations from Table 1.1.1. Values in parenthesis are not given in Ref. [1], and were calculated using equations 1 and 4 with values reported in the other tables.

Material	Temperature	Permeability $\Phi = \Phi_o \exp(-H_\Phi / RT)$		Diffusivity $D = D_o \exp(-H_D / RT)$		Solubility, $S = \Phi/D$ $S = S_o \exp(-\Delta H_s / RT)$	
	Range (K)	Φ_o x 10^3 $\left(\frac{\text{mol } H_2}{\text{m} \cdot \text{s} \cdot \text{MPa}}\right)$	H_Φ $\left(\frac{\text{kJ}}{\text{mol}}\right)$	D_o x 10^6 $\left(\frac{\text{m}^2}{\text{s}}\right)$	H_D $\left(\frac{\text{kJ}}{\text{mol}}\right)$	S_o $\left(\frac{\text{mol } H_2}{\text{m}^3 \cdot \text{MPa}}\right)$	ΔH_s $\left(\frac{\text{kJ}}{\text{mol}}\right)$
PP 0.907 g/cm^3, 50% crystallinity	293 - 343	99.9	38.5	—	—	—	—
PVC unplasticized	298 - 353	0.651	34.5	(55.7)	34.5	(11.7)	0
PTFE	293 - 403	0.0185	21.4	—	—	—	—
Poly(butadiene) (BR)	298 - 323	0.959	27.6	(5.20)	21.3	(185)	6.3
Neoprene G (CR)	288 - 323	3.93	33.9	(26.2)	27.6	(162)	6.3
Poly(isobutene-*co*-isoprene) (IIR) 98/2	298 - 323	5.71	36.4	(133)	33.9	(43.4)	2.5
Poly(butadiene-*co*-acrylonitrile) (NBR) 80/20 (Perbunan†) 61/39 (Hycar†)	298 - 323	1.57 6.65	30.1 36.8	(23.2) (91.1)	26.0 31.8	(69.1) (72.8)	4.1 5.0
Poly(isoprene-*co*- acrylonitrile) (NIR) 74/26	298 - 323	11.7	38.1	(67.1)	31.0	(178)	7.1

† trade names

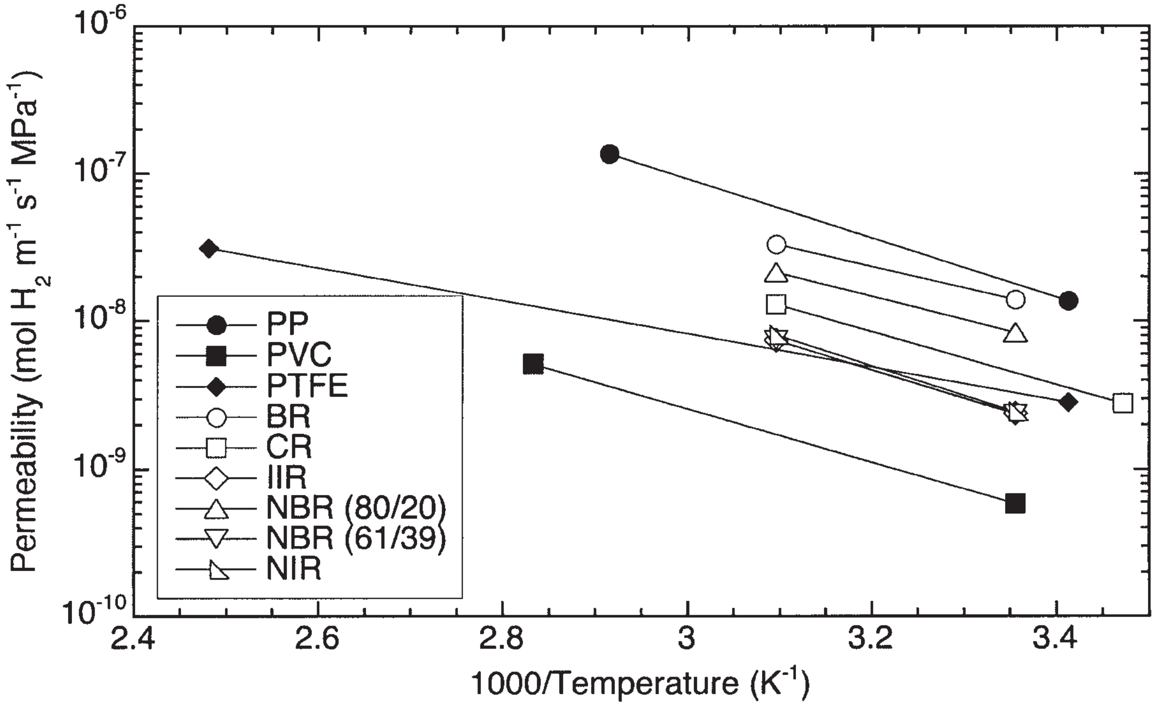

Figure 2.1. Permeability relationships (from Table 2.4) for several polymers.

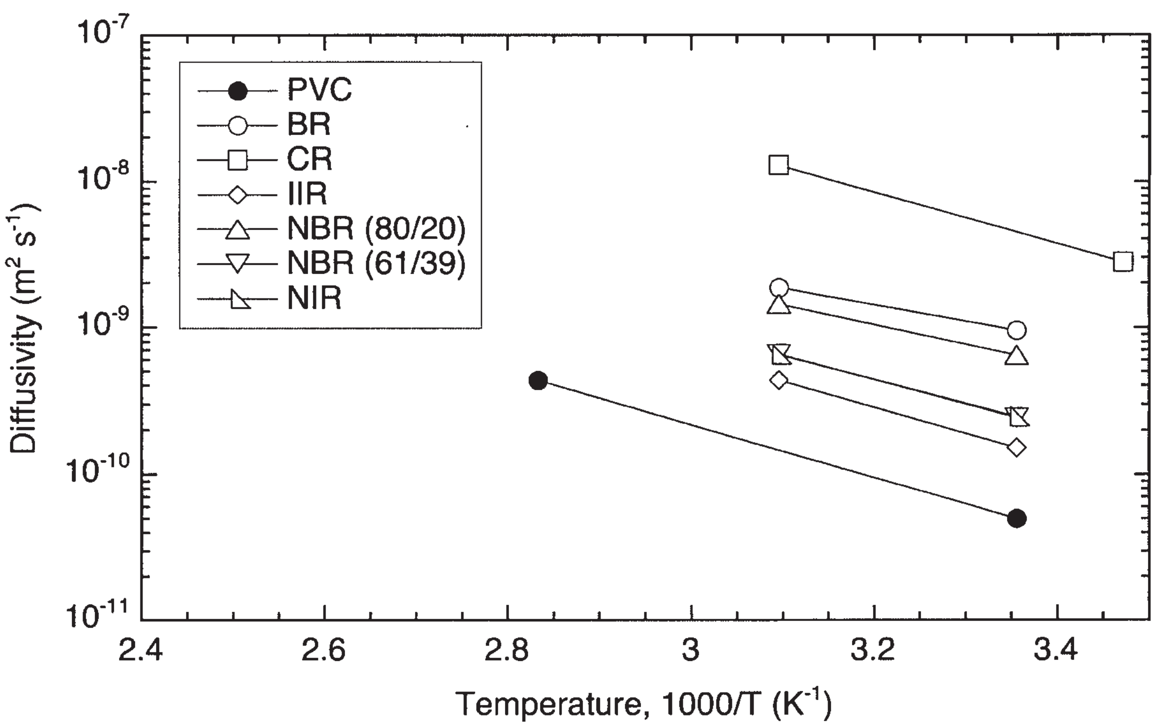

Figure 2.2. Diffusivity relationships (from Table 2.4) for several polymers.

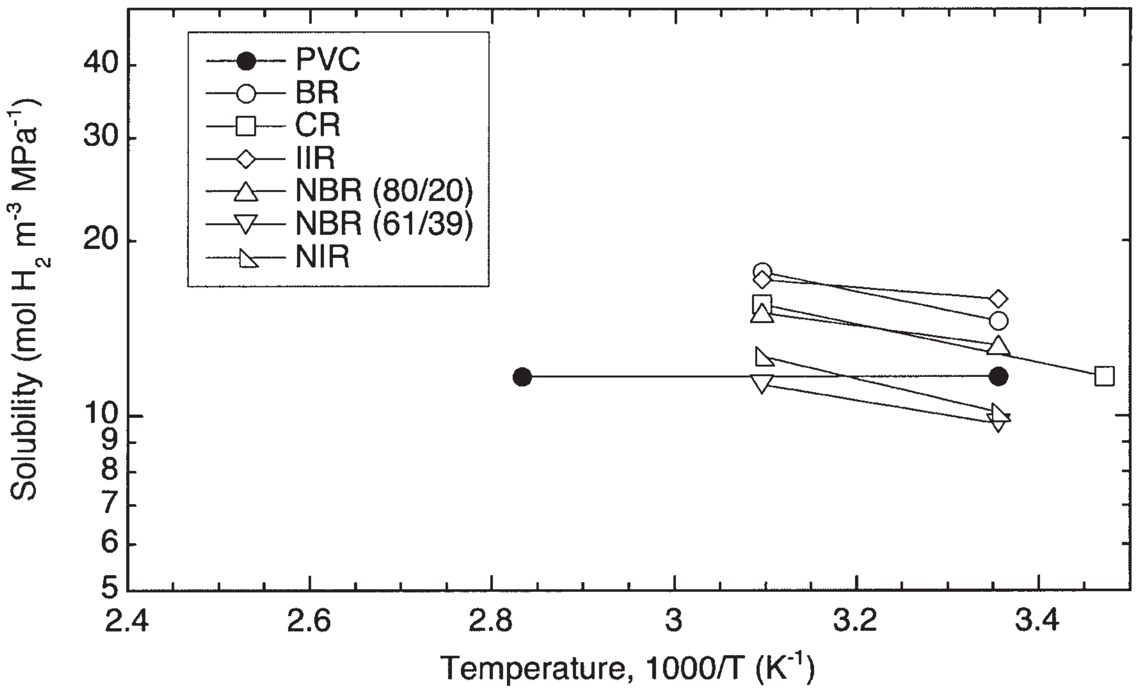

Figure 2.3. Solubility relationships (from Table 2.4) for several polymers.

Attachment D

Revision to LRA
Section 3.2.3, "Aging Effects Requiring Management,"
Including Table 3.2-1, "AMR Results for Casks"

5 pages follow

3.2.3 Aging Effects Requiring Management

This section describes the aging effects/mechanisms that could, if left unmanaged, cause degradation of cask subcomponents and result in loss of intended function(s) during the period of extended operation. The AMR results for individual cask subcomponents are reflected in Table 3.2-1. Based on the material and environment combinations, and consideration of the conditions during the period of extended operation, the following aging effects and associated mechanism(s) were determined to require management:

- Loss of Material due to crevice corrosion – External aluminum, carbon steel and stainless steel surfaces of the cask,
- Loss of Material due to galvanic corrosion – External Carbon steel and aluminum surfaces of the cask that are in contact with stainless steel,
- Loss of Material due to general corrosion – External carbon steel surfaces of the cask, and
- Loss of Material due to pitting corrosion – External aluminum, carbon steel and stainless steel surfaces of the cask.

While the Aging Management Review of the polymeric neutron shielding material did not identify any aging effects that could lead to a loss of intended function, the following aging effect and associated mechanism is conservatively being managed to ensure the continued effectiveness of the neutron shielding material:

- Cracking due to material property changes from radiation exposure – Polypropylene and borated polyester materials of the cask.

Table 3.2-1
AMR Results for Casks

Subcomponent	Intended Function	Material	Environment[4]	Aging Effect	Aging Mechanism	Aging Management Activities
Shell	HT, PB, SH, SS	Carbon Steel	(I) Air/Gas	None	N/A	N/A
			(E) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Lid	HT, PB, SH, SS	Carbon Steel	(I) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Galvanic Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Inner Containment	HT, PB, SH, SS	Nickel-Based Alloys	(I) Air/Gas	None	N/A	N/A
			(E) Air/Gas	None	N/A	N/A
Bottom	HT, PB, SH, SS	Carbon Steel	(I) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Bottom Inner Containment	HT, PB, SH, SS	Nickel-Based Alloys	(I) Air/Gas	None	N/A	N/A
			(E) Air/Gas	None	N/A	N/A

Table 3.2-1
AMR Results for Casks (Continued)

Subcomponent	Intended Function	Material	Environment[4]	Aging Effect	Aging Mechanism	Aging Management Activities
Upper Trunnion Lower Trunnion	SS	Carbon Steel	(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Shield Plate	SH	Carbon Steel	(I) Air/Gas	None	N/A	N/A
			(E) Air/Gas	None	N/A	N/A
Outer shell	SH, SS	Carbon Steel	(I) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Top Neutron Shield	SH	Polypropylene	(E) Air/Gas	Cracking[5]	Material property changes from radiation exposure	ISFSI Inspection and Monitoring Activities Program
Top Neutron Shield Enclosure[1]	SS	Carbon Steel	(I) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Top Neutron Shield Bolts	SS	Carbon Steel	(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program

Table 3.2-1
AMR Results for Casks (Continued)

Subcomponent	Intended Function	Material	Environment[4]	Aging Effect	Aging Mechanism	Aging Management Activities
Radial Neutron Shield	SH	Borated Polyester	(E) Air/Gas	Cracking[5]	Material property changes from radiation exposure	ISFSI Inspection and Monitoring Activities Program
Radial Neutron Shield Box[2]	HT, SS	Aluminum	(I) Air/Gas	None	N/A	N/A
			(E) Air/Gas	None	N/A	N/A
Lid Bolts	PB, SS	Carbon Steel	(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Galvanic Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Lid Seal (O-ring)	PB	Aluminum	(I) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather (outer)	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Galvanic Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Vent Port Covers	PB	Stainless Steel	(I) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Drain Port Covers	PB	Stainless Steel	(I) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Drain and Vent Port Cover Bolts	PB, SS	Carbon Steel	(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Galvanic Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program

Table 3.2-1
AMR Results for Casks (Continued)

Subcomponent	Intended Function	Material	Environment[4]	Aging Effect	Aging Mechanism	Aging Management Activities
Drain and Vent Port Cover Seals (O-ring)	PB	Aluminum	(I) Air/Gas	None	N/A	N/A
			E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Galvanic Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
Basket Rails	HT, SS	Aluminum	(E) Air/Gas	None	N/A	N/A
Fuel Compartment	CC, HT, SS	Stainless Steel	(E) Air/Gas	None	N/A	N/A
Aluminum Plate[3]	HT	Aluminum	(I) Air/Gas	None	N/A	N/A
			(E) Air/Gas	None	N/A	N/A
Poison Plate	CC, HT	Borated Compounds	(E) Air/Gas	None	N/A	N/A
Containment Flange	PB, SS	Carbon Steel	(I) Air/Gas	None	N/A	N/A
			(E) Air/Gas	None	N/A	N/A
			(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Galvanic Corrosion	ISFSI Inspection and Monitoring Activities Program
					General Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program
		Stainless Steel	(E) Atmosphere/ Weather	Loss of Material	Crevice Corrosion	ISFSI Inspection and Monitoring Activities Program
					Pitting Corrosion	ISFSI Inspection and Monitoring Activities Program

[1] This is a Carbon Steel enclosure, plate or shell encasing Polypropylene.

[2] This is an Aluminum enclosure, plate or shell encasing Borated Polyester.

[3] This includes the Aluminum enclosure, plate or shell encasing Borated Compounds (i.e., Boral®, Aluminum Metal Matrix Composite [MMC] or Borated Aluminum).

[4] (I) refers to an internal environment and (E) refers to an external environment.

[5] While the Aging Management Review of the polymeric neutron shielding material did not identify any aging effects that could lead to a loss of intended function, this aging effect is conservatively being managed to ensure the continued effectiveness of the neutron shielding material.

Enclosure 3

**AREVA Affidavit Pursuant
To 10 CFR 2.390**

1 page follows

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

AREVA Inc.)
State of Maryland) SS.
County of Howard)

I, Jeffery Isakson, depose and say that I am a Vice President of AREVA Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The document for which proprietary treatment is sought is listed below. The calculation listed below is intended to be submitted to the United States Nuclear Regulatory Commission by Xcel Energy:

- AREVA Inc., Calculation # TN40HT-0513 Revision 1, "Evaluation of the potential for the buildup of flammable gases in the TN-40 dry cask system"

The document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by AREVA Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the regulations, the following is furnished for consideration in determining whether the information sought to be withheld from public disclosure, included in the files described above, should be withheld.

1) The information sought to be withheld from public disclosure involves details and analysis related to the design of a dry spent fuel storage system which are owned and have been held in confidence by AREVA Inc.

2) The information is of a type customarily held in confidence by AREVA Inc. and not customarily disclosed to the public. AREVA Inc. has a rational basis for determining the types of information customarily held in confidence.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of AREVA Inc. because the information involves details and analysis related to the design of a dry spent fuel storage system, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with AREVA Inc., take marketing or other actions to improve their product's position or impair the position of AREVA Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Jeffery Isakson
Vice President, AREVA Inc.

Subscribed and sworn to me before this 11th day of July, 2014.

Notary Public

My Commission Expires 11 /17 2014

Patricia Hoch
Notary Public
Howard County, MD
My Commission Expires Nov. 17, 2014