George H. Gellrich
Vice President

Calvert Cliffs Nuclear Power Plant, LLC
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410.495.5200
410.495.3500 Fax

CENG
a joint venture of
Constellation Energy · EDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

September 17, 2010

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation
Material License No. SNM-2505, Docket No. 72-8
Site-Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application

Pursuant to 10 CFR 72.42(b) and (c), Calvert Cliffs Nuclear Power Plant, LLC. (Calvert Cliffs) submits an application for renewal of the Calvert Cliffs Nuclear Power Plant site-specific Independent Spent Fuel Storage Installation license. The current license expires on November 30, 2012. The application for the renewal of the license is contained in Attachment (1) and was prepared in accordance with applicable provisions of 10 CFR Part 72, Subpart B. Based on the expected duration of the Calvert Cliffs operating plant licenses and the estimated time needed to remove the storage casks from the site, Calvert Cliffs is requesting, in Attachment (2), an exemption from 10 CFR 72.42(a) to seek a license renewal period of 40 years.

NMSS01
NMSS

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, George H. Gellrich, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, LLC (CCNPP), and that I am duly authorized to execute and file this response on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of St. Mary's, this 17th day of September, 2010.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires: March 1, 2011
Date

GHG/KLG/bjd

Attachments: (1) Application for Renewal of the Site-Specific License
Appendix A Aging Management Program
Appendix B Time-Limited Aging Analysis
Appendix C ISFSI Updated Safety Analysis Report Supplement and Changes
Appendix D Technical Specifications Changes
Appendix E Environmental Report Supplement
Appendix F Additional Information
(2) Calvert Cliffs Independent Spent Fuel Storage Installation Request for Exemption from 10 CFR 72.42(a)

cc: D. V. Pickett, NRC
M. L. Dapas, NRC
Resident Inspector, NRC
S. Gray, DNR
J. Goshen, NMSS
E. Ghigiarelli, MDE
V. Ordaz, NMSS

ATTACHMENT (1)

APPLICATION FOR RENEWAL OF THE SITE-SPECIFIC LICENSE

Calvert Cliffs Nuclear Power Plant, LLC
September 17, 2010

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1.0 GENERAL INFORMATION

Calvert Cliffs Nuclear Power Plant, LLC (Calvert Cliffs) has prepared this application for renewal of the license for the site-specific Independent Spent Fuel Storage Installation (ISFSI) located at Calvert Cliffs Nuclear Power Plant. This application supports ISFSI license renewal for an additional 40 year period beyond the end of the current license term of Materials License Number SNM-2505 (Docket No. 72-8). The original 20 year ISFSI license will expire on November 30, 2012. This application is submitted in accordance with 10 CFR 72.42(b) and includes the applicable general, technical, and environmental supporting information required by 10 CFR Part 72, Subpart B.

1.1 APPLICATION FORMAT AND CONTENT

The format and content of the application are based on 10 CFR Part 72, the Nuclear Regulatory Commission's (NRCs) draft NUREG-1927 (Reference 1.4.1), the NRCs preliminary guidance for renewal of site-specific 10 CFR Part 72 licenses (Reference 1.4.2), and the Oconee Nuclear Station site-specific ISFSI license renewal application (Reference 1.4.3) and include:

1. General Information - Section 1.0 has been expanded beyond the general administrative requirements of 10 CFR 72.22 to provide (1) information on the format and content of the application, and (2) general facility description.
2. Scoping Evaluation - Section 2.0 provides the scoping evaluation for the site-specific ISFSI systems, structures, and components (SSCs).
3. Aging Management Reviews - Section 3.0 includes the methodology and results of the aging management reviews performed for site-specific ISFSI SSCs that are in the scope of license renewal.
4. Appendices:
   Appendix A: Aging Management Program
   Appendix B: Time-Limited Aging Analyses (TLAAs)
   Appendix C: ISFSI Updated Safety Analysis Report Supplement and Changes
   Appendix D: Technical Specifications Changes
   Appendix E: Environmental Report Supplement
   Appendix F: Additional Information

1.2 FACILITY DESCRIPTION

The ISFSI is located on the Calvert Cliffs Nuclear Power Plant site near Lusby, MD. Calvert Cliffs Nuclear Power Plant, LLC owns and operates two 2737 MWt nuclear generating units on the Calvert Cliffs site. The ISFSI is located outside the protected area, but within the owner controlled area approximately 2300' southwest of the plant.

The ISFSI was originally licensed with the Nutech Horizontal Modular Storage (NUHOMS)-24P dry storage system. Use of the NUHOMS-24P dry storage system was approved by the NRC in Reference 1.4.4. The principal components are a horizontal storage module (HSM) comprised of concrete and structural steel and a steel dry shielded canister (DSC) with an internal basket which holds the spent fuel. The exterior walls and roof of the HSM are 3 feet thick, and the interior walls are two feet thick. Each HSM contains one DSC and each DSC contains 24 fuel assemblies.

Calvert Cliffs has reanalyzed the ISFSI to use Transnuclear NUHOMS-32P DSCs to optimize its dry spent fuel storage capacity. Use of the NUHOMS-32P DSCs was approved by the NRC in References 1.4.5 and 1.4.6. The NUHOMS-32P DSC system stores eight more spent fuel assemblies than the NUHOMS-24P DSC using the same external and internal shell dimensions. The NUHOMS-32P DSC

storage capacity is optimized by reducing the space between the locations of each fuel assembly and by slightly reducing the size of the storage locations.

In addition to these primary components, the ISFSI also requires transfer equipment to move the DSCs from the spent fuel pool (SFP) (where they are loaded with spent fuel) to the HSMs (where they are stored). This transfer system consists of a transfer cask, a hydraulic ram, a truck, a trailer, and a cask skid. This transfer system interfaces with the existing SFP, the cask handling crane, and the site layout (i.e., roads and topography).

The ISFSI currently consists of 72 HSMs with 48 of the HSMs loaded with the NUHOMS-24P DSCs and 15 of the HSMs loaded with the NUHOMS-32P DSCs. The remaining empty HSMs are expected to be loaded with the NUHOMS-32P DSCs.

A complete description of the ISFSI is provided in Calvert Cliffs ISFSI Updated Safety Analysis Report (USAR).

1.3 INFORMATION REQUIRED BY 10 CFR 72.22

1.3.1 NAME OF APPLICANT

Calvert Cliffs Nuclear Power Plant, LLC

1.3.2 ADDRESS OF APPLICANT

100 Constellation Way, Suite 1800P
Baltimore, Maryland, 21202

1.3.3 ADDRESS OF CALVERT CLIFFS ISFSI

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

1.3.4 DESCRIPTION OF BUSINESS OR OCCUPATION OF APPLICANT

Calvert Cliffs Nuclear Power Plant, LLC, the sole owner and licensed operator of the Calvert Cliffs ISFSI is engaged in the business of generating electric power. Calvert Cliffs Nuclear Power Plant, LLC is a wholly owned subsidiary of Constellation Energy Nuclear Group, LLC which is partially owned (less than 50%) by EDF Inc. which itself is a subsidiary of a foreign corporation. The NRC license transfer order approving the ownership structure and name change from Calvert Cliffs Nuclear Power Plant, Inc. to Calvert Cliffs Nuclear Power Plant, LLC, was originally issued on October 9, 2009 and was reaffirmed by a superseding order on October 30, 2009.

1.3.5 ORGANIZATION AND MANAGEMENT OF APPLICANT

The business of Calvert Cliffs Nuclear Power Plant, LLC is conducted by its executive officers and senior leadership.

States of Establishment and Places of Business

Calvert Cliffs Nuclear Power Plant, LLC is a limited liability company duly organized and existing under the laws of the State of Maryland. The company's general office, and principal place of business, is located in Maryland.

The company is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government within the meaning of Sections 103d or 104d of the Atomic Energy Act of 1954. This conclusion was affirmed in the NRC orders referenced in Section 1.3.4, above. The company makes this application on its own behalf and is not acting as an agent or representative of any other person.

Executive Officers and Senior Leadership of Applicant

The business address, names, current titles, and citizenship of Calvert Cliffs Nuclear Power Plant, LLC current executive officers and senior nuclear leadership are as follows:

Calvert Cliffs Nuclear Power Plant, LLC
100 Constellation Way, Suite 1800P
Baltimore, Maryland 21202

Name	Position	Citizenship
Henry B. Barron	Chairman of the Board and President	USA
George S. Gellrich	Vice President	USA
Simone Rossi	Treasurer	Italy
Steven L. Miller	Secretary	USA
Carey W. Fleming	Assistant Secretary	USA

1.3.6 FINANCIAL QUALIFICATIONS OF CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

Operating Financial Qualification of Calvert Cliffs Nuclear Power Plant, LLC

Calvert Cliffs Nuclear Power Plant, LLC will remain financially qualified to carry out the operation and decommissioning of the ISFSI during the period of the renewed material license as required by 10 CFR 72.22(e). Information supporting this statement is submitted to the NRC in annual financial reports as required by regulation.

1.4 REFERENCES (SECTION 1.0, GENERAL INFORMATION)

1.4.1 NUREG-1927, September 2009, Staff Review Plan for Renewal of Independent Spent Fuel Storage Installation Licenses and Dry Cask Storage System Certificates of Compliance (Draft Report)

1.4.2 Letter from E. W. Brach (NRC) to W. R. Mathews (Virginia Electric), dated March 29, 2001, Preliminary NRC Staff Guidance for 10 CFR Part 72 License Renewal

1.4.3 Letter from D. A. Baxter (Duke) to Document Control Desk (NRC), dated January 30, 2008, Site Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application

1.4.4 Letter from C. J. Haughney (NRC) to G. C. Creel (BGE), dated November 25, 1992, Materials License No. SNM-2505

1.4.5 Letter from R. J. Lewis (NRC) to G. Vanderheyden (CCNPP), dated June 10, 2005, Amendment 6 to Material License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation

1.4.6 Letter from R. A. Nelson (NRC) to G. Vanderheyden (CCNPP), dated November 2, 2005, Amendment 7 to Materials License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation

2.0 SCOPING EVALUATION

2.1 INTRODUCTION

A general description of the ISFSI is provided in Section 1.2, above. A more thorough description of the ISFSI is contained in the ISFSI USAR (Reference 2.4.1).

Constellation Energy's license renewal process for the ISFSI is consistent with the pilot site-specific ISFSI license renewal process developed by Virginia Electric and Power Company, and the NRC for the Surry Power Station site-specific ISFSI and subsequently followed by Progress Energy for the Robinson Nuclear Plant site-specific ISFSI license renewal and in the Duke Energy license renewal submittal for the Oconee Nuclear Station site-specific ISFSI.

The Calvert Cliffs ISFSI license renewal methodology follows References 2.4.2 and 2.4.3. The proposed 10 CFR Part 72 license renewal process adopts the regulatory philosophy of 10 CFR Part 54. This philosophy is summarized in the two principles of license renewal from 10 CFR Part 54 Final Rule Statements of Consideration (Reference 2.4.4) which are re-stated below:

> *"The first principle of license renewal was that, with the exception of age-related degradation unique to license renewal and possibly a few other issues related to safety only during the period of extended operation of nuclear power plants, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provides and maintains an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security. Moreover, consideration of the range of issues relevant only to extended operation led the Commission to conclude that the detrimental effects of aging is probably the only issue generally applicable to all plants. As a result, continuing this regulatory process in the future will ensure that this principle remains valid during any period of extended operation if the regulatory process is modified to address age-related degradation that is of unique relevance to license renewal. ..."*
>
> *"The second and equally important principle of license renewal holds that the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term. This principle would be accomplished, in part, through a program of age-related degradation management for systems, structures, and components that are important to license renewal ..."*

Based on these principles, license renewal is not intended to impose requirements beyond those that were met by the facility when it was initially licensed by the NRC. Therefore, the current licensing basis for the ISFSI will be carried forward through the renewed license period.

The scoping process involves identification of the SSCs of the ISFSI that are within the scope of license renewal, and thus require evaluation for the effects of aging. A description of the scoping process is provided in Section 2.2, below.

2.2 SCOPING METHODOLOGY

The first step in the license renewal process involves the identification of the in-scope ISFSI SSCs. This is done by evaluating the SSCs that comprise the ISFSI against the following scoping criteria provided in the comments in Reference 2.4.3:

"Any SSC that meets either of the criteria shall be evaluated further in the aging management review (AMR) process described later. The categories of SSCs are those that are:

1. *Important to safety, as they are relied upon to: (a) maintain the conditions required to store spent fuel safely, (b) prevent damage to the spent fuel during handling and storage, or (c) provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public, as identified in the CLB. These SSCs ensure that important safety functions are met for: (1) criticality, (2) shielding, (3) confinement, (4) heat transfer, and (5) structural integrity.*

2. *Classified as not important to safety, but, according to the CLB, whose failure could prevent an important to safety function from being fulfilled or whose failure as a support SSC could prevent an important to safety function from being fulfilled.*

The function performed by a SSC that causes it to be within the scope of license renewal is its intended function. ..."

"Also, SSCs which perform ISFSI support functions are generally not within the scope of license renewal. ..."

"The fuel in storage is considered to be within the scope of license renewal. The effects of aging of fuel shall be evaluated."

Any ISFSI SSC that meets either Scoping Criterion 1 or 2 above is considered within the scope of license renewal (in-scope), and the function(s) it is required to perform during the extended term is identified. The results of the scoping evaluation are presented in Section 2.3.

A basic premise of the license renewal scoping process is that the current licensing basis identifies SSCs and their intended functions. Thus, the current licensing basis is reviewed to determine those SSCs with intended functions that meet either Scoping Criterion 1 or 2, as defined above. The following documents comprise the current licensing basis for the ISFSI:

- ISFSI USAR (Reference 2.4.1)
- Materials License No. SNM-2505 (Reference 2.4.5)
- ISFSI Technical Specifications (Reference 2.4.5)

The ISFSI USAR provides a description of the ISFSI, and ISFSI SSCs and their functions, including safety classifications as established by the safety analyses. The ISFSI Technical Specifications govern the receipt, possession, and storage of irradiated nuclear fuel at the Calvert Cliffs ISFSI, and the transfer of such irradiated fuel to and from the SFP and the ISFSI. Additionally, the Safety Evaluation Reports (Reference 2.4.6), which summarize the results of the NRC staff's safety review of the original licensing, and the Safety Evaluation Reports associated with subsequent amendments were reviewed in the license renewal scoping process.

Other design and design basis documents such as the NUHOMS®-24P Topical Report (Reference 2.4.7) were consulted as appropriate to further clarify SSC descriptions, classifications, and intended functions.

2.3 SCOPING RESULTS

The SSCs comprising ISFSI are identified in Table 2.3-1, Scoping Results. Those SSCs meeting Scoping Criterion 1 or 2 are identified in the table as being within the scope of license renewal.

As indicated in Table 2.3-1, only the HSMs, the DSCs, the irradiated fuel assemblies stored in the canisters, the transfer cask, the transfer cask lifting yoke, spent fuel handling machine (SFHM), spent fuel cask handling crane, and the cask support platform in the Unit 1 SFP were determined to be within the scope of license renewal and to require further review in the aging management review process.

The intended functions performed by the individual subcomponents of these in-scope SSCs are identified in the aging management review summary tables (Tables 3.2-1, 3.3-1, 3.4-1, 3.5-1, 3.6-1, and 3.7-1), which are located at the end of Section 3.0.

Table 2.3-1, Scoping Results

SSC	Criterion 1	Criterion 2	In-Scope
Irradiated Fuel Assemblies	Yes	N/A	Yes
DSC[1]	Yes[2]	N/A	Yes
HSM[3]	Yes[2]	N/A	Yes
Transfer Cask[4]	Yes[2]	N/A	Yes
Transfer Cask Lifting Yoke	Yes[2] (SR)	N/A	Yes
Cask Support Platform[5]	Yes[2] (SR)	N/A	Yes
Spent Fuel Cask Handling Crane	Yes[2] (SR)	N/A	Yes
SFHM	No	Yes	Yes
Transfer Components[6]	No	No	No
Instrumentation	No	No	No
Support Equipment[7]	No	No	No
Welding Equipment[8]	No	No	No
Security Equipment[9]	No	No	No
Heavy Haul Path	No	No	No
Approach Slabs	No	No	No

Notes:

(1) Includes (but not limited to) the DSC internal basket assembly, the shielded end plugs and the DSC containment pressure boundary.

(2) Subsection 3.3.3.1 of Reference 2.4.7 states that the HSM, DSC, and on-site transfer cask are important to safety. Other equipment important to safety associated with the NUHOMS system is the equipment required for handling operations within the plant's fuel building.

(3) Includes (but not limited to) the HSM walls, roof, basemat and foundation, lightning protection system, DSC structural steel support assembly, DSC seismic retainer, cask docking flange and tie-down restraints, heat shield, shielded front access door and door supports, ventilation air opening, shielded ventilation air inlet plenum, ventilation air outlet shielding blocks.

(4) Includes (but not limited to) the cask structural shell assembly, bolted top head assembly, upper lifting trunnions, lower tiling trunnions, lead gamma shielding, solid neutron shielding, ram access penetration cover plate, ram access penetration shield plug assembly, ram mounting frame.

(5) The cask support platform is not described in the current licensing basis for the ISFSI; however, it is discussed in the Updated Final Safety Analysis Report (UFSAR) for the station (Reference 2.4.8).

(6) Includes a hydraulic ram, a truck, a trailer, skid position system, power supplies, a cask support skid and a mobile yard crane for raising and lowering the HSM front access door.

(7) Includes (but not limited to) the annulus seal, vacuum drying system, slings, DSC lift rig, ISFSI electrical power supply, miscellaneous equipment (e.g., tools, hoses).

(8) Includes RX-277 neutron shielding used by welder. This is credited in the shielding results listed in USAR Table 7.3-1 and Table 12.7-1.

(9) Includes (but not limited to) lighting, communication and alarm systems.

N/A Not Applicable
SR Safety-related

2.4 REFERENCES

2.4.1 Calvert Cliffs Independent Spent Fuel Storage Installation Updated Safety Analysis Report, Revision 17, September 8, 2008

2.4.2 Enclosure to letter from Mr. E. W. Brach (NRC) to Mr. W.R. Matthews (Virginia Electric Power Company), dated March 29, 2001, Preliminary NRC Staff Guidance for 10 CFR 72 License Renewal

2.4.3 Attachment to letter from Mr. L. N. Hartz (Virginia Electric Power Company) to Document Control Desk (NRC), dated June 26, 2001, Surry Independent Spent Fuel Storage Installation, Comments on NRC Preliminary Guidance for Part 72 License Renewal

2.4.4 Federal Register, Volume 60, No. 88, Page 22464, dated May 8, 1995, Nuclear Power Plant License Renewal; Revisions, 10 CFR Parts 2, 51, and 54

2.4.5 Calvert Cliffs Independent Spent Fuel Storage of Installation Materials License No. SNM-2505 and Appendix A, Technical Specifications

2.4.6 Safety Evaluation Reports (SER) for Calvert Cliffs Independent Spent Fuel Storage Installation Materials License No. SNM-2505, Docket No. 72-8

2.4.6.1 Amendment No. 0, November 1992
2.4.6.2 Amendment No. 1, July 1995
2.4.6.3 Amendment No. 2, June 30, 2000
2.4.6.4 Amendment No. 3, April 17, 2001
2.4.6.5 Amendment No. 4, August 29, 2001
2.4.6.6 Amendment No. 5, October 25, 2001 (the Safety Evaluation is contained within the October 25, 2001 NRC letter to the Calvert Cliffs)
2.4.6.7 Amendment No. 6, June 10, 2005
2.4.6.8 Amendment No. 7, November 2, 2005
2.4.6.9 Errata for the U.S. Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER) Related to Amendment 7 to Materials License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation, November 23, 2005

2.4-7 NUH-002, Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS®-24P, Revision 2, March 1990

2.4-8 Calvert Cliffs Updated Final Safety Analysis Report, Revision 40, September 8, 2008

3.0 AGING MANAGEMENT REVIEWS

3.1 AGING MANAGEMENT REVIEW METHODOLOGY

The scoping process identifies the ISFSI SSCs within the scope of license renewal which require evaluation for the effects of aging in the aging management review process.

The purpose of the aging management review process is to assess the in-scope SSCs with respect to aging effects that could affect the ability of the SSC to perform its intended function during the renewed license period. The aging management review process involves the following four (4) major steps:

1. Identification of in-scope subcomponents requiring aging management reviews (screening);
2. Identification of materials and environments;
3. Identification of aging effects requiring management; and,
4. Determination of the activities/programs required to manage the effects of aging.

Each of these steps is discussed in Subsections 3.1.1 through 3.1.4. Also, the operating experience review for confirmation of the aging management review process and the document sources used in the process are discussed in Subsections 3.1.5 and 3.1.6.

The results of the aging management review for the subcomponents of the ISFSI SSCs that are in the scope of license renewal are provided in Sections 3.2 through 3.9. Corresponding tables that summarize the aging management review for the ISFSI SSCs that require an aging management review are located at the end of Section 3.0.

3.1.1 IDENTIFICATION OF IN-SCOPE SUBCOMPONENTS REQUIRING AGING MANAGEMENT REVIEW

The scoping process does not identify the specific subcomponents for the in-scope ISFSI SSCs that require an aging management review. Therefore, in the first step of the aging management review process, the in-scope SSCs are further reviewed to identify and describe the subcomponents that support the SSC intended function. The subcomponents and associated intended functions are identified by reviewing the documentation sources identified in Subsection 3.1.6.

Subcomponents that perform or support any one of the identified intended functions in a passive manner, without moving parts or a change in configuration or properties, are determined to require an aging management review.

Those subcomponents that either do not support an intended function, or perform an intended function by a change in configuration or properties (active), or have their condition monitored at some established frequency, are excluded from further evaluation in the aging management review with supporting justification.

Tables 3.2-1, 3.3-1, 3.4-1, 3.5-1, 3.6-1, and 3.7-1 identify the intended functions for the ISFSI subcomponents that require an aging management review. The tables also identify subcomponents that do not support the SSC intended function and are not subject to an aging management review.

3.1.2 IDENTIFICATION OF MATERIALS AND ENVIRONMENTS

The second step of the aging management review process is the identification of the materials of construction and the environments to which these materials are exposed for the ISFSI subcomponents that require an aging management review.

The materials of construction were identified through a review of pertinent design and/or design basis documents, which are discussed in Subsection 3.1.6. A summary of the materials of construction is provided in Subsections 3.2.2, 3.3.2, 3.4.2, 3.5.2, 3.6.2, and 3.7.2 and are reflected in the corresponding aging management review summary tables (Tables 3.2-1, 3.3-1, 3.4-1, 3.5-1, 3.6-1, and 3.7-1). Sections 3.8 and 3.9 discuss the aging management reviews for the spent fuel cask handling crane, and the SFHM.

The environments to which components are exposed play a critical role in the determination of potential aging mechanisms and effects. A review of plant documentation, discussed in Subsection 3.1.6, was performed to quantify the environmental conditions to which the ISFSI SSCs are continuously or frequently exposed. The environmental conditions identified during this review include any conditions known to exist on a recurring basis. They are based on operating experience, unless design features have been implemented to preclude those conditions from recurring. Descriptions of the internal and external environments, which have been used in the aging management review, are included in Subsections 3.2.3, 3.3.3, 3.4.3, 3.5.3, 3.6.3, and 3.7.3, and are reflected in the corresponding aging management review summary tables.

3.1.3 IDENTIFICATION OF AGING EFFECTS REQUIRING MANAGEMENT

The third step in the aging management review process involves the identification of the aging effects requiring management. Aging effects requiring management during the renewed license period are those that could cause a loss of passive SSC intended function(s). If degradation of a subcomponent would be insufficient to cause a loss of function, or the relevant conditions do not exist at the Calvert Cliffs ISFSI for the aging effect to occur and propagate, then no aging management is required.

Potential aging effects, presented in terms of material and environment combinations, have been evaluated and those aging effects requiring management have been determined. Both potential aging effects that theoretically occur, as well as aging effects that have actually occurred based upon industry and Calvert Cliffs operating experience, were considered. The evaluation was applied to subcomponents, regardless of form (i.e., canister body, cover, lid, guide tube, etc.).

As described above, the environments considered in this evaluation are the environments that the subcomponents normally experience. Environmental stressors that are conditions not normally experienced (such as accident conditions), or that may be caused by a design problem, are considered event-driven situations and have not been characterized as sources of aging. Such event-driven situations would be evaluated and corrective actions, if any, implemented at the time of the event.

Aging effects are the manifestation of aging mechanisms. In order to effectively manage an aging effect, it is necessary to determine the aging mechanisms that are potentially at work for a given material and environment application. Therefore, the aging management review process identifies both the aging effects and the associated aging mechanisms which cause them. Various mechanisms are only applicable at certain conditions, such as high temperature or moisture, for example. Each identified mechanism was characterized by a set of applicable conditions that must be met for the mechanism to occur and/or propagate. Given this evaluation process, each subcomponent that was subjected to aging management review was evaluated to determine if the potential aging effects/mechanisms were credible considering the material, environment, and conditions of storage.

3.1.3.1 Identification of Aging Effects Related to NUHOMS Storage System

The potential aging effects/mechanisms that were considered for the ISFSI were based primarily upon the Calvert Cliffs Nuclear Power Plant license renewal (10 CFR Part 54) process. These were

compared to the following guidance documents and reports to ensure that all aging effects/mechanisms applicable to Calvert Cliffs were identified:

- NRC Preliminary Guidance for License Renewal including comments provided by Virginia Electric Power Company (Reference 3.10.1)
- American Society for Testing and Materials (ASTM) C 1562-03 (Reference 3.10.2)
- Electric Power Research Institute (EPRI) Report TR-1003416 (Reference 3.10.3)
- EPRI Report TR-108757 (Reference 3.10.4)
- EPRI Report TR-1002882 (Reference 3.10.5)
- International Atomic Energy Agency Technical Report Series No. 443 (Reference 3.10.6)
- NRC Interim Staff Guidance (ISG) 11, Revision 3 (Reference 3.10.7)

A summary of aging effects requiring management for the subcomponents of irradiated fuel assemblies, the DSCs, the HSMs, the transfer cask, the lifting yoke, and the cask support platform, is provided in Subsections 3.2.5, 3.3.5, 3.4.5, 3.5.5, 3.6.5, and 3.7.5, respectively. The aging effects that require management during the renewed license period are also reflected in the corresponding aging management review summary tables.

Subsections 3.2.4, 3.3.4, 3.4.4, 3.5.4 and 3.6.4 provide a TLAA for the fuel cladding, the DSCs, the HSMs, the transfer cask, and the transfer cask lifting yoke, respectively.

3.1.3.2 Identification of Aging Effects Related to Fuel Assemblies

This section identifies the possible effects of storage on irradiated fuel assemblies. Relevant EPRI, ASTM, and NRC documents were used to identify the possible aging effects.

EPRI Report on Data Needs for Storage

The EPRI contracted with Battelle's Pacific Northwest Division to prepare a report (Reference 3.10.4) on data needs for long-term dry storage. The report is helpful in that it is not limited to data needs. The report also discusses available data and its usefulness in treating some degradation mechanisms. The emphasis of this report is on fuel performance during the period from 20 to 100 years after the fuel is put into dry storage.

The report's summary section provides an assessment of the durability of spent fuel: "The results obtained so far lead to the view that any concerns about long-term dry storage, in all likelihood, do not lie with the behavior of the spent-fuel assemblies – at least for those with burnups lower than approximately 50,000 MWd/MTU." The remainder of the report provides details to support this conclusion.

The maximum assembly average burnup for the fuel currently (2010) stored at the ISFSI does not exceed 47 GWd/MTU per the ISFSI Technical Specification Limiting Condition for Operation 3.1.1(3) (Reference 3.10.8). This fuel is bounded by the conditions analyzed in Reference 3.10.4.

However, Calvert Cliffs has just obtained approval to increase the maximum assembly average burnup allowed to be loaded into the NUHOMS-32P DSCs to 52 GWd/MTU. The aging effects/mechanisms for this higher burnup fuel are addressed in references discussed below.

EPRI Report on Bases for Extended Dry Storage

Electric Power Research Institute produced a second report (Reference 3.10.3), which to some extent is a supplement to Reference 3.10.4. The report reviews possible fuel and cladding degradation mechanisms. As discussed in Reference 3.10.4, only hydrogen embrittlement was considered as an applicable degradation mechanism to extended dry storage under normal conditions.

ASTM Standard on Extended Dry Storage

The ASTM produced a consensus standard (Reference 3.10.2), which discusses possible fuel and cladding degradation mechanisms. This standard also views hydrogen embrittlement as a degradation mechanism applicable to extended dry storage.

Cladding Considerations for the Transportation and Storage of Spent Fuel

The NRC developed ISG 11 (Reference 3.10.7) to define the acceptance criteria needed to provide reasonable assurance that commercial spent fuel is maintained in the configuration that is analyzed in the licensee's safety analysis report for spent fuel storage. Interim Staff Guidance 11 broadened the NRC's technical basis for storage of spent fuel with average burnups exceeding 45 GWd/MTU. At fuel burnups exceeding 45 GWd/MTU, the buildup of hydrogen in the cladding and wall thinning due to corrosion are a concern. Interim Staff Guidance 11 emphasizes the need to limit the formation of radial hydrides, which can occur when high burnup cladding (high hydrogen content) experiences a high tensile stress. Therefore, the ISG limits the peak cladding temperature (peak tensile stress) and the number of thermal cycles to prevent high tensile stress conditions from occurring. The ISG states that maintaining peak cladding temperature less than 400°C will limit cladding hoop stress and limit the amount of soluble hydrogen available to form the radial hydrides. Calculations used in support of the ISFSI license amendment request for storage of higher burnup fuel up to 52 GWd/MTU have used this 400°C limit. As a result, Reference 3.10.7 is considered to adequately bound conditions associated with the higher burnup limit of 52 GWd/MTU.

The NRC also discussed the applicability of this ISG to storage of fuel with burnups less than 45 GWd/MTU. "Based on staffs evaluation, it is expected that fuel assemblies with burnups less than 45 GWd/MTU are not likely to have a significant amount of hydride reorientation due to limited hydride content ... Even if hydride reorientation occurred during storage, the network of reoriented hydrides is not expected to be extensive enough in low burnup fuel to cause fuel rod failures." Thus storage of fuel with burnups of less than 45 GWd/MTU is not considered susceptible to radial hydride formation. This results in the storage of fuel with burnups of less than 45 GWd/MTU to also be bounded by Reference 3.10.7.

Dry Cask Storage Characterization Project

In the mid-1980s Department of Energy sponsored the Dry Cask Storage Characterization Project to evaluate the thermal performance of a Castor dry storage cask system (Reference 3.10.5). Surry pressurized water reactor (PWR) fuel was placed in the Castor dry storage cask system and exposed to six thermal cycles (referred to as "benchmark testing"); the two hottest cycles reached fuel temperatures of 415°C and 398°C. After the last thermal test, the cask was stored on a concrete pad for about 15 years.

As part of an EPRI and NRC program to evaluate dry storage facility license renewal, fuel from this cask was then removed and examined. The fuel assembly was a Westinghouse 15x15

assembly with an assembly-averaged burnup of 35.7 GWd/MTU. The fuel was 3.11% enriched and 95% dense. The cladding was cold-worked/stressed-relieved Zircaloy-4.

Detailed examination showed that the fuel was suitable for extended storage. No deleterious effects such as fission gas release, cladding creep, cladding hydride reorientation, or cladding property degradation was observed.

In terms of cladding material, assembly burnup, and pellet enrichment, this fuel is similar to that being stored at the ISFSI. Therefore, the report's observations are also applicable to ISFSI's license renewal, which are as follows:

1. The rods experienced very little thermal creep during benchmark testing and storage. Little additional creep would be expected for additional storage duration because of the low temperature.
2. No additional fission gas appears to have been released. This means further pressurization of the cask is not expected.
3. No evidence of hydrogen pickup or hydride reorientation was observed. A small amount of axial migration of hydrogen to cooler sections might have occurred.
4. Little or no cladding annealing occurred during either the benchmark testing or long-term storage.
5. Creep tests on post-storage samples showed residual creep strains exceeding 1% with the 400°C sample exceeding 6%. The cladding retains significant creep ductility.
6. The fuel was suitable for extended storage.
7. No deleterious effects from 15 years of dry cask storage were observed.

Conclusions

This section identified hydrogen embrittlement (References 3.10.3 and 3.10.4) and radial hydride formation (Reference 3.10.5) as possible degradation mechanisms. Fuel with burnup of less than 45 GWd/MTU is not impacted by either of these mechanisms (References 3.10.4 and 3.10.7). Fuel with burnup greater than 45 GWd/MTU minimizes the likelihood of these degradation mechanisms from occurring by limiting peak cladding temperature to less than 400°C. The results of the Dry Cask Storage Characterization Project (Reference 3.10.5) support the conclusion that the condition of irradiated fuel assemblies will not degrade under extended storage.

3.1.4 DETERMINATION OF THE ACTIVITIES REQUIRED TO MANAGE THE EFFECTS OF AGING

The final step in the aging management review process involves the determination of the aging management activities or the aging management program to be credited or developed for managing the effects of aging. To the extent practical, the existing ISFSI programs and/or activities were credited for the management of aging effects that could cause a loss of component intended function during the renewed license period.

As indicated in Subsections 3.2.5 and 3.3.5, and reflected in the corresponding aging management review summary tables (Tables 3.2-1 and 3.3-1), there are no aging effects requiring management during the renewed license period for the subcomponents of the irradiated fuel assemblies or DSCs. The aging management programs for the HSM, transfer cask, transfer cask lifting yoke, cask support platform, and

subcomponents are described in Subsections 3.4.5, 3.5.5, 3.6.5, and 3.7.5, respectively, and are listed in the corresponding aging management review summary tables (Tables 3.4-1, 3.5-1, 3.6-1, and 3.7-1).

The demonstration of the effectiveness of the aging management programs that were selected for the HSMs, transfer cask, transfer cask lifting yoke, and the cask support platform, is discussed in Appendix A, Aging Management Program.

3.1.5 OPERATING EXPERIENCE REVIEW FOR PROCESS CONFIRMATION

Further review of industry and plant-specific operating experience for the ISFSI was also performed in order to confirm the applicability of previously identified potential aging effects/mechanisms and to identify any aging effects not previously addressed in aging effect evaluations.

This ISFSI operating experience review is described in the following subsections. A discussion of operating experience, as it pertains to the effectiveness of aging management programs credited with the management of aging, is also contained in the appropriate subsections of Appendix A.

The ISFSI design employs a rugged sealed canister and concrete storage module for dry storage of irradiated fuel, as described in this application. However, for completeness, operating experience for dry storage casks was also considered for applicability to the ISFSI SSCs.

3.1.5.1 Cask and Canister Degradation

A review of industry operating experience identified a large number of events related to dry storage. Many of these were event-driven incidents, and most were not related to aging management. However for those incidents that involved aging mechanisms, barrier analyses were conducted to assess Calvert Cliffs susceptibility to these mechanisms. These reviews indicated the aging mechanisms are bounded by the aging management reviews performed for ISFSI SSCs.

3.1.5.2 Fuel Assembly Degradation

The conditions and findings of the Dry Cask Storage Characterization Project conducted by EPRI (Reference 3.10.5) are considered to be representative of the conditions and materials inside a DSC at the ISFSI. The EPRI project evaluated the hydride distribution of a standard 14x14 Combustion Engineering 4-cycle Calvert Cliffs PWR assembly (ATM-106) which had a peak burnup of 46 GWd/MTU and a fission gas release of 11.3%. This is about 10 GWd/MTU higher than the Surry rods. The Calvert Cliffs rods had Sandvik Zircaloy-4 cladding, but the initial condition of the cladding was not given. At about 1 meter above the fuel midplane, both the Surry and Calvert Cliff rods showed circumferential hydrides rather evenly distributed across the cladding wall. The hydrides were in the range of 500-μm long in both cases. The EPRI project study stated that it is difficult to make any quantitative measure of the amount of hydrides and it did appear that there were more hydrides in the Calvert Cliff rods than Surry rods at this elevation. In neither case was there any evidence of radial hydrides.

The EPRI project study goes on to state that although there are differences in the appearance between the hydrides in the Turkey Point (similar fuel assembly as Surry, but at a lower burnup) and Calvert Cliffs claddings after irradiation, and the Surry cladding after storage, it is difficult to determine if radial and circumferential redistribution of hydrogen occurred following the period in vacuum ($T \approx 414°C$) and/or dry cask storage ($T \leq 350°C$) without baseline micrographs of the Surry cladding after reactor discharge. After years of storage there are very few, if any, radial hydrides in the Surry fuel rods.

The EPRI project evaluated the condition of fuel rods following approximately 15 years of dry storage. Detailed examination showed that the fuel was suitable for extended storage. No deleterious effects such as fission gas release, cladding creep, cladding hydride reorientation, or cladding property degradation was observed.

3.1.5.3 Summary of Operating Experience

Calvert Cliffs staff's review of operating experience did not identify any potential aging effects and associated mechanisms for the ISFSI beyond those described in Subsection 3.1.3. Additionally, the appropriateness of the ISFSI aging management review was confirmed by the operating experience review.

3.1.6 DOCUMENTATION OF SOURCES USED FOR THE AGING MANAGEMENT REVIEW PROCESS

The following documents were the primary sources for determination of the safety classifications, intended functions, materials, and environmental conditions for ISFSI SSCs identified as in-scope for license renewal:

- ISFSI Technical Specifications (Reference 3.10.8)
- ISFSI USAR (Reference 3.10.9)
- Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P (Reference 3.10.10)
- Calvert Cliffs UFSAR (Reference 3.10.11)

Other plant documents such as drawings, technical reports, vendor manuals, procedures were consulted, as appropriate, to further clarify SSC subcomponent safety classifications, intended functions, materials, and environmental conditions.

The documents listed in Section 2.4, were also used in the aging management review process.

Lastly, industry topical reports, reference books, and standards were consulted as appropriate for the fuel only evaluation of aging effects/mechanisms as discussed in Section 3.1.3.1.

3.2 AGING MANAGEMENT REVIEW RESULTS -- IRRADIATED FUEL ASSEMBLIES

This section provides the results of the aging management review of the irradiated fuel assemblies, which were determined to be in the scope of license renewal as identified in Section 2.3.

A summary of the results of the aging management review for the irradiated fuel assemblies subcomponents is provided in Table 3.2-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were therefore, not subjected to further aging management review.

A description of the irradiated fuel assemblies subcomponents which support an SSC intended function is provided in Subsection 3.2.1, and a summary of the materials and environments for the irradiated fuel assemblies is provided in Subsection 3.2.2 and Subsection 3.2.3, respectively. Subsection 3.2.4

references the TLAA for the fuel cladding while the irradiated fuel assemblies are stored in a DSC. Subsections 3.2.5 and 3.2.6 provide a discussion of the aging effects requiring management for the subject irradiated fuel assembly subcomponents, if any, and any aging management activities used to manage the effects of aging, respectively.

3.2.1 DESCRIPTION OF IRRADIATED FUEL ASSEMBLIES SUBCOMPONENTS

Each DSC contains either 24 or 32 PWR irradiated fuel assemblies which, at the time of loading, had a maximum heat generation limit of 0.66 kilowatt per assembly, or 15.84 kilowatts per NUHOMS-24P DSC or 21.12 kilowatts per NUHOMS-32P DSC; a maximum average burnup of 47 GWd/MTU uranium for NUHOMS-24P DSCs and 52 GWd/MTU for NUHOMS-32P DSCs; and that were cooled for at least eight years prior to storage or met the alternative fuel specifications for neutron and gamma sources (Reference 3.10.8).

The intended functions of the irradiated fuel assemblies were conservatively determined during scoping to include criticality control, pressure boundary, structural integrity, and heat transfer. The geometry of the irradiated fuel assembly is a factor in the proper conduction and convection of heat to the DSC surface and in the criticality model. The fuel cladding provides a confinement barrier, and its structural integrity is necessary to maintain a favorable geometry and for retrieval. After fuel loading and DSC drying, the spent fuel assemblies are not moderated, assuring subcriticality during subsequent operations and configurations. Furthermore, total cladding failure has been evaluated from the perspective of both DSC pressurization and DSC leakage, with the dose consequences determined to be acceptable (Sections 8.2.9 and 12.8.2.9 of Reference 3.10.9). The irradiated fuel assembly principle function during dry storage is to maintain proper geometry and position of radioactive material through confinement. The irradiated fuel assemblies currently in storage at the ISFSI are 14x14 CE PWR type fuel. The following subcomponents are applicable to the irradiated fuel assemblies currently in storage at the ISFSI:

- Fuel Rods (Cladding, End Caps/Plugs)
- Guide Tubes
- Spacer Grid Assemblies
- Lower End Fitting (and related subcomponents)
- Upper End Fitting (and related subcomponents)

These subcomponents are further described below and included in Table 3.2-1.

Fuel Rods (Cladding, End Caps/Plugs)

The fuel rods consist of enriched UO_2 pellets inserted into the cladding tubes. In some older assemblies up to 12 fuel rods are replaced with non-fuel bearing burnable poison rods containing Al_2O_3-B_4C pellets and Zircaloy-4 pellets. Plug-type end caps are seal welded to each end. The material of construction for the fuel rods (cladding and end caps) currently in storage at the ISFSI is Zircaloy-4. The cladding and end caps confine the fuel pellets and fission gases. Each rod is pressurized with helium during fabrication. A small number of assemblies in storage have up to five solid stainless steel replacement rods, with a maximum exposure $\leq$ 40,000 MDd/MTU, in place of fuel rods.

Guide Tubes

The guide tubes are mechanically attached and secured to the top and bottom end fittings as described in the Calvert Cliffs UFSAR (Chapter 3 of Reference 3.10.11). The material of construction for the guide tubes is Zircaloy-4 for the irradiated fuel assemblies currently in storage at the ISFSI.

Note that Zircaloy-4 is softer than the Inconel 625 cladding on the control element assembly (CEA), which can result in wear of the guide tube as a result of CEA vibration caused by turbulent coolant flow. As a result, stainless steel sleeves have been installed in fuel assembly guide tubes that show wear or which house CEAs.

Spacer Grid Assemblies

The fuel assembly consists of 176 rods (pins) and 5 guide tubes. The pins may contain fuel and/or a neutron poison. The assembly is held together by spacer grids and is closed at the top and bottom by end fittings.

Lateral support and positioning of the fuel rods within an assembly is provided by spacer grids with either cantilever tab springs or I-springs. The spacer grids are welded to the five full-length guide tubes. The guide tubes provide channels which guide the CEAs over their entire length of travel and form the longitudinal structure of the assembly. In selected fuel assemblies the central guide tube houses incore instrumentation.

The fuel rod spacer grids maintain the fuel rod pitch over the length of the rod. The grid provides positive lateral restraint to the fuel rod but only frictional restraint axially. The grids are fabricated from preformed Zircaloy, interlocked in an egg crate fashion, and welded together. The grid supports each fuel rod by two cantilever tab springs or two I-springs and four arches. The springs press the rod against the arches to restrict relative motion between the grids and the fuel rods.

The spring and arch positions are reversed from grid to grid to provide additional movement restrictions. The perimeter strips contain features designed to prevent hang up of grids during a refueling operation. The eight Zircaloy-4 spacer grids are welded to each Zircaloy-4 guide tube at eight locations, four on the upper face of the grid and four on the lower face of the grid, where the spacer strips contact the guide tube surface.

Lower End Fitting (and Related Subcomponents)

The central guide tube inserts into sockets in the upper and lower end fittings and is thus retained laterally by the relatively small clearance at these locations. The upper end fitting socket is created by the center post which is threaded into the lower cast flow plate and tack welded in two places. The lower end fitting socket is machined out of the lower end fitting casting. There is no positive axial connection between the central guide tube and the end fittings.

The lower end fitting consists of an Inconel grid welded to a cast stainless steel plate which has flow holes and four support legs. The support legs also serve as alignment posts. Precision-drilled holes in the support legs mate with four core support plate alignment pins, thereby properly locating the lower end of the fuel assembly.

Upper End Fitting (and Related Subcomponents)

The fuel assembly upper end fitting consists of a 304 stainless steel flow plate, a stainless steel holddown plate, five machined posts, and five Inconel X-750 compression springs. The upper end fitting attaches to the guide tubes to serve as an aligning and lifting device for each fuel assembly. The flow plate is attached to the top ends of the guide tubes and is designed to prevent excessive axial motion of the fuel rods. Inconel X-750 is selected for the compression springs. The holddown plate together with the compression springs comprises the holddown device.

Excluded Irradiated Fuel Assemblies Subcomponents

In addition to the above irradiated fuel assemblies subcomponents, the following irradiated fuel assemblies subcomponents, although in the scope of license renewal, were excluded from further aging management review because they do not support or impact the intended function of the irradiated fuel assemblies during the renewed license (extended storage) period:

- Control rods
- Fuel pellets and other fuel rod internals

Also, the holddown spring and retainer were determined not to provide or impact an intended function, but were conservatively included in the aging management review.

3.2.2 IRRADIATED FUEL ASSEMBLIES MATERIALS EVALUATED

The materials of construction for the subcomponents of each irradiated fuel assemblies that are subject to aging management review are zirconium-based alloy (Zircaloy-4), stainless steel, and nickel-based alloy (Inconel).

3.2.3 ENVIRONMENTS FOR THE IRRADIATED FUEL ASSEMBLIES

The environments normally experienced by the irradiated fuel assembles while in the DSCs are described below.

External Environment

For irradiated fuel assemblies, external environment refers to the internal DSC atmosphere. For purposes of this evaluation, the storage atmosphere was assumed to be predominantly helium with trace amounts of water vapor, fission product gases, and air.

Additionally, residual boron may coat the irradiated fuel assemblies surfaces since they were exposed to a borated water environment in the SFP prior to storage. Any boric acid residue remaining on the irradiated fuel assemblies will have no deleterious effects due to the minimal amount of water present in the storage atmosphere and the materials of construction for the irradiated fuel assemblies.

Following initial cask loading, the temperature of the fuel cladding was limited to 635°F for normal storage conditions (Table 1.2-1 of Reference 3.10.9). Fuel cladding temperature will then decrease over time while in dry storage.

Internal Environment

For irradiated fuel assemblies, internal environment refers to the fuel rod interior. The fuel rods were pressurized with helium during manufacturing. For purposes of this evaluation, the fuel rod internal environment is assumed to be a combination of the original helium fill gas and fission products produced during reactor operation.

3.2.4 TIME-LIMITED AGING ANALYSIS FOR FUEL CLADDING AND MAXIMUM TEMPERATURES

AREVA Technical Report 10955-0101 (see Appendix B) was prepared to identify and evaluate the effect of time-limited aging, to demonstrate safe operation over the extended service life of the ISFSI. The report evaluated the irradiated fuel cladding and maximum temperatures when stored in NUHOMS-24P and NUHOMS-32P DSCs. Appendix B describes the finding of the TLAA.

As described in Appendix B no further analysis of the cladding thickness or maximum temperature is necessary and the irradiated fuel assembly cladding stored in the NUHOMS-24P and NUHOMS-32P DSCs will adequately perform for an additional 40 years of service.

3.2.5 AGING EFFECTS REQUIRING MANAGEMENT FOR THE IRRADIATED FUEL ASSEMBLIES

Possible fuel and cladding degradation mechanisms were discussed in Subsection 3.1.3. The aging management review was performed by combining this information with the materials of construction identified in Subsection 3.2.2 and the environmental conditions discussed in Subsection 3.2.3. As discussed in Subsection 3.1.3 irradiated fuel assemblies with burnup of less than 45 GWd/MTU is not susceptible to hydrogen embrittlement and radial hydride formation aging mechanisms. Irradiated fuel assemblies with burnup greater than 45 GWd/MTU minimize the likelihood of these aging degradation mechanisms by limiting peak cladding temperature.

3.2.6 AGING MANAGEMENT ACTIVITIES FOR THE IRRADIATED FUEL ASSEMBLIES

There are no aging effects requiring management for the irradiated fuel assemblies. Therefore, no aging management program or activities are credited during the renewed license period for irradiated fuel assemblies subcomponents.

3.3 AGING MANAGEMENT REVIEW RESULTS – DSC

This section provides the results of the aging management review of the DSCs which were determined to be in the scope of license renewal as identified in Section 2.3.

A summary of the results of the aging management review for the DSC subcomponents is provided in Table 3.3-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were, therefore, not subjected to further aging management review.

A description of the DSC subcomponents that support an SSC intended function is provided in Subsection 3.3.1, and a summary of the materials and environments for the DSCs is provided in Subsection 3.3.2 and Subsection 3.3.3, respectively. Subsection 3.3.4 references the TLAA for the DSCs while being stored in the HSMs. Subsections 3.3.5 and 3.3.7 provide a discussion of the aging effects requiring management for the applicable DSC subcomponents, if any, and any aging management activities used to manage the effects of aging, respectively.

3.3.1 DESCRIPTION OF DSC SUBCOMPONENTS

Each DSC serves as the confinement vessel during transfer of irradiated fuel assemblies to and from a HSM, as well as during storage of the irradiated fuel assemblies in a HSM. The shielded end plugs provide biological shielding during closure and transfer operations, and also provide shielding at the front access to the HSM. A single NUHOMS-24P DSC is sized to hold 24 PWR fuel assemblies and a single NUHOMS-32P DSC is sized hold 32 PWR fuel assemblies.

Table 1.3-1 in the ISFSI USAR (Reference 3.10.9) lists the major systems, subsystems, and components of the ISFSI. The subcomponents listed in ISFSI USAR Table 1.3-1 are discussed in this section for the DSCs. A description of each subcomponent is as follows:

DSC Basket

The major difference between the NUHOMS-32P DSC and the NUHOMS-24P DSC is the internal basket assembly. Each NUHOMS-24P DSC contains 24 PWR fuel assemblies and each NUHOMS-32P DSC contains 32 PWR fuel assemblies.

The design of the generic NUHOMS-24P DSC is described in detail in Reference 3.10.10. The NUHOMS-24P DSC is very similar to the referenced design with revisions as necessary to accommodate a slightly different fuel assembly design. The main component of construction of the DSC is a stainless steel cylindrical containment vessel.

The internal basket assembly is comprised of 24 guide sleeves supported by spacer disks at intervals corresponding, for the most part, to the fuel assembly spacer grids. For a few of the fuel assemblies, the spacer grids were found not to be in complete alignment with the NUHOMS-24P DSC spacer disks. Such misalignments were evaluated structurally, and found to be able to withstand normal and cask drop loads. Support rods maintain the spacer disk location. All canister structural components are fabricated from type 304 stainless steel, except the spacer disks and support rods may be fabricated from aluminum coated carbon steel. Lead gamma shielding is used in both the top and bottom end shield plugs (Section 1.3.1, Reference 3.10.9).

The principal differences between the NUHOMS-24P DSC and the generic DSC design are: the addition of one spacer disk for a total of nine to accommodate the Calvert Cliffs fuel which has nine spacer grids; thinner spacer disks with wider ligaments; an additional ½ inch of lead in both shield plugs; and a shorter overall length accounting for the shorter fuel assembly design.

Criticality safety for the NUHOMS-24P DSC, during wet loading operations, is maintained through the geometric separation of the fuel assemblies within the internal basket assembly, the inherent neutron absorption capability of the stainless steel guide sleeves, and the proper selection of sufficiently depleted fuel assemblies.

The NUHOMS-32P DSC design increases the number of stainless steel guide sleeves to 32 (one for each spent fuel assembly) and uses an egg-crate design made of stainless steel and aluminum (borated and unborated plates) to support the guide sleeves. Both the guide sleeves and the egg-crate components run the full length of the DSC cavity. This allows the guide sleeves to be in contact with the egg-crate components over the whole length of the DSC cavity versus only at spacer discs in the NUHOMS-24P DSC design. In the NUHOMS-32P DSC basket design full-length stainless steel rails and aluminum rail inserts between the DSC stainless steel cylindrical shell and the outside guide sleeves were added. As with the NUHOMS-24P DSC design, the basket assembly is not attached to the DSC shell walls or cover plates.

Other differences are the relocation of the vent and siphon ports. They have been moved from the DSC shell wall (in NUHOMS-24P DSC) to the DSC top shield plug (in NUHOMS-32P DSC) to improve the welding, blowdown, and vacuum drying operations. The NUHOMS-32P DSC lifting fixture device is different than the lifting eyes of the NUHOMS-24P DSC. The top shield plug for the NUHOMS-32P DSC is different from the NUHOMS-24P DSC top shield plug. The design change to the top shield plug involves a reduction in the lead shield thickness, and an increase in the outer steel plate thicknesses.

Criticality safety for the NUHOMS-32P DSC is maintained through fixed neutron absorbers in the NUHOMS-32P DSC basket and a soluble boron concentration in the SFP water of 2,450 ppm.

DSC Shell with Bottom Shield Plug

Lead gamma shielding is used in both the top and bottom end shield plugs. The principal differences between the Calvert Cliffs NUHOMS-24P DSC and the generic DSC design is an additional ½ inch of lead in both shield plugs (Reference 3.10.10).

For the NUHOMS-32P DSC the bottom shield plug has a minimum thickness of 4.25 inches. The bottom lead casing plate is welded to the lead casing shell plate by a ¼ inch groove weld. The lead casing shell is welded to the bottom cover plate by a ¼ inch groove weld on the outside, and a 1/8 inch fillet weld on the inside. These welds carry the same loads as the ¼ inch groove weld between the lead liner plate and the lead casing shell and are acceptable. The 5/16 inch bottom weld applied to the plug post and bottom cover plate interface is a non-structural weld.

Shield Plug (Top)

The top shield plug serves to protect operating personnel during the DSC drying and sealing operations. One of the principal differences between the Calvert Cliffs NUHOMS-24P DSC and the generic DSC design is that an additional ½ inches of lead is in both shield plugs (top and bottom). When the identity of all fuel assemblies in the DSC has been verified, the shield plug assembly is positioned over the DSC, and lowered until it is properly seated.

Lead gamma shielding is used in both the top and bottom end shield plugs. The DSC shield plug and cover plate assemblies are independently seal welded to assure long-term confinement of the irradiated fuel.

The top shield plug for the NUHOMS-32P DSC is different than the NUHOMS-24P DSC top shield plug. The design change to the top shield plug involves a reduction in the lead shield thickness, and an increase in the outer steel plate thicknesses. Another difference is that the NUHOMS-32P DSC top shield plug has vent and siphon ports attached where the NUHOMS-24P DSC top shield plug does not.

Cover Plates (Top and Bottom)

The DSC shield plug and cover plate assemblies are independently seal welded to assure long-term confinement of the irradiated fuel. The top cover plate provides a redundant confinement boundary. It is placed on top of the top shield plug after the drying and helium backfill operations are complete, and welded to the DSC shell. The top cover plate is a stainless steel plate.

The bottom cover plate is installed during fabrication of the DSC and welded to the DSC shell. The bottom end structural weld is a full penetration weld and is 100% radiographed. The cover plates are stainless steel plate.

Siphon and Vents Ports

As mentioned, the vent and siphon ports for the NUHOMS-24P DSC and the NUHOMS-32P DSC are at different locations. They have been moved from the DSC shell wall (in NUHOMS-24P DSC) to the DSC top shield plug (in NUHOMS-32P DSC) to improve the welding, blowdown, and vacuum drying operations. Regardless of which DSC is used, prefabricated plugs are seal welded over the DSC vent and siphon port openings. The plates welded to the vent and siphon ports form part of the pressure boundary. For the NUHOMS-32P DSC the plates are 3.40 inches in diameter, and are welded to the drain and fill

ports by 3/16 inch groove welds. The NUHOMS-32P DSC vent/siphon port tube inside diameter is 1.05 inches.

Ram Grapple Ring

The hydraulic ram system includes a grapple at the end of the piston which is used to engage a grapple ring on the DSC for retrieval operations. The grapple ring is welded to the grapple ring shell by a full penetration groove weld. These components are stainless steel.

For the NUHOMS-32P DSC the grapple ring assembly plate is welded to the grapple ring shell on the inside by a 3/8 inch groove weld with a 1/8 inch fillet weld, and on the outside by a ¼ inch fillet weld. These welds carry some of the lead shield during a vertical lift. The ¼ inch groove weld is adequate under this loading condition. The grapple ring shell is welded to the inner cover plate of the DSC by a full penetration weld with a 1/8 inch fillet cover weld.

A 1/8 inch cover fillet weld over a 3/8 inch groove weld is applied to the outside of the grapple ring shell and the bottom cover plate interface. Conservatively ignoring the cover fillet weld, the 3/8 inch groove weld is loaded by a maximum of 95,000 lbs when the DSC is pushed by the ram assembly.

3.3.2 DSC MATERIALS EVALUATED

The materials of construction for DSC subcomponents that are subject to further aging management review include stainless steel, carbon steel, aluminum, and lead. Borated aluminum alloy is considered to perform similar to aluminum from an aging management perspective based on qualification testing discussed in Reference 3.10.13 and comparison between borated aluminum and unborated aluminum (Boral wrapper) in a low pH aqueous environment in Reference 3.10.14. The material type of individual DSC subcomponents is identified in Table 3.3-1.

3.3.3 ENVIRONMENTS FOR THE DSCs

The environments that affect the subcomponents of each DSC, both externally and internally, are those that are normally experienced and are described below.

External

Each DSC is positioned for long-term storage inside an HSM. As such, the external surface of each DSC is exposed to the same environment, including neutron fluence and integrated gamma dose, described in Section 3.4.3 for the interior of an HSM – that is, a sheltered environment which is protected from precipitation and wetting. The normal operating temperature of the outside DSC surface is highest at the top of the cylinder and was expected to be 247°F for the NUHOMS-24P and 292°F for the NUHOMS-32P (for 70°F ambient air) during the original license term. These surface temperatures are conservatively extended into the license renewal period.

Internal

Design temperatures of 400°F and 460°F were used for the NUHOMS-24P and NUHOMS-32P DSCs, respectively, for the internal structures in the aging management review. The heat generated in the fuel regions of the irradiated fuel assemblies inside a sealed DSC is transferred towards the canister shell by conduction, convection, and radiation. Helium is present in the canister to facilitate the conduction and convection. A parametric study of temperature versus time has shown that the fuel temperature and; therefore, the helium temperature and DSC internal structure temperature, will decrease over time (Reference 3.10.10). For conditions of 70°F ambient temperatures, the normal expected helium temperatures during the original license term was determined to be 362°F for NUHOMS-24P, and an

average gas temperature of 484°F for NUHOMS-32P. In the case of the NUHOMS-32P, the use of 460°F is appropriate because only the fatigue analysis of the DSC shell and shield plugs use temperature as an input, and temperatures of those components are much lower than the average helium gas temperature.

After 20 years of dry storage, the fast neutron flux and gamma radiation doses are expected to be on the order of 1 x 10^{14} neutrons/cm^2 and 1x10^9 Rads respectively (Reference 3.10.2). In Table 3.3-1, the helium inside a DSC is listed as an air and gas environment (air and gas environment is helium inside DSC cavity, with possible trace amounts of air, water vapor and fission product gases).

3.3.4 DSC TIME-LIMITED AGING ANALYSIS

AREVA Technical Report 10955-0101 (see Appendix B) was prepared to identify and evaluate the effect of time-limited aging, to demonstrate safe operation over the extended service life of the ISFSI. The report evaluated the DSCs stored in the HSMs. Appendix B describes the finding of the TLAA.

Appendix B concludes, the stored NUHOMS-24P and NUHOMS-32P DSCs will adequately perform their functions for an additional 40 years of service.

3.3.5 AGING EFFECTS REQUIRING MANAGEMENT FOR THE DSCs

Based on a review of the DSC materials of construction and the environments (e.g., relevant conditions and stressors) experienced during extended ISFSI storage, there are no aging effects requiring management during the renewed license period for the subject DSC subcomponents.

There are no aging effects requiring management for the carbon steel, stainless steel, aluminum, or the lead subcomponents of the DSCs because of the durable construction, double sealwelded closure, and the environments to which each DSC is exposed. Each DSC was sealed during the original license term to contain 24 or 32 appropriately aged PWR fuel assemblies and an inert (helium) environment. Each sealed DSC was then placed into an HSM and, thereafter, exposed to only a relatively mild sheltered external environment. In addition, a continued decrease of irradiated fuel assemblies temperatures and radiation levels are expected over time.

The operating experience discussed in Subsection 3.1.5 is applicable to the DSCs, which together with the HSMs serve as dry storage for the irradiated fuel assemblies. Aging management review results for individual DSC subcomponents are shown in Table 3.3-1.

3.3.6 AGING MANAGEMENT ACTIVITIES FOR THE DSCs

There are no aging effects requiring management for the DSCs. Therefore there are no aging management programs or activities required for DSC subcomponents during the renewed license period.

3.4 AGING MANAGEMENT REVIEW RESULTS – HSM

This section provides the results of the aging management review of the HSMs which were determined to be in the scope of license renewal as identified in Section 2.3.

A summary of the results of the aging management review for the HSM subcomponents is provided in Table 3.4-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would

not compromise, the SSC intended function and were; therefore, not subjected to further aging management review.

A description of the HSM subcomponents that support an SSC intended function is provided in Subsection 3.4.1, and a summary of the materials and environments for the HSMs is provided in Subsection 3.4.2 and Subsection 3.4.3, respectively. Subsection 3.4.4 references the TLAA for the HSMs. Subsection 3.4.5 and Subsection 3.4.6 provide a discussion of the aging effects requiring management for the applicable HSM subcomponents and the aging management activities used to manage the effects of aging, respectively.

3.4.1 DESCRIPTION OF THE HSM SUBCOMPONENTS

The HSMs provide a unitized, low profile, modular storage location for irradiated fuel. Each HSM structure is constructed from reinforced concrete and structural steel. The thick concrete walls and roof of the HSM provide neutron and gamma shielding. Inlets and outlets, and associated pathways, are provided in each HSM for the dissipation of decay heat. The HSM, which consists of the concrete shielding, the DSC support assembly, and the foundation is considered important to safety because it protects the DSC during storage. The concrete HSM is designed in accordance with American Concrete Institute 349-85 and the level of testing, inspection, and documentation provided during construction and maintenance is in accordance with the Calvert Cliffs Nuclear Power Plant Quality Assurance program. The HSM walls, roof, and foundation are connected by reinforcing steel.

The approach slabs in front of the HSMs are constructed separately from the HSM foundations. The transfer system is designed to accommodate any credible differential settling between the two slabs. The approach slabs and HSM foundations have been designed to minimize differential settlement over the life of the facility.

Reference 3.10.9, Table 1.3-1 lists the major systems, subsystems, and components of the HSM. The subcomponents listed in Reference 3.10.9, Table 1.3-1 are discussed in this section for the HSM. A description of each subcomponent is as follows:

Reinforced Concrete Walls, Roof, and Foundation

Above the foundation, concrete and reinforcing rebar form the roof slab, outer walls, and the between-module walls for each HSM. An air inlet pre-cast slab is included in each HSM. This slab is supported on the front wall and an interior wall in each HSM. The concrete walls form interconnected sub-units of modules which are six wide and two back-to-back to form a 2x6 array. Each array of 12 HSMs is constructed on a common reinforced concrete foundation slab. The foundations of the HSM are established in soil which will not undergo reduction in strength or increased settlement under design basis earthquake conditions.

Three foot thick end walls provide shielding on the sides of each HSM array. The front walls of the HSMs are 3-1/2 feet thick. Two foot thick interior common walls provide shielding between modules to help mitigate radiation levels in adjacent modules during the DSC loading and retrieval process. The roof slab for the HSMs is 3 feet thick. An internal slab and roof caps are provided to shield the ventilation inlet and outlet openings. A concrete shielding cap is located over each outlet.

The concrete codes for design and construction are included in Reference 3.10.9.

DSC Structural Steel Support Assembly

The DSC is designed to slide from the transfer cask into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces. This is accomplished by the addition of Nitronic 60 hard sliding rails to the transfer cask and HSM. The HSM and cask rails are coated with dry film lubricants.

The DSC support assembly in the HSMs, consists of 8x40 support rail members with 8x48 cross members. A ¾ inch x 6 inch rail cover plate runs the full length of the support rail members and extends into the HSM access opening. The support rail members are supported by structural embedments at the HSM front wall and by the cross members at the center and rear of the module.

DSC Seismic Retainer

The seismic restraint weighs less than 30 pounds and is not equipped with a handle. After placement of the DSC into the HSM, a removable seismic restraint is attached to the DSC at the front of the HSM.

Cask Docking Flange and Tie-Down Restraints

The HSM access opening sleeve has a stepped flange sized to facilitate docking of the transfer cask for DSC transfer. This configuration was chosen to minimize streaming of radiation through the HSM opening during DSC transfer. During DSC loading/unloading operations, the transfer cask is docked in the HSM door opening recess and mechanically secured to tie-down embedments provided in the front wall of the HSM. The tie-down embedments are equally spaced above and below the HSM horizontal center line on either side of the HSM access opening. The HSM embedments are designed as conventional embedded hooks in accordance with the American Concrete Institute 349-85 Code. The transfer cask restraint system is designed for loads which occur during normal DSC transfer operations and during an off-normal jammed DSC event (Reference 3.10.10).

Heat Shield

The HSM heat shield and attachment details are identical to those of the generic design except the roof attachment bolts are smaller diameter. The heat shield analysis presented in Section 8.1.1.7 of Reference 3.10.10 is directly applicable to the Calvert Cliffs design except that maximum heat shield bending stress is 4.4 ksi and the maximum side wall bolt bending stress is 14.6 ksi. This thermal radiation shield is used to reduce the HSM roof temperature to within acceptable limits for all conditions.

Shielded Front Access Door and Door Supports

The HSM access door consists of a 1-3/4 inch steel plate, 10-3/4 inches of concrete shielding material, and a ¼ inch steel cover plate. Steel angle sections are attached to embedded plates in the HSM front wall to form guides to slide the HSM door vertically. The door support frame is welded to embedded plates with stud anchors.

The HSM steel doors are tack welded after insertion of a loaded DSC. The HSM steel doors weigh approximately 6 tons and require heavy lifting equipment for removal.

Ventilation Air Openings (One Inlet, Two Outlets)

Two HSM air outlet vents are located on the roof of each HSM. A concrete shielding cap is located over each outlet. Irradiated fuel assemblies confined in storage are cooled by conduction, convection, and radiation within the DSC, and conduction, convection, and radiation from the DSC surface (see

Section 3.3.3). An air inlet near the bottom of the HSM front wall and outlets in the HSM roof allow convective cooling by natural circulation.

No auxiliary ventilation is used or required at the ISFSI. No air sampling systems are required at the ISFSI.

Shielded Ventilation Air Inlet Plenum

The air inlet plenum is in a 3 feet 6 inch thick concrete block. The air inlet plenum is 1 foot high by 6 feet wide and passes through the concrete block via a sloped (incline) path. On the inside of the HSM, 5 feet from the air inlet plenum opening, is a 1-1/2 foot long by 2-1/2 foot high concrete wall. This shield has a 6 inch opening that runs along its top. A precast slab rests on top of the shield and is tied into the HSM front wall. Therefore, there is not a direct line of sight through the air inlet plenum. There is an intrusion screen that covers the ventilation air inlet plenum.

Ventilation Air Outlet Shielding Blocks

At the 2 foot thick back wall that is between the HSMs there is a 6 inch by 6 foot opening (slot) in the roof of the HSM that forms the back ventilation air outlet plenum. There is a similar opening in the roof at the front of the HSM which forms the front ventilation air outlet of the HSM. These openings are covered with a 1-1/2 foot concrete slab that is 1-1/2 foot above the roof opening. This slab is supported on three sides with concrete walls that are at least 9 inches thick. The ventilation air outlet shielding blocks form a labyrinth so that there is not a direct line of sight through the ventilation air outlet plenum. There is an intrusion screen that covers the ventilation air outlet plenums on the outside of the HSM.

Lightning Protection System

Should lightning strike an HSM, the normal operation of the HSM will not be affected since the current discharged by the lightning will follow the low impedance path to ground provided by the lightning protection system. Therefore, the DSC and HSM will not be damaged by heat or mechanical forces generated by an electrical current passing through the higher impedance concrete. Since the HSMs require no electrical equipment for continued operation, other than for lighting and security systems, the resulting current surge from a lightning strike will not affect the normal operation of the ISFSI.

Other Subcomponents

Threaded Fasteners and Expansion Anchors

Stainless steel HILTI KWIK bolts are use in the HSMs.

Dry Film Lubricant

The lubricant for DSC support assembly is used only for reducing friction while sliding a DSC along the support assembly inside an HSM. Once the DSC is in place within the HSM, the lubricant performs no function during storage of the DSC.

Excluded Subcomponents

The following subcomponents were excluded from aging management review because they do not support or impact the intended function of the HSM during the renewed license period:

- Polyvinyl chloride pipe (drain and electrical conduit)
- Ladder and attachments
- Alignment targets

- Caulk
- Lubricants for DSC support structure rails
- Clips for attachment of lightning protection system cables
- Electrical conduit, boxes, and cable attached to the HSMs

3.4.2 HSM MATERIAL EVALUATED

The materials of construction for HSM subcomponents that are subject to further aging management review include carbon steel, stainless steel, copper, bronze, concrete, and carbon steel rebar. The material type of the individual HSM subcomponents is identified in Table 3.4-1.

3.4.3 ENVIRONMENTS FOR THE HSMs

The environments that are normally experienced by HSM subcomponents are described below:

The HSMs are located outdoors at the Calvert Cliffs ISFSI site near Lusby, MD. The environment for the HSMs is bounded by the extreme winter or summer conditions with an ambient temperature range of -3°F to 103°F. For the HSM, the vents were assumed to be open. A solar heat flux of 127 BTUH/ft^2 is conservatively included for the 103°F ambient temperature to maximize the HSM roof concrete surface temperatures. This condition is assumed to occur for a sufficient period of time such that steady-state conditions are achieved.

Inside each HSM, considered to be a sheltered environment with no air conditioning, subcomponents are protected from outdoor effects (e.g., precipitation), but do experience higher temperatures and radiation in an air environment. Since each HSM may store either a NUHOMS-24P DSC or a NUHOMS-32P DSC, the impact on the HSM from each type of DSC must be evaluated. For the NUHOMS-32P DSC, the current ISFSI Technical Specifications limits the maximum allowed fuel burnup to not exceed 47 GWd/MTU. However, Calvert Cliffs has just received approval that will allow the maximum allowed fuel burnup limit that can be stored in a NUHOMS-32P DSC to increase to 52 GWd/MTU. As a result all source-term values are evaluated below.

For a HSM containing a NUHOMS-24P DSC the expected gamma energy fluence deposited in the HSM concrete is 6.8 x 10^{10}MeV/cm^2-sec (Reference 3.10.10, Page 8.1.20). A gamma energy fluence at this level will result in a negligible temperature rise in the HSM concrete. Similarly, the accumulated neutron fluence for the HSM with a NUHOMS-24P DSC is 1.2 x 10^{14} neutrons/cm^2 in 50 years (Reference 3.10.10, Page 8.1-20), which linearly extrapolates out to 1.44 x 10^{14} neutrons /cm^2 in 60 years. Reference 3.10.10 indicates that the compressive strength and modulus of elasticity of concrete is not affected by a neutron fluence of 1.2 x 10^{14} neutrons /cm^2. The extrapolated neutron fluence of 1.44 x 10^{14} in this case is only marginally higher and will have no significant impact on the strength and elasticity of the concrete.

A similar evaluation was performed for a NUHOMS-32P DSC, based on the 47 GWd/MTU limit, in Section 12.8.1.1.5 of Reference 3.10.9. It was based on the original evaluation for the NUHOMS-24P DSC and showed that the expected gamma fluence was less than the gamma fluence value determined for the NUHOMS-24P DSC. As a result there will be a negligible temperature rise experienced in the HSM concrete. The evaluation also demonstrated that the neutron fluence emitted from the NUHOMS-32P DSC will be at levels below the threshold value that would result in neutron induced degradation of the concrete.

Calculations supporting the use of the higher burnup limit of 52 GWd/MTU for the NUHOMS-32P DSC demonstrated the gamma energy fluence from the higher burnup NUHOMS-32P DSC would also be less

than the gamma energy fluence calculated in the NUHOMS-24P DSC. As a result, there will only be a negligible rise in concrete temperature. The calculation further demonstrated the neutron fluence level will remain well below the threshold value that would cause neutron induced degradation of HSM concrete.

Therefore the evaluations show that, regardless of the DSC used, there will be negligible impact on the HSMs concrete and they will not negatively impact the HSMs ability to perform their intended function for the duration of the license renewal period.

3.4.4 HSM TIME-LIMITED AGING ANALYSIS

AREVA Technical Report 10955-0101 (see Appendix B) was prepared to identify and evaluate the effect of time-limited aging, to demonstrate safe operation over the extended service life of the HSM. Appendix B describes the finding of the TLAA.

Appendix B concludes the HSMs will adequately perform for additional 40 years of service.

3.4.5 AGING EFFECTS REQUIRING MANAGEMENT FOR THE HSMs

This section describes the aging effects that could, if left unmanaged, cause degradation of HSM subcomponents and result in loss of the SSC intended function(s) during the renewed license period. The aging management review results for individual HSM subcomponents are reflected in Table 3.4-1.

Based on the HSM material and environment combinations and consideration of the conditions during extended storage, the following aging effect was determined to require management for the applicable metal subcomponents of the HSMs:

Loss of Material due to General Corrosion and Pitting - Carbon steel subcomponents of the HSMs (Yard Environment and Sheltered Environment) and bronze subcomponents of the lightning protection system (Yard Environment).

The review of industry and plant-specific operating experience discussed in Subsection 3.1.5 supported the management of the above aging effects, but did not identify any other aging effects for an HSM during an extended service life. With respect to concrete and reinforcing steel, no aging effects/mechanisms were identified that require management for the above-grade concrete. However, the NRC previously indicated its expectation that licensees for plant (10 CFR Part 54) license renewal should include concrete in an aging management program for the period of extended operation because loss of material, cracking, and change in material properties are plausible and applicable aging effects for above ground concrete structures and should be managed. Therefore, to meet the current NRC position (Reference 3.10.12), the condition of accessible HSM concrete that is above grade is conservatively considered to require aging management.

3.4.6 AGING MANAGEMENT ACTIVITIES FOR THE HSMs

The activities associated with the HSM aging management program, when continued in the renewed license period, will manage the aging effects for the steel portion of the HSMs, and will include the conservative evaluation of the condition of accessible concrete for the HSM subcomponents identified in Table 3.4-1.

A description of this aging management program is provided in Appendix A, along with the demonstration that the aging will be effectively managed during the renewed license period.

3.5 AGING MANAGEMENT REVIEW RESULTS – TRANSFER CASK

This section provides the results of the aging management review of the transfer cask which was determined to be in the scope of license renewal as identified in Section 2.3.

A summary of the results of the aging management review for the transfer cask subcomponents are provided in Table 3.5-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and therefore, were not subjected to further aging management review.

A description of the transfer cask subcomponents that support an SSC intended function is provided in Subsection 3.5.1, and a summary of the materials and environments for the transfer cask is provided in Subsection 3.5.2 and Subsection 3.5.3, respectively. Subsection 3.5.4 references a TLAA for the transfer cask. Subsections 3.5.5 and 3.5.6, respectively, provide a discussion of the aging effects requiring management for the applicable transfer cask subcomponents and the aging management activities used to manage the effects of aging.

3.5.1 DESCRIPTION OF THE TRANSFER CASK SUBCOMPONENTS

As discussed in the ISFSI USAR (Reference 3.10.9, Section 1.3.1.3), the transfer cask provides radiological shielding during the DSC closure operations and during transfer of the DSC to the HSM. To ensure structural integrity, the transfer cask also provides protection of the DSC against potential natural and operational hazards during transport and transfer of the DSC to the HSM. Both solid neutron and lead gamma shielding are incorporated into the transfer cask design.

The major transfer cask components are divided into sub-components that are included in Table 3.5-1, along with the particular function the individual sub-component performs to support the overall cask intended functions. A summary of those cask subcomponents is provided below:

Cask Body

The transfer cask body is a cylindrical shape which is enclosed on one end and consists of a structural shell and an inner shell. The structural shell assembly is comprised of a carbon steel cylinder that is welded at the top to the forged stainless steel top flange. At the other end, it is welded to the cask bottom which is comprised of a forged stainless steel bottom ring and a stainless steel bottom plate. The inner shell is comprised of a stainless steel cylinder that is welded to the forged stainless steel top flange on one end and to the forged stainless steel bottom ring at the other end. The cavity between the structural shell cylinder and the inner shell cylinder is filled with NS-3 to serve as the radiological or biological shield for the transfer cask.

Cask Attachments

The transfer cask has several attachments to the cask body. Two parallel rails (Nitronic 60) are welded to the inner shell to facilitate sliding of the DSC as it is transferred to the HSM. Two upper trunnion assemblies (carbon steel, stainless steel, Inconel, and BISCO NS-3) are located near the top of the cask and are used for lifting and moving the transfer cask in the Auxiliary Building. Two lower trunnion assemblies (carbon steel, stainless steel, Inconel, and BISCO NS-3) are located near the base of the cask and are used for support and rotation of the transfer cask during movement to the transfer cask support skid.

Cask Penetrations

The transfer cask has several penetrations to the cask body. Located on the bottom plate (stainless steel, BISCO NS-3) is the ram access penetration ring (stainless steel) which allows access to the DSC for transfer operations. Additionally a penetration for the annulus drain valve is located on the bottom support ring (stainless steel) and a similar penetration for the annulus fill valve is located on the top flange (stainless steel).

Cask Neutron Shield

A neutron shield jacket surrounds the cask structural shell, extending axially from the bottom ring to the top flange. It is comprised of an outer stainless steel jacket, top and bottom support rings, and axially spaced support angles. The neutron shielding material is BISCO NS-3. Relief valves are provided to protect the neutron shield jacket from any internal pressurization.

Additionally, BISCO NS-3 is included for neutron shielding in the upper and lower trunnions, cask bottom, ram access cover plate, top cover assembly, inner plug assembly, and outer plug assembly. Relief valves are provided to protect the bottom end plate from any internal pressurization.

Cask Cover Plates and Accessories

The cask is provided with a top cover assembly which is bolted to the cask body during transport of a fuel-loaded DSC from the fuel building to the HSM. Lifting eyes are provided to allow removal from the transfer cask when it is in either a horizontal or vertical orientation.

Threaded Fasteners

Threaded fasteners are used in the following locations for the transfer cask:

- Top cover assembly to top flange connections
- Annulus drain valve cover plate
- Bottom cover assembly to ram access penetration ring connections
- Two guide pins provide closure alignment and orientation for the top cover assembly
- Top cover eyebolts

Excluded Subcomponents

The following subcomponents were excluded-from aging management review because they do not support or impact the intended function of the transfer cask during the renewed license period:

- Lubricants, sealants
- Eyebolts for bottom access plate and top cover plate
- Cover plates, gaskets, and screws for annulus fill/drain valves
- Swagelok relief valves for solid neutron shield
- Polyethylene caps for Swagelok valves
- Alignment pin for top cover plate
- Stick-on alignment targets

3.5.2 TRANSFER CASK MATERIALS EVALUATED

The materials of construction for the individual subcomponents of the transfer cask that are subject to aging management review are listed in Table 3.5-1 and include carbon steel, stainless steel, Inconel, ethylene propylene, lead, and BISCO NS-3.

3.5.3 ENVIRONMENT FOR THE TRANSFER CASK

The exterior of the transfer cask is exposed to borated water during fuel loading while the transfer cask is in the SFP, and to demineralized water in the annulus between the DSC and inner cavity wall of the transfer cask. Following fuel loading into the DSC, the transfer cask is removed from the SFP. Then, the external surfaces are decontaminated and rinsed with demineralized water. The annulus water is removed following welding of the lid(s) to the DSC body, purging of the water in the DSC, vacuum drying and backfilling with helium.

During transfer to and loading operations at the ISFSI, the transfer cask is briefly exposed to outside, ambient conditions.

The brief exposure of the transfer cask to the borated and demineralized water while in the Auxiliary Building, and to the outside environment during transfer/loading operations, does not contribute to the aging of the transfer cask materials during the renewal period. It is the prolonged or frequently recurring exposure to environmental conditions and stresses that must be evaluated for aging effects, such as those encountered during storage or staging prior to use for DSC transfers.

The environment to which the transfer cask is exposed during storage or staging prior to and between infrequent use for DSC transfers is sheltered.

Additionally, the transfer cask experiences radiation exposure during DSC transfers.

3.5.4 TRANSFER CASK TIME-LIMITED AGING ANALYSIS

AREVA Technical Report 10955-0101 (see Appendix B) was prepared to identify and evaluate the effects of time-limited aging to demonstrate safe operation over the extended service life of the ISFSI. The report evaluated the transfer cask. A fatigue evaluation of the transfer cask and transfer cask trunnions was required and is also contained in Appendix B.

The fatigue evaluation of transfer cask and transfer cask trunnions shows that the transfer cask operations do not pose potential risk of fatigue failure of cask, trunnion, or trunnion sleeve throughout the planned 60-year service time.

Appendix B concludes the transfer cask will adequately perform its intended function for the additional 40 years of service.

3.5.5 AGING EFFECTS REQUIRING MANAGEMENT FOR THE TRANSFER CASK

Because of the durable steel construction and relatively mild environments to which the transfer cask subcomponents are normally exposed during staging prior to and between infrequent ISFSI transfers, only the carbon steel subcomponents requires aging management.

The structural shell of the transfer cask body, bolts/washers associated with the top cover, and the bottom plate are constructed of carbon steel. The transfer cask (including these subcomponents) is stored in a sheltered environment between loadings. Additionally, no credit is taken for the coating on the top cover assembly. Thus the following aging effect was determined to require management during the renewed license period:

- Loss of Material due to General Corrosion and Pitting (Sheltered Environment)

Operating experience described in Subsection 3.1.5 supports age-related degradation primarily in wetted environments. Table 3.5-1 provides a summary listing of the aging effects requiring management and the activity used to manage the effects.

3.5.6 AGING MANAGEMENT ACTIVITIES FOR THE TRANSFER CASK

The transfer cask aging management program is credited with managing either the effect of loss of material, or the relevant conditions that could lead to the onset and propagation of a mechanism leading to loss of material during the renewed license period as identified in Table 3.5-1 for carbon steel subcomponents.

A description of this aging management program is provided in Appendix A, Aging Management Program, along with the demonstration that the identified aging effect will be effectively managed for the renewed license period.

3.6 AGING MANAGEMENT REVIEW RESULTS - TRANSFER CASK LIFTING YOKE

This section provides the results of the aging management review of the transfer cask lifting yoke, which was determined to be in the scope of license renewal as identified in Section 2.3.

A summary of the results of the aging management review for the subcomponents of the transfer cask lifting yoke are provided in Table 3.6-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were; therefore, not subjected to further aging management review.

A description of the transfer cask lifting yoke subcomponents that support an SSC intended function is provided in Subsection 3.6.1. A summary of the materials and environments for the transfer cask lifting yoke is provided in Subsection 3.6.2 and Subsection 3.6.3, respectively. Subsection 3.6.4 references the TLAA for the transfer cask lifting yoke. Subsection 3.6.5 provides a discussion of the aging effects requiring management for the applicable transfer cask lifting yoke. Subsection 3.6.6 provides a discussion of the aging management activities used to manage the effects of aging.

3.6.1 DESCRIPTION OF THE TRANSFER CASK LIFTING YOKE

As described in the ISFSI USAR (Reference 3.10.9, Section 4.7.3.4), the lifting yoke is a special lifting device which provides the means for performing all cask handling operations within the plant's auxiliary building. It is designed to support a loaded transfer cask weighing up to 109.25 tons. A lifting pin connects the spent fuel cask handling crane hook and the transfer cask lifting yoke.

The lifting yoke is designated safety-related since it is in the direct load path of the cask. The codes and standards used to design and fabricate the lifting yoke are presented in the ISFSI USAR (Reference 3.10.9, Section 4.7.4).

The lifting yoke is a passive, open hook design with two parallel lifting beams fabricated from thick, high-strength carbon steel plate material, with a decontaminable coating. It is designed to be compatible with the spent fuel cask handling crane hook and load block. The lifting yoke engages the outer shoulder of the transfer cask lifting trunnions. To facilitate ease of shipment and maintenance, all yoke subcomponent structural connections are bolted.

The transfer cask lifting yoke is a bolted assembly consisting of two parallel lift beams and two open J-hooks. The lift beams and J-hooks are fabricated from carbon steel plate. A stainless steel connecting pin is used to attach the transfer cask lifting yoke to the 150 ton spent fuel cask handling crane hook. Stainless steel sling assemblies are attached to the bottom of the lift beams and are used for remote removal and replacement of the DSC shield plug when the transfer cask is in the cask loading area in the SFP. The transfer cask lifting yoke has bronze bushings and hook bearing plates and the lifting yoke palms are lubricated as needed with Nickel Never Seize or similar compound.

3.6.2 TRANSFER CASK LIFTING YOKE MATERIALS EVALUATED

The materials of construction for the transfer cask lifting yoke that are subject to further aging management review are carbon steel, bronze, and stainless steel. The materials type of individual subcomponents is identified in Table 3.6-1.

3.6.3 ENVIRONMENT FOR THE TRANSFER CASK LIFTING YOKE

The transfer cask lifting yoke is exposed to borated water during operations with the transfer cask in the SFP. After removal of the loaded DSC/transfer cask from the SFP, the transfer cask and the transfer cask lifting yoke are rinsed with demineralized water.

The brief exposure to borated and demineralized water does not contribute to the aging of the materials during the license renewal period. It is the prolonged or frequently recurring exposure to environmental conditions and stresses that must be evaluated for aging effects, such as those encountered during storage or staging prior to use for DSC transfers.

The environment to which the transfer cask lifting yoke is exposed during storage or staging prior to and between infrequent use for DSC transfers is sheltered.

3.6.4 TIME-LIMITED AGING ANALYSIS OF THE TRANSFER CASK LIFTING YOKE

AREVA Technical Report 10955-0201 (see Appendix B) was prepared to identify and evaluate the effects of time-limited aging to demonstrate safe operation over the extended service life of the ISFSI. The report evaluated the transfer cask lifting yoke. A fatigue evaluation of the transfer cask lifting yoke was required and is also contained in Appendix B.

In the case of the transfer cask lifting yoke assembly, Appendix B demonstrates that the structural adequacy against fatigue failure for up to 286 transfer cask loading/unloading operations in the planned 60-year service time.

3.6.5 AGING EFFECTS REQUIRING MANAGEMENT FOR THE TRANSFER CASK LIFTING YOKE

For the carbon steel subcomponents, no credit is taken for coating. The following aging effect could result in loss of component intended function(s), and thus requires management during the renewal period:

Loss of Material Due to General Corrosion and Pitting (Sheltered Environment)

For the stainless steel and bronze subcomponents, no aging effects are identified.

Table 3.6-1 provides a summary listing of the aging effects requiring management and the activity used to manage the effects.

3.6.6 AGING MANAGEMENT ACTIVITIES FOR THE TRANSFER CASK LIFTING YOKE

The transfer cask lifting yoke aging management program is credited with managing the aging effects for the carbon steel. A description of this program is provided in Appendix A, Aging Management Program, along with the demonstration that the identified aging effect will be effectively managed during the renewal period.

Per the plant procedures the slings and lifting yoke are visually inspected and verified to be operable within the last 7 days before they are used. Additionally, pre-operational and periodic tests and preventive maintenance are performed per plant procedures.

A lifting yoke annual inspection is conducted to assure continuing compliance under American National Standards Institute 14.6 has been maintained.

3.7 AGING MANAGEMENT REVIEW RESULTS – CASK SUPPORT PLATFORM

This section provides the results of the aging management review of the cask support platform which was determined to be in the scope of license renewal as identified in Section 2.3.

A summary of the results of the aging management review for the subcomponents of the cask support platform is provided in Table 3.7-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were; therefore, not subjected to further aging management review.

A description of the cask support platform subcomponents that support an SSC intended function is provided in Subsection 3.7.1. A summary of the materials and environments for the cask support platform is provided in Subsections 3.7.2 and 3.7.3, respectively. Subsection 3.7.5 provides a discussion of the aging effects requiring management for the applicable cask support platform subcomponents. Subsection 3.7.6 provides a discussion of the aging management activities used to manage the effects of aging.

3.7.1 DESCRIPTION OF THE CASK SUPPORT PLATFORM

The cask support platform is located in the cask pit area of the Unit 1 SFP. The cask support platform serves to elevate the pit floor surface so that the lifting trunnions on the transfer cask do not interfere with the cask seismic restraint.

The cask support platform is a welded assembly of stainless steel plates that supports the transfer cask. Inside the cask support platform stainless steel shell is Hexcel aluminum honeycomb material. The cask support platform is designed to be removable. It positions the transfer cask at the appropriate elevation for loading irradiated fuel assemblies into the DSC.

The performance of honeycomb material is based on tests conducted in a dry atmosphere. Therefore, the honeycomb material is located in a watertight stainless steel shell (outer shell). This outer shell is sealed, and then pressurized with air. The pressure will be maintained at approximately 10 psi above hydrostatic pressure. With positive pressure in the shell, water will not leak into the cavity if a leak does develop in the shell of the cask support platform. This ensures that the honeycomb material remains dry and

functions under the same conditions under which it was tested. The cask support platform is equipped with a tubing manifold, including a relief valve, pressure gauge, and quick-connect, to maintain and monitor the pressure in the outer shell.

The Hexcel honeycomb does contribute to the overall structural capability relied on for proper support of casks placed on the platform. Therefore the honeycomb material and fabrication associated with its construction (i.e., steel shell, seal welds, etc.) are safety-related. However, the components necessary for providing and maintaining a pressurized compartment are not safety-related. The pressurized air is not required for the cask platform to perform its support function.

No subcomponents of the cask support platform were excluded from aging management review.

3.7.2 CASK SUPPORT PLATFORM MATERIALS EVALUATED

The materials of construction for the cask support platform that are subject to aging management review are stainless steel and aluminum. The material type of the individual subcomponents is identified in Table 3.7-1.

3.7.3 ENVIRONMENTS FOR THE CASK SUPPORT PLATFORM

The cask support platform is continuously exposed to a borated water environment as it remains in the SFP during and between DSC transfers. However, the Hexcel aluminum honeycomb material inside the cask support platform shell is maintained in a compressed air environment.

3.7.4 TIME-LIMITED AGING ANALYSIS OF THE CASK SUPPORT PLATFORM

AREVA Technical Report 10955-0101 (see Appendix B) did not identify the cask support platform as an item that required a TLAA. The aging management of the cask support platform is discussed below.

3.7.5 AGING EFFECTS REQUIRING MANAGEMENT FOR THE CASK SUPPORT PLATFORM

Based on the stainless steel material and borated water environment combination, the following aging effects were determined to require management during the renewed license period:

Loss of Material and Cracking due to Pitting and Stress Corrosion Cracking (Borated Water) - Stainless Steel

Table 3.7-1 provides a summary listing of the cask support platform subcomponents. The review of operating experience discussed in Subsection 3.1.5 supports the management of the above aging effects. No other aging effects were identified for stainless steel components in the SFP.

3.7.6 AGING MANAGEMENT ACTIVITIES FOR THE CASK SUPPORT PLATFORM

The Chemistry Control Program is credited with managing the aging effects for stainless steel in a borated water environment. A description of this program is provided in Appendix A, along with the demonstration that the identified aging effect will be effectively managed during the renewal period.

3.8 AGING MANAGEMENT REVIEW RESULTS – SPENT FUEL CASK HANDLING CRANE

This section provides the results of the aging management review of the spent fuel cask handling crane which was determined to be in the scope of license renewal as identified in Section 2.3. The spent fuel cask handling crane is a 10 CFR Part 50 SSC that is used during the dry cask storage operations in the

Auxiliary Building. Transfer equipment to move the DSCs from the SFP (where they are loaded with the irradiated fuel assemblies) to the HSMs where they are stored interfaces with the spent fuel cask handling crane.

At Calvert Cliffs, heavy loads (loads in excess of 1600 lbs) are prohibited from travel over irradiated fuel assemblies in the SFP unless such loads are handled by a single-failure-proof device. The spent fuel cask handling crane, which is designed in accordance with the single-failure-proof criteria of NUREG-0554 and NUREG-0612, is used to handle heavy loads in the SFP area. The maximum design rated load for the spent fuel cask handling crane is 150/15 ton (150 ton for the main hoist and 15 ton for the auxiliary hoist). Its maximum critical load rating is 125/15 ton (Section 9.7.2.4 of Reference 3.10.11). The spent fuel cask handling crane has an annual inspection and is checked prior to use.

As discussed in Section 3.6, a lifting pin connects the spent fuel cask handling crane hook and the transfer cask lifting yoke.

Since the operation of the spent fuel cask handling crane is controlled by 10 CFR Part 50, no 10 CFR Part 72 aging management review was performed for the spent fuel cask handling crane. While the current operating licenses of the Calvert Cliffs Units will expire (2034 and 2036) before the end of the requested ISFSI license renewal period (2052), even if Calvert Cliffs chooses not to further extend the operating licenses, Calvert Cliffs will remain responsible for maintaining the spent fuel pool and associated support equipment until such time that all irradiated fuel assemblies are transferred offsite. The requirements of 10 CFR 50.54(bb) will ensure components covered under 10 CFR Part 50 which support the site's ISFSI operation will continue to be appropriately maintained for the remaining duration of the ISFSI renewal period. Aging management for the spent fuel cask handling crane is managed by Calvert Cliffs programs.

Chapter 16 of the Calvert Cliffs UFSAR (Reference 3.10.11) contains the programs credited for managing the effects of aging in the Calvert Cliffs Nuclear Power Plant renewed license period. These programs consist of existing programs, existing programs that are or will be modified for managing specific aging mechanisms, and new programs to manage the effects of age-related degradation where a program did not exist. Calvert Cliffs UFSAR Table 16-2, Item 11 pertains to the spent fuel cask handling crane. The components included are the handling equipment and heavy load handling crane components and the carbon steel wire rope. The aging mechanism is general corrosion, fatigue, wear, and mechanical degradation.

Existing Calvert Cliffs programs are credited for discovery and management of fatigue, wear, and mechanical degradation of carbon steel wire rope and general corrosion of carbon steel components of fuel handling equipment by performing visual inspections. Thus, there are preventive maintenance tasks that explicitly present the inspection requirements discussed in the Calvert Cliffs UFSAR Table 16-2.

Since the aging management for the spent fuel cask handling crane is managed by Calvert Cliffs programs (10 CFR Part 50), the aging of the spent fuel cask handling crane will be adequately managed so that there is reasonable assurance that SSC intended function(s) will be maintained.

3.9 AGING MANAGEMENT REVIEW RESULTS – SPENT FUEL HANDLING MACHINE

This section provides the results of the aging management review of the SFHM which was determined to be in the scope of license renewal as identified in Section 2.3. The SFHM is a 10 CFR Part 50 SSC that is used during the dry cask storage operations in the auxiliary building.

The SFHM is located above the SFP. It is a bridge and trolley arrangement which rides on rails set in the concrete on each side of the pool. The handling machine is designed to pass over the dividing wall (separating the two halves of the pool) and to serve both halves of the pool. Latitude and longitude motors on the bridge and trolley position the machine over the specified rack location in the SFP.

The SFHM serves several purposes; one purpose is to transfer irradiated fuel assemblies from the upending mechanism to a location in the SFP for decay, or to transfer new fuel to the upending machine. A second purpose is to take fuel from the new fuel elevators and transfer the fuel to a rack location in the SFP or to the fuel upending mechanism. A third function is to move fuel to and from the spent fuel inspection elevator, inspection/repair stations, and to move fuel between storage rack locations. A fourth function of the SFHM is to transfer irradiated fuel assemblies to shipping or storage casks.

Some of the safety features incorporated in this equipment are interlocks to prevent movement into the walls. These interlocks can be bypassed and restored in accordance with approved procedures. Additional safety features include limit switches to prevent the hoist from raising fuel above the point where adequate water for shielding is available. A redundant mechanical stop will prevent a fuel assembly from being raised above specified limits. The fuel grappling tool is designed so that the fuel assembly cannot be released accidentally. Inadvertent disengagement of a fuel assembly from the fuel handling machine is prevented by mechanical interlocks; consequently, the possibility of dropping and damaging of a fuel assembly is remote. All motors are equipped with mechanical brakes or self-locking gears to prevent movement in case of a loss of power.

The SFHM, reactor refueling machine, and their associated components are inspected prior to refueling campaigns (i.e., defuel/refuel or fuel shuffle). These inspections involve performing a walkdown for foreign material and cleanliness, inspecting the SFHM and associated equipment for damaged, corroded, or deteriorated parts, and checking cleanliness of rail surfaces. The plant's nuclear operations procedures have numerous levels of controls and reviews, including assignment of responsibility for conducting performance evaluations as required, reviewing all the evaluations for accuracy and completeness, and analyzing data for trends, if applicable. Specific responsibilities are assigned to Calvert Cliffs personnel for monitoring these programs through periodic audits. These controls provide reasonable assurance that the associated activities will continue to be an effective means of monitoring the fuel handling equipment for the effects of general corrosion/oxidation.

Since the operation of the SFHM is controlled by 10 CFR Part 50 no 10 CFR Part 72 aging management review was performed for the SFHM. While the current operating licenses of the Calvert Cliffs Units will expire (2034 and 2036) before the end of the requested ISFSI license renewal period (2052), even if Calvert Cliffs chooses not to further extend the operating licenses, Calvert Cliffs will remain responsible for maintaining the spent fuel pool and associated support equipment until such time that all irradiated fuel assemblies are transferred offsite. The requirements of 10 CFR 50.54(bb) will ensure components covered under 10 CFR Part 50 which support the site's ISFSI operation will continue to be appropriately maintained for the remaining duration of the ISFSI renewal period. Aging management for the SFHM is managed by Calvert Cliffs programs.

Chapter 16 of the Calvert Cliffs UFSAR (Reference 3.10.11) contains the programs credited for managing the effects of aging in the Calvert Cliffs Nuclear Power Plant renewed license period. These programs consist of existing programs, existing programs that are or will be modified for managing specific aging mechanisms, and new programs to manage the effects of age-related degradation where a program did not exist. Calvert Cliffs UFSAR Table 16-2, Items 12 and 13 pertains to the SFHM. The components included are carbon steel components and the stainless steel wire rope. The aging mechanism is general corrosion, fatigue, wear, and mechanical degradation.

Existing Calvert Cliffs programs are credited for discovery and management of fatigue, wear, and mechanical degradation of stainless steel wire rope and general corrosion of carbon steel components of spent fuel handling equipment by performing visual inspections. Thus, there are preventive maintenance tasks that explicitly present the inspection requirements discussed in the Calvert Cliffs UFSAR Table 16-2.

Since the aging management for the SFHM is managed by Calvert Cliffs programs (10 CFR Part 50), the aging of the SFHM will be adequately managed so that there is reasonable assurance that SSC intended function(s) will be maintained.

3.10 REFERENCES (SECTION 3.0 AGING MANAGEMENT REVIEWS)

3.10.1 Attachment to letter from Mr. L. N. Hartz (Virginia Electric Power Company), to Document Control Desk (NRC), dated June 26, 2001, Surry Independent Spent Fuel Storage Installation, Comments on NRC Preliminary Guidance for Part 72 License Renewal

3.10.2 American Society for Testing and Materials, ASTM C 1562-03, July 2007, Standard Guide for Evaluation of Materials Used in Extended Storage of Interim Spent Nuclear Fuel Dry Storage Systems

3.10.3 EPRI TR-1003416, J. Kessler and R. Einziger, December 2002, Technical Bases for Extended Dry Storage of Spent Nuclear Fuel, Electric Power Research Institute, Palo Alto, CA, (Final Report)

3.10.4 EPRI TR-108757, R. E. Einziger, D. L. Baldwin, and S. G. Pitman, April 1998, Data Needs for Long-Term Dry Storage of LWR Fuel, Electric Power Research Institute, Palo Alto, CA (Interim Report)

3.10.5 EPRI TR-1002882, J. Kessler, September 2002, Dry Cask Storage Characterization Project, Final Report, Electric Power Research Institute, Palo Alto, CA (Final Report)

3.10.6 IAEA Technical Report Series No. 443, 2006, Understanding and Managing Ageing of Material in Spent Fuel Storage Facilities, International Atomic Energy Agency, Vienna

3.10.7 NRC Interim Staff Guidance 11, November 17, 2003, Revision 3, Cladding Considerations for the Transportation and Storage of Spent Fuel, NCR, Spent Fuel Project Office

3.10.8 Technical Specifications, Appendix A to the Calvert Cliffs Independent Spent Fuel Storage Installation Materials License No. SNM-2505, Amendment 7

3.10.9 Calvert Cliffs Independent Spent Fuel Storage Installation Updated Safety Analysis Report, Revision 17, September 8, 2008

3.10.10 NUH-002, Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS®-24P, Revision 2, March 1990

3.10.11 Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report, Revision 40, September 8, 2008

3.10.12 NUREG-1801, Volume 1 and 2, September 2005, Generic Aging Lessons Learned (GALL) Report

3.10.13 EPRI TR-1019110, Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications, 2009 Edition, Electric Power Research Institute, Palo Alto, CA

3.10.14 UCRL-ID-130386, March 1998, Scoping Corrosion Tests on Candidate Waste Package Basket Materials for the Yucca Mountain Project, Lawrence Livermore National Laboratory

Table 3.2-1, Aging Management Review Results for the Irradiated Fuel Assemblies

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Fuel Rod (Cladding and End Caps)	CC, HT, PB, SS	Zircaloy-4	Air and Gas[1,2] Residual Boron Coating[3]	None Identified	None Required
Guide Tubes	SS	Zircaloy-4 Stainless Steel (Chrome Plated)	Air and Gas[1]	None Identified	None Required
Spacer Grid Assemblies	CC, SS	Zircaloy-4	Air and Gas[1]	None Identified	None Required
Lower End Fitting (and Connectors)	SS (CC, SS)	Stainless Steel Inconel	Air and Gas[1]	None Identified	None Required
Upper End Fitting (Connectors and Holddown Spring)	SS	Stainless Steel Inconel X-750	Air and Gas[1]	None Identified	None Required
Holddown Spring Retainer and Upper End Plugs	None	N/A	N/A	N/A	N/A
Fuel Assembly Control Components	None	N/A	N/A	N/A	N/A
Fuel Rod Pellets and Other Internal Portions	None	N/A	N/A	N/A	N/A

Table 3.2-1, Aging Management Review Results for the Irradiated Fuel Assemblies

Notes:

[1] Air and gas environment outside the fuel rods (inside the DSC) is helium at atmospheric pressure with trace amounts of air and water vapor. Minimal amounts of fission product gases may also be present. Temperature and radiation have been considered as described in Section 3.2.3, Environments for the Irradiated Fuel Assemblies.

[2] Air and gas environment inside the fuel rods is pressurized helium and fission product gases. Temperature and radiation have been considered as described in Section 3.2.3, Environments for the Irradiated Fuel Assemblies.

[3] Residual boron may coat the irradiated fuel assemblies surfaces since they were exposed to a borated water environment in the SFP prior to storage. Any boric acid residue remaining on the irradiated fuel assemblies will have no deleterious effects due to the minimal amount of water remaining on the irradiated fuel assembles and the materials of construction for the irradiated fuel assemblies.

CC Provides criticality control of spent fuel
HT Provides heat transfer
PB Directly or indirectly maintains a pressure boundary (confinement)
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

Table 3.3-1, Aging Management Review Results for the DSCs (NUHOMS-24P and -32P)

Subcomponent[1]	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
DSC Basket:	SS, CC	Stainless Steel	Air and Gas[2]	None Identified	None Required
- Guide Sleeves	SS, CC	Stainless Steel	Air and Gas[2]	None Identified	None Required
- Spacer Disks	SS, CC	Stainless Steel / Aluminum Coated Carbon Steel	Air and Gas[2]	None Identified	None Required
- Support Rods	SS, CC	Stainless Steel / Aluminum Coated Carbon Steel	Air and Gas[2]	None Identified	None Required
- Rails	SS	Stainless Steel	Air and Gas[2]	None Identified	None Required
- Rail Inserts	SS	Aluminum	Air and Gas[2]	None Identified	None Required
- Fixed Neutron Absorbers	CC	Borated Aluminum Alloy	Sheltered[3]	None Identified	None Required
DSC Shell w/Bottom Shield Plug	PB, SH, SS, HT	Stainless Steel Stainless Steel and Lead	Air and Gas[2] Air and Gas[2] / Sheltered[3]	None Identified	None Required
Shield Plug (Top)	PB, SH, SS, HT	Stainless Steel and Lead	Air and Gas[2] Sheltered[3]	None Identified	None Required
Cover Plates (Top and Bottom)	PB, SH, SS, HT	Stainless Steel	Air and Gas[2]	None Identified	None Required
Siphon and Vent Ports	PB, SH	Stainless Steel	Air and Gas[2]	None Identified	None Required
Ram Grapple Ring	SS	Stainless Steel	Air and Gas[2]	None Identified	
Dry Film Lubricant	None	N/A	N/A	N/A	N/A
Swagelok Quick Disconnects	None	N/A	N/A	N/A	N/A
Siphon Tube	None	N/A	N/A	N/A	N/A
Aluminum Coating (Carbon Steel Spacer Discs and Top Shield Plugs)	None	N/A	N/A	N/A	N/A

Table 3.3-1, Aging Management Review Results for the DSCs (NUHOMS-24P and -32P)

Subcomponent[1]	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Nickel-based Thread Lubricant; Thread Tape or Sealant	None	N/A	N/A	N/A	N/A
Stainless Steel Plugs/Bolts (Non-Structural)	None	N/A	N/A	N/A	N/A
DSC Lifting Lugs	None	N/A	N/A	N/A	N/A

Notes:

[1] Each individual DSC may not contain all of the listed subcomponents

[2] Air and gas environment is helium inside DSC cavity, with possible trace amounts of air, water vapor and fission product gases. Temperature and radiation have been considered as described in Section 3.3.3, Environments for the DSCs.

[3] Sheltered environment for DSC interior/exterior surfaces that are not part of helium filled DSC cavity.

CC Provides criticality control of spent fuel
HT Provides heat transfer
PB Directly or indirectly maintains a pressure boundary (confinement)
SH Provides radiation shielding
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

Table 3.4-1, Aging Management Review Results for the HSM

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Reinforced Concrete Walls, Roof, and Foundation	HT, SH, SS	Concrete Steel	Yard-Salt, Air	Loss of Material Spalling, Scaling and Cracking / Freeze-Thaw[1] Change in Material Properties[1]	ISFSI Aging Management Program
Reinforced Concrete Walls, Roof, and Foundation (Underground)	HT, SH, SS	Concrete Steel	Embedded / Underground	None Identified	None Required
DSC Structural Steel Support Assembly	SS	Carbon Steel	Sheltered	Loss of Material	ISFSI Aging Management Program
		Nitronic 60 Stainless Steel	Sheltered	None Identified	None Required
DSC Seismic Retainer for HSMs	SS	Carbon Steel	Sheltered	Loss of Material	ISFSI Aging Management Program
Cask Docking Flange and Tie Restraints	SS	Carbon Steel	Sheltered	Loss of Material	ISFSI Aging Management Program
			Yard	Loss of Material	ISFSI Aging Management Program
Heat Shield	HT	Stainless Steel	Sheltered	None Identified	None Required
Shielded Front Access Door and Door Supports	SH, SS	Carbon Steel	Yard	Loss of Material	ISFSI Aging Management Program
		Concrete	Embedded	None Identified	None Required
Ventilation Air Openings (One Inlet / Two Outlets)	HT	Stainless Steel	Yard	None Identified	None Required

Table 3.4-1, Aging Management Review Results for the HSM

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Shielded Ventilation Air Inlet Plenum (Concrete)	HT	Concrete	Yard	Loss of Material, Spalling, Scaling and Cracking / Freeze-Thaw[1] Change in Material Properties[1]	ISFSI Aging Management Program
Shielded Ventilation Air Inlet Plenum (Stainless Steel)	HT	Stainless Steel	Embedded / Yard	None Identified	None Required
Ventilation Air Outlet Shielding Blocks (Concrete)	HT	Concrete	Yard	Loss of Material, Spalling, Scaling and Cracking / Freeze-Thaw[1] Change in Material Properties[1]	ISFSI Aging Management Program
Ventilation Air Outlet Shielding Blocks (Stainless Steel)	HT	Stainless Steel	Embedded / Yard	None Identified	None Required
Lightning Protection System	SS	Copper	Yard	None Identified	None Required
Threaded Fasteners and Expansion Anchors	HT, SS	Stainless Steel	Yard Embedded / Yard	None Identified	None Required
Handrail	SS	Carbon Steel	Yard	None Identified	None Required
Ladder and Attachments	None	N/A	N/A	N/A	N/A
Caulk, Sealants, Expansion Joint Fillers	None	N/A	N/A	N/A	N/A
Dry Film Lubricants	None	N/A	N/A	N/A	N/A
Polyvinyl Chloride Drain Pipe	None	N/A	N/A	N/A	N/A
Electrical Conduit, Boxes, and Cable	None	N/A	N/A	N/A	N/A
Alignment Targets	None	N/A	N/A	N/A	N/A

Notes:

[1] Aging effect conservatively included to meet NRC position for 10 CFR Part 54 plant license renewal (ISG 3).

HT Provides heat transfer
SH Provides radiation shielding
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

Table 3.5-1, Aging Management Review Results for the Transfer Cask

Subcomponent	Intended Function	Material Group	Environment[1]	Aging Effects Requiring Management	Aging Management Activities
Cask Body – Structural Shell	SS, HT, SH	Carbon Steel	Embedded	None Identified	None Required
Cask Body –Bottom Support Ring	SS	Stainless Steel	Sheltered	None Identified	None Required
Cask Body – Bottom Cover Plate	SS	Stainless Steel	Sheltered	None Identified	None Required
Cask Body – Top Flange	SS	Stainless Steel	Sheltered	None Identified	None Required
Cask Body – Inner Shell	SS, HT, SH	Stainless Steel	Sheltered	None Identified	None Required
Cask Body – Lead	SH, HT	Lead	Embedded	None Identified	None Required
Cask Attachments - Rails	SS	Stainless Steel (Nitronic 60)	Sheltered	None Identified	None Required
Attachments – Upper Trunnions	SS	Stainless Steel	Sheltered	None Identified	None Required
Attachments – Upper Trunnion Sleeves	SS	Stainless Steel	Embedded	None Identified	None Required
Attachments – Upper Trunnion Nickel Alloy	SS	Inconel		None Identified	None Required
Attachments – Upper Trunnion Neutron Shielding	HT, SH	NS-3	Embedded	None Identified	None Required
Attachments – Upper Trunnion Cover Plate	SS, SH	Stainless Steel	Sheltered	None Identified	None Required
Attachments – Lower Trunnions	SS	Stainless Steel	Sheltered	None Identified	None Required
Attachments – Lower Trunnion Sleeves	SS	Stainless Steel	Embedded	None Identified	None Required

Table 3.5-1, Aging Management Review Results for the Transfer Cask

Subcomponent	Intended Function	Material Group	Environment[1]	Aging Effects Requiring Management	Aging Management Activities
Attachments – Lower Trunnion Sleeve Nickel Alloy Weld Overlay	SS	Stainless Steel	Embedded	None Identified	None Required
Attachments – Lower Trunnion Neutron Shielding	HT, SH	NS-3	Embedded	None Identified	None Required
Penetration – Ram Access Penetration Ring	SS	Stainless Steel	Sheltered	None Identified	None Required
Cast Neutron Shield – Upper and Lower Rings, Outer Shell, Relief Valve Support Plates	SH, HT	Stainless Steel	Sheltered	None Identified	None Required
Cask Neutron Shield – Inner and Outer Support Angle	SH, HT	Stainless Steel	Embedded	None Identified	None Required
Cask Neutron Shield - Shielding Material	SH, HT	NS-3	Embedded	None Identified	None Required
Cask Cover Plate and Accessories:					
Top Cover Assembly – Inner, Outer, and Side Plates	SS	Stainless Steel	Sheltered	None Identified	None Required
Top Cover Assembly – Ring; Eye Bolt Stand-offs	SS	24 Hot Galvanized Finish	Sheltered	None Identified	None Required
Top Cover Assembly – Neutron Shielding	SH	NS-3	Embedded	None Identified	None Required

Table 3.5-1, Aging Management Review Results for the Transfer Cask

Subcomponent	Intended Function	Material Group	Environment[1]	Aging Effects Requiring Management	Aging Management Activities
Bottom Cover Assembly – Inner, Outer, and Side Plates	SH	Stainless Steel	Sheltered	None Identified	None Required
Bottom Cover O-ring Seals	HT, SH	Polymer (Ethylene Propylene)	Sheltered	Materials Property Change	N/A – Subject to Routine Replacement
Bottom Cover Assembly – Neutron Shielding	SH	NS-3	Embedded	None Identified	None Required
Cask Bottom Cover Plate	SH	Stainless Steel	Sheltered	None Identified	None Required
Cask Bottom – Neutron Shielding	SH	NS-3	Embedded	None Identified	None Required
Bolts, Washers, and Threaded Fasteners for Top Cover Plate and Ram Access Plate	SS	Carbon Steel	Sheltered	Loss of Material	Transfer Cask Aging Management Program

Table 3.5-1, Aging Management Review Results for the Transfer Cask

Subcomponent	Intended Function	Material Group	Environment[1]	Aging Effects Requiring Management	Aging Management Activities
Miscellaneous Subcomponents (Lubricants, sealants; Eyebolts for bottom access plate and top cover plate; Cover plates, gaskets, and screws for annulus fill / drain valves; Swagelok relief valves for solid neutron shield; Polyethylene caps for Swagelok vales; Alignment pin for top cover plate; Stick-on alignment targets)	None	N/A	N/A	N/A	N/A

Notes:

[1] Sheltered environment represents ambient conditions on the interior of the transfer cask, conservatively including connecting and embedded surfaces. Some subcomponents may have interior surfaces that are considered embedded. No aging effects are identified for the embedded surfaces and no aging management is required. Temperature and radiation were considered, as described in Section 3.5.3, Environments for the Transfer Cask.

HT Provides heat transfer
SH Provides radiation shielding
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

Table 3.6-1, Aging Management Review Results for the Transfer Cask Lifting Yoke

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Lifting Hook Plates	SS	Carbon Steel	Sheltered	Loss of Material	Transfer Cask Lifting Yoke Aging Management Program
Lifting Beam Plates	SS	Carbon Steel	Sheltered	Loss of Material	Transfer Cask Lifting Yoke Aging Management Program
Later Brace Plates	SS	Carbon Steel	Sheltered	Loss of Material	Transfer Cask Lifting Yoke Aging Management Program
Support Brace Plates	SS	Carbon Steel	Sheltered	Loss of Material	Transfer Cask Lifting Yoke Aging Management Program
Pin (Round Bar)	SS	Stainless Steel	Sheltered	None Identified	None Required
Pin Handle	None	N/A	N/A	N/A	N/A
Pin Cradle Pipe	None	N/A	N/A	N/A	N/A
Rear Pin Stop	None	N/A	N/A	N/A	N/A
Pin Lock	None	N/A	N/A	N/A	N/A
Main Assembly Bolts, Nuts Washers	SS	Carbon Steel	Sheltered	Loss of Material	Transfer Cask Lifting Yoke Aging Management Program
Support angles, and miscellaneous screws, bolts, nuts, and washers; bumpers and mounting flange plates, studs; eyebolts, connectors (safety anchor shackles), turnbuckles, wire rope	None	N/A	N/A	N/A	N/A
Hook Bearing Plate	SS	Bronze	Sheltered	None Identified	None Required

SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

Table 3.7-1, Aging Management Review Results for the Cask Support Platform

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Base Plate	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Web Plates	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Mid Plate	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Top Plate	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Honeycomb Energy Absorber	SS	Aluminum	Compressed Air	None Identified	None Required
Honeycomb Base Plate	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Honeycomb Casing Plate	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Honeycomb Outer Plate	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Bottom Location Plates	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Lifting Lugs	SS	Stainless Steel	Borated Water	Loss of Material, Cracking	Chemistry Control Program
Tubing Manifold, Relief Valve, Pressure Gauge, Quick-Connect,	None	N/A	N/A	N/A	N/A

SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

Appendix A

Aging Management Program

A1.0 INTRODUCTION

This appendix is a summary of the activities that manage the effects of aging for subcomponents that have been identified in the license renewal application as being subject to aging management review. The following aging management programs have been credited for the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI):

- Horizontal Storage Module (HSM) Aging Management Program
- Transfer Cask Aging Management Program
- Transfer Cask Lifting Yoke Aging Management Program
- Cask Support Platform Aging Management Program

Each of these aging management programs is discussed in Subsections A2.1 through A2.4. These subsections provide a description of the aging management program which includes an introduction, an evaluation of the aging management program in terms of the attributes of an effective aging management program, and a summary.

Tables that summarize the results of aging management reviews are found in Section 3.0 of Attachment (1). These tables identify the programs and activities credited for managing the aging effects for each subcomponent listed in the review. Each aging management program manages the aging effects, or the relevant conditions that could lead to the aging effects, applicable to a subcomponent, and provides reasonable assurance that the integrity of the subcomponent will be maintained under current licensing basis conditions during the renewed license period.

A2.0 EXISTING AGING MANAGEMENT PROGRAMS

A2.1 HSM AGING MANAGEMENT PROGRAM

The ISFSI provides for long-term dry interim storage for irradiated fuel assemblies until such time that the irradiated fuel assemblies are shipped off site for final disposition. The fuel assemblies are confined in stainless steel canisters. Each canister is protected and shielded by a concrete horizontal storage module (HSM) and rests on a steel support rail assembly that is anchored to the walls of the storage module and restrained inside the storage module. Other steel components provide heat shielding, screens, and attachments, both inside and outside the modules. The HSM aging management program includes the concrete and steel members associated with the HSMs.

The purpose of the HSM aging management program is to:

- Ensure that no significant degradation to the HSMs occurs
- Maintain the air inlets and outlets free from obstructions

A description of the HSM aging management program is provided below using each attribute of an effective aging management program as described in the preliminary guidance for the renewal of site specific Part 72 licenses (Reference A3.1).

Scope

The scope of the HSM aging management program involves monitoring the exterior surfaces of the HSM. It includes visual inspection of the accessible concrete and exposed steel, monitoring area radiation levels for compliance with dose limits, and ensuring that the inlets and outlets do not become blocked.

Preventive Actions

The aging management program is primarily a condition monitoring program. With the exception of daily surveillances to ensure HSM inlets and outlets are not obstructed, no preventive actions are performed. Maintaining the inlets and outlets free from obstruction ensures temperatures are not elevated for prolonged periods, the concrete is not subject to related damage, and overheating of the components inside an HSM is prevented.

Parameters Monitored or Inspected

Consistent with the current Nuclear Regulatory Commission position relative to including concrete in an aging management program, the accessible concrete is visually examined for indication of surface deterioration. Degradation could affect the ability of the concrete to provide support to the dry shielded canisters (DSCs), to provide radiation shielding, to provide missile shielding, or to provide a path for heat transfer. The above-grade exterior concrete is accessible. The below-grade exterior concrete surfaces are inaccessible. Interior concrete is accessible for remote exams. The above-grade exterior concrete is a leading indicator for the interior concrete.

Steel on the external surface of the HSMs which is subject to wetting/moisture is visually examined for the aging effect of loss of material (corrosion). This aging effect could affect the ability of the steel to perform its intended function.

Surveillances of area radiation levels are made and compared to established limits. Levels exceeding limits are investigated for potential degradation of the HSM components. Increased levels could indicate a reduction in the ability of the concrete and steel to provide adequate radiation shielding, or could indicate a breach in the containment function of the DSC or irradiated fuel assemblies. Dose rates are measured at predetermined locations on the HSMs.

Daily surveillances are performed by security personnel to ensure the air inlets and outlets are free from obstructions, thereby preventing reduced air flow and potential overheating of the components located inside an HSM. Plant procedures are in place for these inspections and surveillances.

Detection of Aging Effects

The examination method used for the accessible concrete and steel is primarily a visual examination at an established frequency. Refer to Table A-1 for inspection details.

The visual examination involves monitoring the interior and exterior surfaces of the HSMs, including visual inspection of the accessible concrete, any exposed steel subcomponents, embedments, and attachments, and the lightning protection system. Interior inspections are performed on the HSMs prior to cask loadings in accordance with the applicable plant procedure. Exterior inspections are conducted yearly.

Monitoring and Trending

The surveillance tests for monitoring radiation and contamination could identify a crack in the shielding or a loss of the containment function. This surveillance is performed monthly per the Radiological Environmental Monitoring Program to monitor the ISFSI dose. If any pre-established limits are exceeded, site engineering is required to be notified.

Acceptance Criteria

A set of inspection attributes and acceptance standards for steel and concrete that is commensurate with industry codes, standards, and guidelines has been established. Components are determined to be either Acceptable, Acceptable with Defects, or Unacceptable. Acceptable signifies that a component is free of significant deficiencies or degradation that could lead to the loss of structural integrity. Acceptable with Defects signifies that a component contains deficiencies or degradation but will remain able to perform its design basis function until the next inspection or repair. Unacceptable signifies a component contains deficiencies or degradation that either prevents (or could prevent prior to the next inspection) the ability to perform their design basis function.

A daily surveillance performed by security personnel ensures that the inlet and outlet vents are free from obstruction consistent with the ISFSI Technical Specifications.

Corrective Actions

Corrective actions, including root cause determinations are performed in accordance with the Corrective Action Program. Corrective actions are taken in a timely manner commensurate with the significance of the defect. Deficiencies are either promptly corrected or are evaluated to be acceptable through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions.

Confirmation Process

Activities initiated in accordance with the implementing procedures for the HSM aging management program, such as corrective actions, are subject to Quality Assurance Program controls. Thus, the effectiveness is monitored using Corrective Action Program procedures, review and approval processes, and administrative controls, which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. Use of these procedures, processes, and controls ensures that corrective actions are taken and are effective.

Administrative Controls

The HSM aging management program is subject to Corrective Action and Quality Assurance procedures, review and approval processes, and administrative controls. These are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, and will continue to be adequate for the renewed license (extended storage) period.

Operating Experience

The ISFSI has been in operation since 1992. Examinations and inspections are performed in accordance with plant procedures and repetitive maintenance tasks. See Table A-1 attached. A review of the Corrective Action Program indicated that any deficiencies identified for the ISFSI have been administrative and were not related to aging mechanisms and effects. Minor cracking was noted on top of the HSMs which required cosmetic crack repair.

As discussed in Section 3.1.5 of Attachment (1), plant-specific and industry operating experience, as well as a review of ISFSI files, did not indicate any aging related deficiencies with the HSM components.

Summary

A review of industry operating experience identified a large number of events related to dry storage. Many of these were event-driven incidents, and most were not related to aging management. However

for the incidents that involved aging mechanisms, barrier analyses were conducted to assess Calvert Cliffs susceptibility to these mechanisms. Our review indicated the aging mechanisms are bounded by the aging management review performed for the Calvert Cliffs SSCs.

Operating experience to date has not indicated any significant degradation to any of the HSM components. Inspections and surveillances that would identify any deficiencies continue to be conducted. A corrective action program is in place to track and correct deficiencies in a timely manner. Continued implementation of the ISFSI aging management program provides reasonable assurance that the aging effects will be managed, such that the intended functions of the HSM components, particularly the structural concrete and steel of the HSMs, will be maintained under current licensing basis conditions for the renewed license period.

A2.2 TRANSFER CASK AGING MANAGEMENT PROGRAM

The transfer cask is used to transport the DSCs containing the irradiated fuel assemblies from the spent fuel pool to the corresponding HSMs (and back as necessary). The transfer cask subcomponents, materials, environments and aging effects requiring management are described in Section 3.5 of Attachment (1).

The purpose of the transfer cask aging management program is to ensure that no significant degradation to the transfer cask occurs, with the focus being on the frequently wetted surfaces, as well as carbon steel surfaces, prior to its use for future retrieval of a DSC from the corresponding HSM.

A description of the transfer cask is provided below using each attribute of an effective aging management program as described in the preliminary guidance for the renewal of site Part 72 licenses (Reference A3.1):

Scope

The transfer cask aging management program is applicable to the transfer cask and subcomponents. The focus of this aging management program is on the stainless steel subcomponents that have frequently wetted surfaces and, conservatively, those external surfaces exposed to outdoor conditions and intermittent wetting. It also conservatively includes carbon steel subcomponents that are exposed to weather and other forms of moisture (e.g., humidity).

The program performs visual inspections of the exterior surfaces.

Preventive Actions

The transfer cask aging management program includes guidance and direction for maintaining a suitable environment that precludes the onset or propagation of a loss of material due to crevice or pitting corrosion for continuously wetted surfaces.

The parameter inspected by the transfer cask aging management program is visual evidence of degradation of external surfaces of the transfer cask.

Visual inspections of external cask and cask lid surfaces are performed annually or prior to moving a DSC (if no other inspection has been performed), to ensure that the intended function of the cask subcomponents are not compromised. Visual inspections look for signs of deterioration (corrosion).

Detection of Aging Effects

Loss of material for stainless steel subcomponents due to crevice and/or pitting corrosion in wetted locations and for carbon steel subcomponents due to general corrosion in moist atmospheric environments is an aging effect that is managed by the aging management program. The transfer cask aging management program relies upon a visual inspection to determine the physical condition of the exterior surfaces of the transfer cask and lid, prior to its use for DSC transfers. These inspections check for loss of material (corrosion).

Monitoring and Trending

Visual inspections will determine the existence of loss of material on the external surfaces of the transfer cask, and observations regarding the material condition are recorded in accordance with inspection procedures and are corrected or evaluated as satisfactory before use of the transfer cask. These inspections are either performed periodically or during the preparations for retrieval of a DSC from the corresponding HSM.

Evaluation of this information during the preparations for DSC retrieval and transfers provides adequate predictability and allows for corrective action prior to the need for the intended function of the component to be performed.

Acceptance Criteria

The transfer cask aging management program acceptance criteria for exterior surfaces are no unacceptable loss of material that could result in a loss of component intended function(s).

Unsatisfactory degradation is entered in the Corrective Action Program for resolution.

Corrective Actions

Corrective actions, including root cause determinations are performed in accordance with the Corrective Action Program. Corrective actions are taken in a timely manner commensurate with the significance of the defect. Deficiencies are either promptly corrected or are evaluated to be acceptable through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. Each of the implementing procedures associated with the transfer cask aging management program is within the scope of the Corrective Action Program.

Confirmation Process

Activities initiated in accordance with the implementing procedures for the transfer cask aging management program, such as corrective actions, are subject to Quality Assurance Program controls. Thus, the effectiveness is monitored using Corrective Action Program procedures, review and approval processes, and administrative controls, which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. Use of the procedures, processes, and controls ensures that corrective actions are taken and are effective.

Administrative Controls

The transfer cask aging management program is subject to Corrective Action and Quality Assurance Program procedures, review and approval processes, and administrative controls. These are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, and will continue to be adequate for the renewed license (extended storage) period.

Operating Experience

The ISFSI has been in operation since 1992. The transfer cask has been in use since the initial loading of the ISFSI.

Inspections have been performed on the transfer cask prior to each movement to the ISFSI. These examinations and inspections are currently performed in accordance with a combination of procedures. The overall effectiveness of these inspections in maintaining the condition and functionality of the cask is confirmed by the continued use of the cask. Any deficiencies identified are promptly corrected prior to moving fuel. This same process will be followed, as applicable, for moving the DSCs from the HSM back to the Calvert Cliffs spent fuel pool.

A discussion of pertinent operating experience is contained in Section 3.1.5 of Attachment (1). Furthermore, the lack of identification of cask degradation through the existing inspections is evidence that transfer cask activities have been effective in maintaining its condition and functionality.

A review of the operating experience provided objective evidence that the effects of aging have, and will continue to be, adequately managed during the extended period of operation.

Summary

The transfer cask aging management program is credited for the management of conditions that could lead to degradation of transfer cask subcomponents from the aging effects and mechanisms shown in Table 3.5-1 of Attachment (1), and for the management of actual degradation. Based on the above, the continued implementation of the transfer cask aging management program activities will provide reasonable assurance that aging effects will be managed, such that the transfer cask subcomponents within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis throughout the renewed license period.

A2.3 TRANSFER CASK LIFTING YOKE AGING MANAGEMENT PROGRAM

The transfer cask lifting yoke is used to move the transfer cask between the spent fuel pool and the transport trailer. The transfer cask lifting yoke, materials, environments and aging effects requiring management are described in Section 3.6 of Attachment (1).

The purpose of the transfer cask lifting yoke aging management program is to ensure that no significant degradation to the transfer cask lift yoke occurs, with the focus being on the repeatedly and intermittently wetted surfaces prior to its use for movement of the transfer cask.

A description of the transfer cask lifting yoke is provided below using each attribute of an effective aging management program as described in the preliminary guidance for the renewal of site Part 72 licenses (Reference A3.1):

Scope

The transfer cask lifting yoke aging management program is applicable to the transfer cask lifting yoke and the pertinent subcomponents. The focus of this aging management program is on the subcomponents that have external surfaces exposed to intermittent wetting.

The program performs visual inspections of the exterior surfaces.

Preventive Actions

The transfer cask lifting yoke aging management program includes guidance and direction for maintaining a suitable environment that precludes the onset or propagation of a loss of material due to corrosion for intermittently wetted surfaces.

The parameter inspected by the transfer cask lifting yoke aging management program is visual evidence of degradation of external surfaces of the transfer cask lifting yoke.

Visual inspections of transfer cask lifting yoke are performed annually or prior to moving a DSC (if no other inspection has been performed), to ensure that the intended function of the pertinent subcomponents are not compromised. Visual inspections look for signs of deterioration (corrosion).

Detection of Aging Effects

Loss of material due to corrosion in wetted locations or due to general corrosion in moist atmospheric environments is an aging effect that is managed by this aging management program. The transfer cask lifting yoke aging management program relies upon a visual inspection to determine the physical condition of the exterior surfaces of the transfer cask lifting yoke, prior to its use for transfers. These inspections check for loss of material due to corrosion.

Monitoring and Trending

Visual inspections will determine the existence of loss of material on the external surfaces of the transfer cask lifting yoke, and observations regarding the material condition are recorded in accordance with inspection procedures. These issues are corrected or evaluated as satisfactory before use of the transfer cask lifting yoke. These inspections are either performed periodically or during the preparations for movement of the transfer cask.

Evaluation of this information during the preparations for DSC retrieval and transfers provides adequate predictability and allows for corrective action prior to the need for the intended function of the component to be performed.

Acceptance Criteria

The acceptance criteria for the transfer cask lifting yoke aging management program for exterior surfaces is no unacceptable loss of material that could result in a loss of component intended function(s).

Unsatisfactory degradation is entered in the Corrective Action Program for resolution.

Corrective Actions

Corrective actions, including root cause determinations and prevention of recurrence, are performed in accordance with the Corrective Action Program. Corrective actions are taken in a timely manner commensurate with the significance of the defect. Deficiencies are either promptly corrected or are evaluated to be acceptable through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. Each of the implementing procedures associated with the transfer cask lifting yoke aging management program is within the scope of the Corrective Action Program.

Confirmation Process

Activities initiated in accordance with the implementing procedures for the transfer cask lifting yoke aging management program, such as corrective actions, are subject to Quality Assurance Program controls. Thus, the effectiveness is monitored using Corrective Action Program procedures, review and approval processes, and administrative controls, which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. Use of the procedures, processes, and controls ensures that corrective actions are taken and are effective.

Administrative Controls

The transfer cask lifting yoke aging management program is subject to Corrective Action and Quality Assurance Program procedures, review and approval processes, and administrative controls. These are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, and will continue to be adequate for the renewed license (extended storage) period.

Operating Experience

The ISFSI has been in operation since 1992. The transfer cask lifting yoke has been in use since the initial loading of the ISFSI.

Inspections are performed on the transfer cask lifting yoke annually. These examinations and inspections are currently performed in accordance with a combination of procedures. Any deficiencies identified are promptly corrected prior to shipping fuel.

A discussion of pertinent operating experience is contained in Section 3.1.5 of Attachment (1). Furthermore, the lack of identification of transfer cask lifting yoke degradation through the existing inspections is evidence that transfer cask lifting yoke activities have been effective in maintaining the condition and functionality of the transfer cask lifting yoke.

A review of the operating experience provided objective evidence that the effects of aging have, and will continue to be, adequately managed during the extended period of operation.

Summary

The transfer cask lifting yoke aging management program is credited for the management of relevant conditions that could lead to degradation of transfer cask lifting yoke subcomponents from the associated aging effects/mechanisms as shown in Table 3.6-1, and for the management of actual degradation. Based on the above, the continued implementation of the transfer cask lifting yoke aging management program activities will provide reasonable assurance that aging effects will be managed, such that the transfer cask lifting yoke subcomponents within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis throughout the renewed license period.

A2.4 CASK SUPPORT PLATFORM AGING MANAGEMENT PROGRAM

The cask support platform sits in the spent fuel pool cask loading pit area which positions the transfer cask at the appropriate elevation for loading irradiated fuel assemblies into the DSC. The cask support platform materials, environments and aging effects requiring management are described in Section 3.7 of Attachment (1).

The purpose of the cask support platform aging management program is to ensure that no significant degradation occurs while the cask support platform is in the borated water environment of the spent fuel pool.

A description of the cask support platform aging management program is provided below using each attribute of an effective aging management program as described in the preliminary guidance for the renewal of site specific Part 72 licenses (Reference A3.1):

Scope

The cask support platform aging management program is applicable to the cask support platform and the pertinent subcomponents. The focus of this aging management program is on the stainless steel subcomponents that are continuously exposed to borated water.

The cask support platform aging management program credits the Chemistry Control Program which monitors chlorides in the spent fuel pool water.

Preventive Actions

The cask support platform aging management program includes guidance and direction for maintaining a suitable environment that precludes the onset or propagation of a loss of material and cracking due to pitting and/or stress corrosion for stainless steel components in the spent fuel pool.

The parameter inspected by the cask support platform aging management program is the chloride concentration in the spent fuel pool water. Samples are taken on a monthly basis.

Detection of Aging Effects

Loss of material and cracking for stainless steel subcomponents, due to pitting and/or stress corrosion in borated water, and for stainless steel subcomponents, is an aging effect that is managed by this aging management program. The cask support platform aging management program relies upon chloride sampling of the spent fuel pool water to ensure favorable conditions for this aging effect do not develop.

Monitoring and Trending

Chloride sampling will determine whether conditions favorable for development of this aging effect have developed. In that event, additional inspections would be required before use of the cask support platform.

Acceptance Criteria

The acceptance criterion for the cask support platform aging management program is spent fuel pool chlorides within acceptable limits.

Unsatisfactory degradation is entered in the Corrective Action Program for resolution.

Corrective Actions

Corrective actions, including root cause determinations and prevention of recurrence, are performed in accordance with the Corrective Action Program. Corrective actions are taken in a timely manner commensurate with the significance of the defect. Deficiencies are either promptly corrected or are evaluated to be acceptable through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. Each of the implementing procedures associated with the cask support platform aging management program is within the scope of the Corrective Action Program.

Confirmation Process

Activities initiated in accordance with the implementing procedures for the cask support platform aging management program, such as corrective actions, are subject to Quality Assurance Program controls. Thus, the effectiveness is monitored using Corrective Action Program procedures, review and approval processes, and administrative controls, which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. Use of the procedures, processes, and controls ensures that corrective actions are taken and are effective.

Administrative Controls

The cask support platform aging management program is subject to Corrective Action and Quality Assurance Program procedures, review and approval processes, and administrative controls. These are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, and will continue to be adequate for the renewed license (extended storage) period.

Since initial operation, and continuing through present day operation, Calvert Cliffs has maintained a well-defined Chemistry Control Program for fluid systems. A key aspect of the Chemistry Control Program is the sampling and analysis of fluid systems to determine the concentration of chemical impurities and chemical additives. Fluid systems at Calvert Cliffs are sampled and analyzed by procedure. Parameters monitored, frequency of sampling, acceptance criteria (i.e., specifications), and corrective actions for out-of-specification results are similarly addressed by procedure. Chemistry data for monitored parameters is routinely trended to identify subtle trends in the data which may be indicative of an underlying operational problem. In many cases, this allows correction prior to a parameter becoming out-of-specification.

Operating Experience

A discussion of pertinent operating experience is contained in Section 3.1.5 of Attachment (1).

The overall effectiveness of the chemistry program is supported by the excellent operating experience for those systems, structures, and components influenced by the Chemistry Control Program.

Summary

The cask support platform aging management program credits the Chemistry Control Program for the management of relevant conditions that could lead to degradation of cask support platform subcomponents from the associated aging effects/mechanisms as shown in Table 3.7-1, and for the management of actual degradation. Based on the above, the continued implementation of the cask support platform aging management program activities will provide reasonable assurance that aging effects will be managed, such that the cask support platform subcomponents within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis throughout the renewed license period.

A3.0 REFERENCES (AGING MANAGEMENT PROGRAMS)

A3.1 Letter from Steven Baggett (NRC) to John Meyer (CP&L), dated May 17, 2001, Preliminary NRC Staff Guidance for 10 CFR Part 72 License Renewal, Spent Fuel Storage

Table A-1, ISFSI Aging Management Examinations and Inspection Procedures

Program	Frequency	Inspection
HSM Aging Management	Annually	• To perform HSM rebar inspection and looking for spalled and cracking concrete.
	Daily	• To check the HSM inlet and outlet screens are not obstructed.
	Prior to Loading Cask	• Inspect in accordance with plant procedures.
Transfer Cask Aging Management	Annually	• To perform visual inspection of transfer cask exterior surfaces and of carbon steel bolts/washers looking for signs of significant degradation (corrosion).
Transfer Cask Lifting Yoke Aging Management	Annually	• To perform visual inspection focusing on subcomponents that which external surfaces are exposed to intermittent wetting.
Cask Support Platform Aging Management	Monthly	• The Chemistry Control Program monitors chloride concentration in spent fuel pool to ensure favorable conditions for stress corrosion/ pitting do not occur.

Appendix B

Time-Limited Aging Analysis

INTRODUCTION

The Nutech Horizontal Modular Storage (NUHOMS)-24P and NUHOMS-32P systems are utilized for storage of spent fuel at Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI). The Nuclear Regulatory Commission issued a 20-year 10 CFR Part 72 license (SNM-2505) for this facility on November 30, 1992. The license renewal application is required to contain an evaluation of Time-Limited Aging Analysis to demonstrate the safe operation over the extended service life for the ISFSI facility. The Time-Limited Aging Analysis is contained in this Appendix and is comprised of the following items:

1. AREVA Technical Report 10955-0101, Revision 1, May 21, 2010, ISFSI Time Limited Aging Analysis Report
2. AREVA Calculation 10955-0201, Revision 0, March 30, 2010, Fatigue Analysis for NUHOMS® 32P Transfer Cask Trunnion and Lifting Yoke System
3. AREVA Calculation 10955-0202, Revision 0, March 30, 2010, DSC Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P
4. AREVA Calculation 10955-0203, Revision 0, March 30, 2010, Transfer Cask Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P

AREVA

TECHNICAL REPORT

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ISFSI TIME-LIMITED AGING ANALYSIS REPORT

Summary:

The NUHOMS®-24P and NUHOMS®-32P systems are utilized for storage of spent fuel at the Calvert Cliffs ISFSI (Independent Spent Fuel Storage Installation) by Calvert Cliffs Nuclear Power Plant, Inc (CCNPP). The NRC issued a 20-year 10CFR72 license (SNM-2505) [3] for storage of spent fuel at this facility on November 30, 1992 CCNPP is in the process of applying for license renewal for the ISFSI for an additional period of 40 years. This report identifies and evaluates the effect of Time-Limited Aging Analysis (TLAA) to demonstrate safe operation over the extended service life for the CCNPP spent fuel storage facility. In summary, all the items were found to be functionally adequate for a total service life of 60 years. Hence, it is concluded that, the entire NUHOMS® system with 24P and 32P DSCs will adequately perform for additional 40 years of service.

REVISION	NAMES AND INITIALS OF PREPARERS & DATES	NAMES AND INITIALS OF VERIFIERS & DATES	APPROVER NAME AND SIGNATURE	APPROVAL DATE
0	Venkata Venigalla	U.B. Chopra	Kamran Tavassoli	03/30/10
1	Venkata Venigalla Venkat	UB Chopra U.B. Chopra	Kamran Tavassoli	05/21/10

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1.0 INTRODUCTION

The NUHOMS®-24P and NUHOMS®-32P systems are utilized for storage of spent fuel at the Calvert Cliffs ISFSI (Independent Spent Fuel Storage Installation) by Calvert Cliffs Nuclear Power Plant, Inc (CCNPP). The NRC issued a 20-year 10CFR72 license (SNM-2505) [3] for storage of NUHOMS®-24P system at this facility on November 30, 1992. A modified version of the NUHOMS®-24P DSC designated as NUHOMS®-32P DSC, was added via a 72.48 evaluation and the supporting changes to the Technical Specifications are documented in [13] and [17]. A license amendment application for the NUHOMS®-32P DSC [22] to allow loading of increased burnup fuel up-to 52,000 megawatt-days per metric ton uranium the from the existing limit of 47,000 megawatt-days per metric ton uranium as specified in [17] is currently under review by the NRC. CCNPP is in the process of applying for license renewal for the ISFSI for an additional period of 40 years. Technical information provided for the license renewal is required to contain an evaluation of Time-Limited Aging Analysis (TLAA), to demonstrate safe operation over the extended service life. This report has been prepared to identify and evaluate the effect of TLAA for the CCNPP spent fuel storage facility.

2.0 METHODOLOGY

CCNPP calculations that form the basis for the Calvert Cliffs ISFSI plant specific design, were reviewed along with the Generic (Topical Report [2]) and plant specific Updated Safety Analysis Report (SAR [6]).

The TLAA were identified based on meeting all of the following attributes:

(1) Involve systems, structures and components within the scope of licensing renewal,

(2) Consider effects of aging,

(3) Involve time-limited assumptions defined by the current operating term of 20 years,

(4) Determined to be relevant by the licensee in making a safety determination,

(5) Involve conclusions or provide basis for conclusions related to the capability of the system, structure, and component to perform its intended function, and

(6) Are contained or incorporated in the Current Licensing Basis (CLB)

A "screening" review was performed to identify any calculations/evaluations where a time parameter was either explicitly included or was implied. After the calculations/evaluations were identified in the screening review, they were further evaluated to identify the limiting conditions. The identified time-limited components/analysis/assumptions/materials were evaluated and resolved using one or more of the alternatives identified below:

(i) The analyses remain valid for a renewal period of 40 years.

(ii) The analyses have been projected to the end of the renewal period.

(iii) The effects of aging on the intended function(s) will be adequately managed for the renewal period.

It is important to note, however, that the CCNPP Design Basis Documents for the Calvert Cliffs ISFSI (Specifications for the DSC, HSM and other components) required the facility to be designed for a 50-year service life. Most importantly, the design basis calculations documented in the Topical Report [2] for the NUHOMS® System are based on a service life of 50 years.

3.0 SCREENING

The screening identified several calculations/drawings/documents having potential for the TLAA. The documents shown in Table 1 are based on a final "shortlist" of documents among all the calculations, drawings, documents, specifications, and reports that were reviewed for the purpose of this evaluation. All documents pertaining to the ISFSI were screened for potential TLAA considerations, including non-conformances and "changes" arising out of licensing evaluations (72.48 changes). Documents that were screened but not included in the TLAA are presented in Appendix A. A brief description of the basis for the resolution of the TLAA considerations for the shortlisted documents is also provided in Table 1.

It is very important to consider that the source terms – thermal and radiological have reduced over the 20 year period and are expected to reduce further over the next 40 year period.

Table 1
Summary of Documents Screened

Item #	Document No Revision No.	Date Issued	Title	Time Dependent Parameter	Resolution	Comments
1	CA03948 Rev. 0	09/05/97	BG&E NUHOMS® HSM Heat Transfer Analysis	A. Decay Heat B. Concrete Temperature	Temperature Calculated with Decay Heat at the time of Initial Storage.	CCNPP ISFSI USAR Rev. 17, Section 8.1.3
2	CA06299 Rev.0	9/25/03	Exit Air Temperature and Bulk Air Temperatures within the HSM	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.1
3	CA06301 Rev.0	9/25/03	HSM Thermal Analysis Normal Storage Conditions	A. Decay Heat B. Concrete Temperature		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.1
4	CA06302 Rev. 0	1/15/04	HSM Thermal Analysis Accident Conditions (Max. Summer Temp)	A. Decay Heat B. Concrete Temperature		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.1
5	CA06303 Rev. 0	1/15/04	HSM Thermal Analysis (Min. Winter Temp)	A. Decay Heat B. Concrete Temperature		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.1
6	CA06304 Rev. 1	02/16/05	HSM Thermal Analysis Accident Conditions (blocked vents)	A. Decay Heat B. Concrete Temperature		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.1
7	CA03949 Rev. 1	07/19/00	Dry Storage Cladding Temperature Limits for the 24P NUHOMS® System Using the CSFSM Model Presented in PNL-6189	A. Decay Heat B. Cladding Temperature Limits	See Section 4.1	CCNPP ISFSI USAR Rev. 17, Section 8.1.3.2 & Section 12.8.1.3.2 SER Section 2.2.6 [11]
8	CA03946 Rev. 0	09/05/97	DSC Heat Transfer Analysis for 8, Year Old Fuel	Decay Heat	Analyzed with Decay Heat at the time of Initial Storage.	CCNPP ISFSI USAR Rev. 17, Section 8.1.3.2 & Section 8.1.3.3
9	CA06305 Rev. 0	1/15/04	DSC Thermal Analysis Normal Storage Conditions	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.2
10	CA06306 Rev. 0	1/15/04	DSC Thermal Analysis Off Normal Conditions (Max Summer Temp.)	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.2
11	CA06307 Rev. 0	1/15/04	DSC Thermal Analysis Off Normal Conditions (Min. Winter Temp.)	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.2
12	CA06312 Rev. 0	9/25/03	Thermal Analysis Storage Cases Poison Plates in the Basket	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.2
13	CA03942 Rev. 1	09/21/98	ISFSI-Thermal analysis Of The DSC/Cask Geometry For The Calvert Cliffs Design In The Radial and Axial Directions	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 8.1.3.3
14	CA06297 Rev.1	6/2/04	NUHOMS-32P, Transfer Thermal Analysis 103°F Ambient	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.3
15	CA06298 Rev.0	9/25/03	NUHOMS-32P, Transfer Thermal Analysis -3°F Ambient	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.3
16	CA06313 Rev.0	1/15/04	Thermal Analysis of Transfer Cask with Poison Material in Basket	Decay Heat		CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3.3

Table 1
Summary of Documents Screened
(continued)

Item #	Document No Revision No.	Date Issued	Title	Time Dependent Parameter	Resolution	Comments
17	CA03950 Rev. 0	09/19/97	Heating6 Data Reduction	Decay Heat	Helium Temperature Calculated with Decay Heat at the time of Initial Storage.	CCNPP ISFSI USAR Rev. 17, Section 8.1.3
18	CA03947 Rev. 1	07/19/00	DSC Internal Pressure	A. Decay Heat B. Cladding Thickness	Internal Pressure Calculated with Decay Heat at the time of Initial Storage.	CCNPP ISFSI USAR Rev. 17, Section 3.1.1.1 & Section 8.2.9
19	CA06300 Rev. 1	6/2/04	Maximum Operating Pressure, Storage and Transfer	Decay Heat	Helium Temperature Calculated with Decay Heat at the time of Initial Storage. Internal Pressure Calculated with Decay Heat at the time of Initial Storage.	CCNPP ISFSI USAR Rev. 17, Section 12.8.1.3
20	CA06637 Rev.0	10/26/05	Sensitivity Analysis of Homogenized Fuel Region	Decay Heat	Helium Temperature Calculated with Decay Heat at the time of Initial Storage. Internal Pressure Calculated with Decay Heat at the time of Initial Storage.	CCNPP ISFSI USAR Rev. 17, Section 12.8.2.9
21	M-94-108 Rev. 0	06/24/94	Compute Maximum Levels of Oxidizing Gas Left in DSC No.1 and No.2	Degradation of Fuel Cladding due to Oxidizing Gases	The oxidizing gases are below the 0.25% concentration. Therefore no effect on the long term degradation.	
22	PNL-6189, UC-85	May, 1987	Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas	Cladding Thickness & Max. Temp. Limit	See Section 4.1	
23	CA04132 Rev.3	02/27/01	NUTECH Horizontal Module System (NUHOMS) 24P ISFSI Dry Shielded Canister Structural Analysis For DSC Assemblies R001 Through R024	A. Fatigue Design B. Support Rail Friction (Jammed DSC)	Unchanged for TLAA. See Section 4.1	SER Sec. 2.2.3.2 [11] & Appendix C.4 of Topical Report [2]
24	CA04977 Rev.1	02/27/01	NUTECH Horizontal Module System (NUHOMS) 24P ISFSI Dry Shielded Canister Structural Analysis For DSC Assemblies R025 and Beyond	A. Fatigue Design **(See Page 64)** B. Support Rail Friction (Jammed DSC, **See Page 58**)	Unchanged for TLAA. See Section 4.1	SER Sec. 2.2.3.2 [11] & Appendix C.4 of Topical Report [2]
25	CA04141 Rev.3	10/14/03	ISFSI Transfer Cask Structural Analysis	A. Fatigue Design B. Support Rail Friction **(Jammed DSC, See Page 48)**	Unchanged for TLAA. See Section 4.3	

Table 1
Summary of Documents Screened
(concluded)

Item #	Document No Revision No.	Date Issued	Title	Time Dependent Parameter	Resolution	Comments
26	CA06359 Rev.5	9/27/05	NUHOMS-32P DSC Structural Analysis	A. Fatigue Design (**See Page 19**) B. Support Rail Friction (Jammed DSC, **See Page 21**)	Unchanged for TLAA. See Section 4.1.	CCNPP ISFSI USAR Rev. 17, Section 12.8.1.2
27	CA06329 Rev.1	2/9/05	NUHOMS-32P Transfer Cask Structural Analysis	A. Fatigue Design B. Support Rail Friction (Jammed DSC, **See Page 42**)	See Section 4.3	CCNPP ISFSI USAR Rev. 17, Section 12.8.1.1.9
28	CA05955 Rev 0	06/02/02	ISFSI Structural Analysis of the Transfer Cask Yoke System	Lifting yoke allowable fatigue cycles **(See Page 18)**	See Section 4.3	
29	CA06327 Rev.0	12/5/03	Shielding Evaluation with the new Top Shield Plug for NUHOMS 32P	Dose Levels	Shielding Evaluation with source strength at the time of initial storage.	CCNPP ISFSI USAR Rev. 17, Section 12.7.3.2
30	CA06292 Rev.0	10/10/03	NUHOMS-32P Radiation Dose Rates Loading and Transfer	Neutron and Gamma Shielding with NS-3 in Transfer Cask	See Section 4.3	
31	ES200200585 Rev.0	07/23/02	Evaluation of the Shielding Source Terms for the ISFSI 32P Phase I Design	Neutron and Gamma Radiation Source Strength	Shielding Evaluation with source strength at the time of initial storage.	CCNPP ISFSI USAR Rev. 17, Section 12.7.3.2
32	CA05924 Rev.0	09/18/02	Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer	Neutron and Gamma Shielding with NS-3 in Transfer Cask	See Section 4.3	
33	CA05925 Rev.0	09/18/02	Calvert Cliffs ISFSI/NUHOMS-24P HSM Dose Rates	Dose Levels	Shielding Evaluation with source strength at the time of initial storage.	
34	CA05926 Rev.0	09/18/02	Calvert Cliffs ISFSI/NUHOMS-24P Occupational Dose Rates	Dose Levels		
35	CA06750 Rev. 0	08/12/08	Loading and Transfer Dose Rates for ISFSI 32P Burnup Extension	Neutron and Gamma Shielding with NS-3 in Transfer Cask	See Section 4.3	
36	CA06751 Rev.0	09/23/08	Horizontal Storage Module Dose Rates For ISFSI 32P Burnup Extension	Dose Levels	Shielding Evaluation with source strength at the time of initial storage.	
37	CA06771 Rev .0	09/16/08	Effect of updated Fuel Performance Data on NUHOMS® 32P+ DSC Internal Pressure	A. Decay Heat B. Cladding Thickness	CA06300, Rev 1 for the 32P DSC used the bounding fission plus fill gas per fuel assembly.	

4.0 ISFSI MATERIALS AND ANALYSES REVIEW

4.1 DSC Materials

Stainless Steel

The 24P and 32P DSC construction uses Stainless Steel 304 material and compatible weld metal. Since the DSC is filled with the inert Helium gas there is no significant corrosion of the DSC shell and other components. Neutron fluence can affect mechanical properties of steels. However, studies on fast neutron damage in stainless steel and low alloy steels rarely evaluate damage at fluence levels below 10^{17} n/cm^2 because it is not significant [1]. For the DSC, the neutron fluence is much less than this level for the intended storage period and hence, a TLAA is not required.

The DSC and weld stresses due to temperature and pressure inside the DSC are an important aspect of the design. The 24P DSC normal operating design temperature and pressure are 400°F and 10 psig and the accident temperature and pressure are 460°F and 50 psig [Calc # CA04132 Rev.003 for DSC Number R001 to R0024, Calc # CA04977 Rev. 1 for DSC Numbers R025 and above, See Table 1].

The 32P DSC normal operating design temperature is 460°F for the DSC Shell and 380°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 30 psig. The design temperatures for the accident conditions for the 32P DSC Shell is 575°F and 475°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 100 psig [Calc # CA06359, Rev 5, See Table 1]. The accident pressure values are established using 100% cladding breach [Calc# CA06300 Rev. 1, See Table 1].

The design temperatures are calculated at the beginning of NUHOMS® storage. The DSC and weld stress calculations presented in the calculations listed above are bounding since the DSC temperatures are shown to monotonically decrease as a function of time.

Therefore, the heating effect (and hence, the internal pressure effect) on the DSC for the future 40 years of service will be much less severe than that during the past 20 years of service. Hence, the stresses in the DSC components will be acceptable for the extension period.

Topical Report Section 8.1.1.2 C [2] and Section 4.1 thru 4.3 of [12] documents that sufficient clearances are provided in both the radial and axial direction between the DSC internal components to permit free thermal expansion for 24P and 32P DSCs, respectively. This design feature acts to minimize the thermal cycling and fatigue on the DSC. There will be more room for free thermal expansion as the decay heat from the fuel decreases causing the DSC internal component temperatures to decrease as the storage time is increased from 50 years to 60 years. Therefore, thermal cycling and fatigue on the DSC will not be impacted when the storage period is increased from 50 years to 60 years.

Lead in the Shield Plugs

The DSC uses lead in the shield plugs. The lead shielding function is not adversely affected by the radiation level in the DSC.

Poison Plates (32P DSC Only)

Since the 32P DSC for the Calvert Cliffs ISFSI design uses fixed neutron absorbers in the NUHOMS®-32P basket, calculations were performed to assure criticality safety. Time dependency for criticality calculations may result due to depletion of boron in the poison plates utilized in the 32P DSC baskets.

SER for Amendment No.6 for Calvert Cliffs ISFSI Material License No. 2505 [13], Section 6.3, states that "The continued efficacy of the neutron absorber materials over a 20-year storage period is assured by the design of the NUHOMS-32P canister which ensures that the neutron absorbers will remain in place during accident conditions. Additionally, the neutron flux from the irradiated fuel will result in negligible depletion of the B^{10} content in the neutron absorber materials over the life of the storage system."

The neutron activity in the NUHOMS-32P DSC with fuel burnup limit of $\leq$ 47,000 MWD/MTU is 3.3×10^8 n/s/assembly (Section 3.2 of Calc # CA06292 Rev. 0, Table 1) and that for the 32P DSC with extended burnup limits of $\leq$ 52,000 MWD/MTU is 4.175E+08 n/s/assembly (See Table 2-4 of Calc # CA06750 Rev.0). The surface area of the fuel compartment is 34,658 cm^2.

Surface Area of the Fuel Compartment = Length of the Basket x Fuel Comp. Opening
= 158 in * 8.5 in * 4 * (2.54^2) cm^2/in^2
= 34658 cm^2

Assuming a conservative, 25,000 cm^2 surface area per FA the scalar flux is estimated to be 1.67×10^4 n/cm^2 s (4.175E+08 n/s/assembly / 25000 cm^2 = 1.67×10^4 n/cm^2 s).

Considering a sub critical multiplication factor (k_{eff}) of 0.6 and a peaking factor of 1.2, the total scalar flux is approximately 9×10^4 n/cm^2 s (1.67×10^4 n/cm^2 s x 1.2^4 x $(1-0.6)^{-1}$ = 8.65×10^4 n/cm^2 s).

For the total storage period of 60 years, the total neutron fluence is 1.7×10^{14} n/cm^2. The neutron fluence for the 60 year period ignores that the actual flux is mostly fast and epithermal, and declining with time.

Total Neutron Fluence for 60 years = 9×10^4 n/cm^2 s * (60*365*24*60*60) s

= 1.7×10^{14} n/cm^2.

Reference 23, specifies a minimum areal density 10 mg B^{10}/cm^2 for the neutron absorber poison plates. However for conservatism, considering an areal density of 7.5 mg B^{10}/cm^2, the total B^{10} atoms available at the beginning of storage period are 4.5×10^{20} B^{10} atoms/cm^2.

$$\text{Number of } B^{10} \text{ atoms} = 7.5 \text{ mg } B^{10}/cm^2 * 6.023 * 10^{23} \text{ atoms} / 10*1000 \text{ mg}$$
$$= 4.5 \times 10^{20} \text{ atoms}/cm^2$$

As seen above, the total number of B^{10} atoms/cm^2 available exceeds the neutron fluence for the total storage period of 60 years assuring the continued efficacy of the neutron absorber materials. This conservatively assumes that all neutrons are absorbed in the B^{10}, however, most neutrons do not get absorbed and contribute to the HSM dose rate during storage.

DSC Fatigue Evaluation

The fatigue evaluation of the NUHOMS®-24P DSC is documented in Section C.4.1 of reference [2] for a 50 year operational life of the ISFSI. The fatigue effects on the 24P DSC were addressed using the criteria contained in Section III NB-3222.4 of the ASME Code [9]. The analysis evaluated the DSC under six criteria and concluded that the DSC and other components satisfy these criteria and that no consideration of fatigue is required for a service life of 50 years.

In order to extend the operational life by another 10 years, it is necessary to re-evaluate the DSC against these six criteria using an approach that is consistent with that utilized in reference [2]. The fatigue analysis for the 24P and 32P DSC are presented in Reference [19] for a 60 year service life. Reference [19] uses the six criteria in NB-3222.4 (d) of ASME Code [16] and determines that the DSC service loads of the NUHOMS®-24P DSC and NUHOMS®-32P DSC systems do not create any potential risk of the DSC design fatigue failure and that no detailed fatigue evaluation is necessary.

Fuel Cladding Thickness and Maximum Temperature

The NRC SER [11] for the Baltimore Gas and Electric Company's SAR for an ISFSI at Calvert Cliffs, Section 2.2.6, Thermal Evaluation, states that "The applicant uses the same methods as in the NUHOMS-24P topical report [2] for calculating temperature limits for the safe dry storage of spent fuel which resulted in a maximum temperature limit of 335°C for normal operating conditions and 570°C for abnormal and accident conditions." Further, in order to ensure that the cladding temperature limits are acceptable, three limitations were placed on fuel assemblies that are considered for storage in the NUHOMS®-24P DSC:

1) Maximum burnup and associated maximum decay time of either: of 50,000 MWD/MTU and 12 years; 45,000 MWD/MTU and 14 years; or 42,000 MWD/MTU and 15 years

2) Maximum initial fill gas pressure less than 435 psia and

3) Maximum individual fuel assembly decay heat of no more than 660 Watts.

The SER further states that "The 335°C limit is conservative when compared to the 340°C temperature limit in the NUHOMS 24P topical report. The 570°C accident temperature limit is also conservative when compared to the measured inert gas (e.g. helium) environment fuel rod failure temperatures of from 765° to 800°C."

The technical basis report for cladding temperature limits, in its concluding section states that "The CSFM model produces a family of generic temperature limit curves to accommodate the variations due to spent fuel storage parameters. These limit curves establish the recommended temperature limits for 40-year dry storage of zircaloy-clad spent fuel." The report also states that "The CSFM methodology has been verified to be technically sound and defensible, and it has a demonstrable degree of conservatism."

The maximum fuel cladding temperature at the initiation of storage for the 24P DSC was calculated to be 322°C (612°F) (Section 2.2.6.2 of [11]) for long-term storage ambient temperature of 70°F.

The maximum fuel cladding temperature at the initiation of storage for the 32P DSC was calculated to be 313°C (596°F) (Calc# CA06312, Rev 0, See Table 1 and Section 12.8.1.3.2 of [6]), for long-term storage ambient temperature of 70°F.

After 20 years of storage the peak cladding temperatures in both the 24P DSC and 32P DSC are expected to be much lower (Figure 8.1-28 of reference [2]) due to the monotonic reduction in decay heat as a function of time. Therefore, the fuel cladding temperatures will remain below the limits for the additional 40 years of storage.

The PNL-6189 report [5] which is used for calculating cladding temperature limit is applicable to 40 years storage and the methodology does not provide for extension to 60 years of storage. The current guidance from NRC on acceptable fuel cladding temperature limits is given in ISG-11, Rev 3 [18] and no limit on storage term is stated. The maximum cladding temperature from the PNL-6189 method (635°F, [5]) is well below the ISG-11 limit of 752 °F.

Hence no further analysis of the cladding thickness or maximum temperature is necessary.

4.2 Storage Module Concrete and Steel

Concrete

The storage module is a reinforced concrete structure. The effect of radiation on the HSM concrete is evaluated in Section 8.1.1.5.D and Section 12.8.1.1.5.D of Calvert Cliffs ISFSI USAR [6]. This evaluation demonstrates that the magnitude of the neutron fluence incident on the concrete is low enough to not affect the properties of the concrete. This

evaluation also demonstrates that the magnitude of the gamma-ray energy deposition on the concrete is not sufficient to cause any radiation heating in the concrete. Therefore, the thermal analyses documented in [Calvert Cliffs ISFSI USAR [6] implicitly considered the radiation heat effects adequately.

The maximum predicted temperature of concrete at the beginning of storage with 24P DSC was estimated to be below 150°F (Table 5-1 of Calc# CA03948 Rev.0, See Table 1) using a bounding decay heat at the beginning of storage life.

The maximum temperature of the concrete at beginning of storage with a 32P DSC is 157°F (Calc# CA06301 Rev.0, See Table 1).

The maximum concrete temperatures for both the 24P and 32P DSCs in HSM are below the maximum concrete temperature limit of 200°F (Calc# CA06301 Rev.0, See Table 1).

The maximum concrete temperatures for the additional 40 years of service (as shown in Figure 8.1-27 of reference [2]) will be lower because the decay heat reduces monotonically as a function of time. Hence, the heating effect on the concrete, for the future 40 years of service, will be much less severe than the past 20 years of service.

The environmental degradation of reinforced concrete will not be significant, as proper concrete cover has been provided to the reinforcing bars made of carbon steel.

The Topical Report Section 8.2.10.6 [2] documents the analysis of thermal cycling of HSM based on the 50 year storage life. The number of cycles will increase from 18,250 to 21,900 when the design life is extended from 50 years of storage to 60 years of storage. These are still significantly below the limit of 10,000,000 (See Section 8.2.10.6 of [2]). Therefore, thermal cycling will have negligible impact on the HSM reinforced concrete.

DSC Support Rail Steel

The DSC support structure is designed to support the DSC during normal, off-normal and accident conditions. Since the DSC Support rails are fabricated from NITRONIC 60 austenitic stainless steel, it is expected that there would be no corrosion of the rail material and is expected to maintain its function for the additional 40 years of service.

DSC Support Rail Lubricant

The HSM and transfer cask support rails are coated with a dry film lubricant Perma-Slik to minimize friction during insertion and retrieval of the DSC [8]. The material specification of the lubricant indicates that these are suitable of very high and cryogenic temperature applications. The presence of a non-corrosive environment due to the absence of a formal sea breeze and relatively milder temperature fluctuations at CCNPP ISFSI site [7] ensure that the lubricant does not degrade with age. The effect of radiation on these lubricants is not specified, however, it is expected that it is minimal since these are inorganic and consist entirely of graphite, a moderating material.

24P DSC

The coefficient of friction associated with these lubricants is below 0.05 while the design basis calculations employed a coefficient of friction of 0.25 (Section 8.1.1.4 of reference [2]). The mechanical system to be used for DSC transfer is capable of exerting a force equal to the loaded weight of a DSC and this condition has been evaluated in Section 8.1.2.1 of reference [2] for the 24P DSC. A coefficient of friction of 1.0 has been used (for these "jammed DSC" analyses) without relying on the solid film lubricant. The support structure is designed for this loading. Hence, no further analysis is required.

32P DSC

The coefficient of friction associated with these lubricants is below 0.05 while the design basis calculations employed a coefficient of friction of 0.25 (Section 4.1.7.2 of CA06359, Rev.5, See Table 1). The mechanical system to be used for DSC transfer is capable of exerting a force equal to the loaded weight of a DSC and this condition has been evaluated in Section 4.1.7.4 of CA06359, Rev.5 for the 32P DSC. A coefficient of friction of 1.0 has been used (for these "jammed DSC" analyses) without relying on the solid film lubricant. The support structure is designed for this loading. Hence, no further analysis is required.

4.3 Transfer Cask

Transfer Cask Fatigue Evaluation

The fatigue evaluation of transfer cask is documented in reference [21] and is performed in accordance with ASME code criteria listed in Section NC-3219.2 to determine whether the transfer cask service loads of NUHOMS®-24P and NUHOMS®-32P systems create potential risk of the design fatigue failure. The criteria evaluation, documented in the reference [21], shows that transfer cask service loads of NUHOMS®-24P and NUHOMS® -32P systems do not create potential risk of the TC design fatigue failure, for 600 cycles in 60-year transfer cask life, and that detailed fatigue evaluation is not necessary.

Transfer Cask Trunnions, Lifting Yoke Fatigue Evaluation

The fatigue evaluation of transfer cask trunnions and lifting yoke system is documented in reference [20] shows that the transfer cask operations do not pose potential risk of fatigue failure of trunnion or trunnion sleeve throughout the planned 60-year service time.

In case of yoke assembly, reference [20] demonstrates that the structural adequacy against fatigue failure is secured for up to 286 transfer cask loading/unloading operations in planned 60-year service time.

The CCNPP ISFSI allows for the construction and operation of a total of 120 HSMs, and considering one loading and conservatively one unloading transfer cask operation for the

120 DSCs to be stored in the HSMs, a total of 240 transfer cask operations are estimated throughout the 60 year life of the ISFSI and are below the number of cycles allowed for the transfer cask, trunnions and lifting yoke as noted above.

NS-3 in Transfer Cask

The transfer cask contains 3" NS-3 neutron shielding sandwiched between the cask outer shell material and neutron shield jacket. The gamma and neutron dose at 1" from the cask surface for the accident conditions is 135 mrem/hr and 1000 mrem/hr, respectively for the 24P DSC and 85 mrem/hr and 1433 mrem/hr, respectively for the 32P DSC with burnup limit of ≤ 47,000 MWD/MTU [Table 5-6 of Calc# CA06292, Rev. 0, See Table 1] and 79 mrem/hr and 1673 mrem/hr, respectively for the 32P DSC with extended burnup limit of ≤ 52,000 MWD/MTU [Table 6-1 and Table 6-4 of Calc # CA06750, Rev.0, See Table 1] . Per "Calc# CA06292, Rev. 0", the dose rates at 1" from the cask surface for the accident conditions with the NS-3 at the side of the cask replaced with air bounds the dose rates at the inner surface of NS-3 in the cask during normal conditions and that the cask is only subjected to this gamma exposure when a loaded DSC is in the cask during loading and transfer operations which are short term durations. Also considering the effect of the NS-3 thickness, Cask outer shell and surrounding air, the gamma dose of 135 mrem/hr is conservatively increased by a factor of 4. This results in a gamma dose of approximately 3×10^5 Rads over the service life of 60 years. This calculation is based on an assumption that 1 Rad = 1Rem (Section 9.26 of reference [10]) and is considered reasonable for gamma radiation for hydrogenous materials. This is significantly below the exposure limit of 1.5×10^{10} Rads [4] for the material.

To estimate the neutron fluence, a neutron dose to flux factor of 1mrem/hr = 100 n/cm^2-s is used [15]. The dose to flux factor for neutrons is based on dose rate spectra results from various NUHOMS® ISFSI evaluations. The integrated fluence is estimated to be approximately 3.16×10^{14} neutrons/cm^2 over the service life of 60 years for the NS-3 in the transfer cask. Note that the thermal neutron exposure limit is 1.5×10^{19} neutrons/cm^2 [4] for the NS-3 material. Therefore, it is concluded that there is no significant degradation for the additional 40 years of storage operations.

The exposure to radiation sources for an additional 40 years of service is shown to have no significant impact on the shielding capability of the NS-3 in the transfer cask.

Integrated Fluence = 1673 (mrem/hr) *(100 n/cm^2-s/mrem/hr)*(60*365*24*60*60 seconds)

$$= 3.16 \times 10^{14} \text{ neutrons/cm}^2$$

Therefore, no significant hydrogen loss in the NS-3 material is expected due to radiation exposure.

5.0 CONCLUSIONS

Based on the methodology discussed in Section 2 of this report, the following two items were identified as TLAA items. These items needed further evaluation to qualify their functional adequacy for additional 40 years of service.

1. Applicability of the DSC fatigue evaluations.
2. Applicability of the Transfer Cask, Transfer Cask trunnion, and Lifting Yoke system fatigue evaluations.

For item no. 1, the DSC was re-evaluated for fatigue considerations for an additional 40 years of required service using alternative (ii) discussed in Section 2. The results of this evaluation demonstrate that the DSC design meets the fatigue considerations, per the six fatigue criteria of the ASME code [16], for a service life of 60 years.

For item no. 2, the transfer cask Lifting Yoke load cycles in 60 years of service life are limited to 286 transfer cask loading/unloading operations from fatigue considerations.

In summary, all the items were found to be functionally adequate for a total service life of 60 years. Hence, it is concluded that, the entire NUHOMS® system with 24P and 32P DSCs will adequately perform for additional 40 years of service.

However, if any of the loading criteria is redefined as a result of this time extension, due to the recurrence interval of the loading, then a further review of the affected design documents may be required.

6.0 REFERENCES

1 Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988

2 Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS® -24P (NUH002.0100), Rev. 1A, NRC Docket No. M-39.

3 License for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, License No: SNM-2505

4 BISCO Products, Inc., "NS-3 Castable Neutron and/or Gamma Shielding Material," specification sheet. Transnuclear File No. DUK003.0016.2.

5 Pacific Northwest Laboratory Report PNL-6189/UC-85, "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas," by Levy I. S. et al, May 1987.

6 Calvert Cliffs Independent Spent Fuel Storage Installation, Updated Safety Analysis Report, Rev. 17.

7 Calvert Cliffs Independent Spent Fuel Storage Installation, Updated Environmental Report, Rev. 1.

8 Technical Data on Everlube Perma-Slik RN and RGAC Specification Sheets.

9 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1983 Edition.

10 Samuel Glasstone and Alexander Sesonske, "Nuclear Reactor Engineering," Third Edition, Van Nostrand Reinhold Press.

11 NRC Safety Evaluation Report for the Baltimore Gas and Electric Company's Safety Analysis Report for an Independent Spent Fuel Storage Installation at Calvert Cliffs, November 1992.

12 Calculation, CA06340, "NUHOMS 32P – Thermal Expansion Calculation for Transfer and Storage Conditions", Revision 0.

13 NRC Safety Evaluation Report for Amendment No.6 for Calvert Cliffs ISFSI Material License No. 2505

14 *Not Used.*

15 CoC 1030, Updated Final Safety Analysis Report for the NUHOMS® HD System for Irradiated Nuclear Spent Fuel, Revision 2.

16 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsections NB, 1998 Edition including 1999 Addenda.

17 NRC Safety Evaluation Report for Amendment No.7 for Calvert Cliffs ISFSI Material License No. 2505.

18 "Cladding Considerations for Transportation and Storage of Spent Fuel", Interim Staff Guidance-11, Rev.3, NRC.

19 Transnuclear Calculation, 10955-0202, "DSC Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P", Rev.0.

20 Transnuclear Calculation, 10955-0201, "Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P Transfer Cask Trunnion and Lifting Yoke System", Rev.0.

21 Transnuclear Calculation, 10955-0203, "Transfer Cask Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P", Rev.0.

22 Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation, Docket No. 72-8, License Amendment Request: Allow Increased Burnup fuel to be loaded into NUHOMS-32P Dry Shielded Canister, 15th June 2009.

23 Transnuclear, Inc., "Procurement Specification for Borated Aluminum Sheets for the NUHOMS®-32P Dry Shielded Canister", E-20112, Rev. 3.

7.0 APPENDIX A

This appendix presents a summary of the documents reviewed as a part of the TLAA but are not included either in Table 1 or in the references listed in Section 6.0.

The following are the short list of the documents reviewed from the list of all the 72.48s provided by CCNPP.

List of 72.48's Reviewed

94-0-101-002-R00	EVALUATES THE EFFECTS OF USING THE ISFSI SITE FOR STORAGE OF NEW EMPTY DRY SHIELDED CANISTERS
SE00005	ISFSI SAFETY EVALUATION OSC SPACER DISK & SUPPORT ROD MATERIAL
SE00012	ISFSI SAFETY EVALUATION SPACER DI SC SURFACE FINISH REQUIREMENTS
SE00025	ISFSI SAFETY EVALUATION ADDI TION OF SHIELD PLUG BACKING BAR
SE00033	ISFSI SAFETY EVALUAT ION UPPER & LOWER TRUNNION DESIGN CHANGES
SE00037	ISFSI SAFETY EVALUATI ON TRANSFER CASK SURFACE FINISH REQUIREMENTS DESIGN CHANGE
SE00045	ISFSI SAFETY EVALUATION TRANSFER CASK TOP COVER PLATE MATERIAL DESIGN CHANGE
SE00048	ISFSI SAFETY EVALUATION TRANSFER CASK SHIELD PLUG MATERIAL DESIGN CHANGE
SE00054	ISFSI SAFETY EVALUATION TRANSFER CASK LEAD SILVER CONTENT NONCONFORMANCE
SE00055	ISFSI SAFETY EVALUATION TRANSFER CASK SHELL 0.D. MAXIMUM TEMPERATURE NON CONFORMANCE
SE00056	ISFSI SAFETY EVALUATION TRANSFER CASK INNER LINER I.D. MAXIMUM TEMPERATURE NON CONFORMANCE
SE00062	ISFSI SAFETY EVALUATION TRANSFER CASK NITRONIC RAIL MINIMUM WIDTH NON CONFORMANCE
SE00078	ISFSI SAFETY EVALUATION HSM CONTACT DOSE RATES
SE00096	ISFSI SAFETY EVALUATION HSM MINIMUM CONCRETE COVER OF VERTICAL OUTLET VENT REBAR
SE00101	ISFSI SAFETY EVALUATION - HSM#8 BAR ADDED FOR CRACK CONTROL
SE00102	ISFSI SAFETY EVALUATION HSM 5 CC 69 BARS ADDED FOR CRACK CONTROL
SE00120	ISFSI SAFETY EVALUATION HSM RAIL LUBRICANT CHANGE
SE00127	ISFSI SAFETY EVALUATION HSM BEAM SUPPORT ANGLE CHANGE
SE00133	ISFSI SAFETY EVALUATION ANALYSIS OF DSC UNDER POSTULATED CASK DROP ACCIDENT
SE00136	ISFSI SAFETY EVALUATION ANALYSIS OF DSC UNDER BLOWDOWN REFLOOD INTERNAL PRESSURE
SE00145	72.48 ADDRESSES CHANGES TO ECN 98 - 0516
SE00146	ISFSI EVALUATION OF LOADED DSC'S WITH UNANALYZED CLIPS
SE00152	ISFSI - FOREST FIRE RE-ANALYSIS - SAR REVISION
SE00154	ISFSI - FUEL AS SEMBLY INTEGRITY
SE00155	ISFSI - USAR UPDATE OF DSC PEAK PRESSURE
SE00158	ISFSI - INDEPENDENT TECHNICAL REPORT (LTR) ISSUES
SE00165	EXTEND NUHOMS-32P TS 3.1.1 BURNUP LIMIT OT 52 GWD/MTU

Calculations

1) CCNPP Calculation CA05797, DSC Horizontal Drop – Fuel Rod Cladding Integrity during Impact with a Broken Spacer Grid Fragment.

2) CCNPP Calculation CA05760, Effective Conductivity of the Reconfigured Fuel Assembly

3) BGE001.0202, "TRANSFER CASK STRUCTURAL ANALYSIS"

4) BGE001.0203, "DRY SHIELDED CANISTER STRUCTURAL ANALYSIS"

5) *CCNPP Calculation CA06364,* "NUHOMS 32P CCNP ISFSI HSM Facility Evaluation"

6) CCNPP Calculation CA06770, Thermal Analysis of NUHOMS 32P+ DSC for Vacuum Drying Conditions

Drawings

1) CCNPP Drawing 84003 *BGEDRWG*, NUHOMS-24 ISFSI DSC Shell Assembly, *Rev. 0000.*

2) CCNPP Drawing 84004 *BGEDRWG*, NUHOMS-24P ISFSI DSC Shell Assembly, *Rev. 0000.*

3) CCNPP Drawing 84006 *BGEDRWG*, NUHOMS-24P ISFSI DSC Main Assembly. *Rev. 0000.*

4) CCNPP Drawing 84007 *BGEDRWG*, NUHOMS-24 ISFSI DSC Main Assembly, *Rev. 0001.*

5) CCNPP Drawing 84001 *BGEDRWG*, NUHOMS-24P ISFSI DSC Basket Assembly, *Rev. 0001.*

6) CCNPP Drawing 84005 *BGEDRWG*, NUHOMS-24P ISFSI DSC Shell Assembly, *Rev. 0000.*

7.) CCNPP Drawing 84218 *BGEDRWG*, NUHOMS-32P DSC Shell & Bottom Plug Assembly, *Rev. 0000.*

8.) CCNPP Drawing 84219 *BGEDRWG*, NUHOMS-32P DSC Shell & Siphon Pipe Assembly Details, *Rev. 0000.*

9.) CCNPP Drawing 84220 *BGEDRWG*, NUHOMS-32P DSC Siphon Pipe/Adapter & Lifting Block Details, *Rev. 0000.*

10.) CCNPP Drawing 84221 *BGEDRWG*, NUHOMS-32P DSC Top Shield Plug Details, *Rev. 0000.*

11.) CCNPP Drawing 84222 *BGEDRWG*, NUHOMS-32P Top Cover Plate & Siphon/Vent Port Covers, *Rev. 0000.*

12.) CCNPP Drawing 84223 *BGEDRWG*, NUHOMS-32P DSC Basket Assembly, *Rev. 0000.*

13.) CCNPP Drawing 84224 *BGEDRWG*, NUHOMS-32P DSC Basket Details, *Rev. 0000.*

14.) CCNPP Drawing 84225 *BGEDRWG*, NUHOMS-32P DSC Basket Rails & Shims, *Rev. 0000.*

15.) CCNPP Drawing 84226 *BGEDRWG*, NUHOMS-32P DSC Basket Plate & Rail Inserts, *Rev. 0000.*

16.) CCNPP Drawing 84227 *BGEDRWG*, NUHOMS-32P DSC Parts List, *Rev. 0000.*

17.) CCNPP Drawing 84234 *BGEDRWG*, NUHOMS-32P DSC Final Assembly, Field Welding &Testing, *Rev. 0000.*

18.) CCNPP Drawing 84235 *BGEDRWG*, NUHOMS-32P DSC ASME Code Exceptions, *Rev. 0000.*

19.) CCNPP Drawing 84021 *BGEDRWG*, NUHOMS-24P ISFSI Onsite Transfer Cask Structural Shell, *Rev. 0000.*

20.) CCNPP Drawing 84025 *BGEDRWG*, NUHOMS-24P ISFSI Onsite Transfer Cask Inner and Outer Shell Assembly, *Rev. 0001.*

21.) CCNPP Drawing 84026 *BGEDRWG*, NUHOMS-24P ISFSI Onsite Transfer Cask Inner and Outer Shell Assembly, *Rev. 0000.*

22.) CCNPP Drawing 84027 *BGEDRWG*, NUHOMS-24P ISFSI Onsite Transfer Cask Main Assembly, *Rev. 0001.*

23.) CCNPP Drawing 84028 *BGEDRWG*, NUHOMS-24P ISFSI Onsite Transfer Cask Main Assembly, *Rev. 0002.*

24.) CCNPP Drawing 84030 *BGEDRWG*, NUHOMS-24P ISFSI Onsite Transfer Cask Accessories, *Rev. 0000.*

25.) CCNPP Drawing 84036 *BGEDRWG*, ISFSI Cask Lifting Yoke Assembly, *Rev. 0003.*

26.) CCNPP Drawing 84022 *BGEDRWG*, NUHOMS-24P ISFSI Onsite Transfer Cask Structural Shell Assembly, *Rev. 0000.*

AREVA TRANSNUCLEAR INC.	Form 3.2-1 Calculation Cover Sheet TIP 3.2 (Revision 4)	Calculation No.:	10955-0201
		Revision No.:	0
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DCR NO (if applicable) :	**PROJECT NAME:** License Renewal for CCNPP
PROJECT NO: 10955	**CLIENT:** CENG - Calvert Cliff Nuclear Power Plant (CCNPP)

CALCULATION TITLE:

Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P Transfer Cask Trunnion and Lifting Yoke System

SUMMARY DESCRIPTION:

1) Calculation Summary

This calculation evaluates the allowable fatigue cycles for the transfer cask top trunnion and lifting yoke of NUHOMS® 24P and NUHOMS® 32P systems.

2) Storage Media Description

If original issue, is licensing review per TIP 3.5 required?

Yes ☐ **No** ☒ **(explain below)** **Licensing Review No.:** _____

This calculation is prepared to support a Site Specific License Application by CCNPP that will be reviewed and approved by the NRC. Therefore, a 10CFR72.48 licensing review per TIP 3.5 is not applicable.

Software Utilized (subject to test requirements of TIP 3.3):	**Version:**
NONE	N/A

Calculation is complete:

Originator Name and Signature: Huan Li — Date: 03/30/2010

Calculation has been checked for consistency, completeness and correctness:

Checker Name and Signature: Jerzy Mackiewicz — Date: 03/30/2010

Calculation is approved for use:

Project Engineer Name and Signature: Kamran Tavassoli — Date: 03/30/10

REVISION SUMMARY

REV.	DESCRIPTION OF CHANGES	AFFECTED PAGES	AFFECTED Computational I/O
0	Initial Issue	All	All

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1.0 Purpose

The purpose of this calculation is to perform fatigue evaluation for the transfer cask top trunnions, and yoke system components of NUHOMS® 24P and NUHOMS® 32P systems, for the life extension of total 60 years. The evaluation is based on stress analysis results documented in References [2.1, 2.2] for NUHOMS® 24P system and References [2.7, 2.8] for NUHOMS® 32P system.

2.0 References

2.1. TN Calculation NUH002.0401, Rev. 2, "Transfer Cask Structural Analysis Package".

2.2. TN Calculation NUH002.0408, Rev. 2, "Design and Analysis of Transfer Cask Yoke System".

2.3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 1992.

2.4. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix I, 1992 Edition.

2.5. Regulatory Guide 7.6, Revision 1, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," U.S. Nuclear Regulatory Commission, March 1978.

2.6. Not used.

2.7. TN Calculation No. 1095-35, Rev. 2, "NUHOMS 32P Transfer Cask Structural Analysis."

2.8. Calvert Cliffs NPP Calculation CA05955, Rev. 0, "ISFSI Structural Analysis of the Transfer Cask Yoke System."

3.0 Assumptions

The following assumptions are employed in calculation:

- The maximum stresses calculated for lifting loads for the NUHOMS® 24P and NUHOMS® 32P systems are used in the fatigue evaluation to obtain the allowable fatigue cycles.
- Only the stress values for the upper trunnions are collected for the lifting loads since the upper trunnion is the critical component during the lifting events.
- The dynamic load factor DLF=1.15 is used in stress range calculation.
- The mean stress for lifting load cycle is defined as S.I./2, where S.I. denotes maximum stress at the loading cycle, and zero-stress denotes minimum stress at loading cycle.
- The stress values, component materials and their properties are taken from References [2.1, 2.2, 2.7, 2.8]. Therefore, all assumptions made in those references are still applicable in the current calculation.

4.0 Fatigue Evaluation Criteria

Suitability of transfer cask top trunnions and yoke system components for the 60-year service loading is based criteria from ASME code, Section III, Subsection NB-3222.4 [2.3] and fatigue curves from the ASME code Appendix I [2.4].

5.0 Methodology and Calculations

The maximum stresses for the transfer cask lifting trunnions and lifting yoke of the NUHOMS® 24P and NUHOMS® 32P systems, calculated in Ref. [2.1, 2.2, 2.7, 2.8] for lifting loads for normal service conditions, are collected and summarized in Table 1 and Table 2, respectively. These stresses are used in the fatigue evaluation to obtain the allowable fatigue cycles.

Table 1 Data for NUHOMS® 24P Transfer Cask Trunnions and Lifting Yoke

TC Assembly	Components	Materials	Material properties (ksi)	S.I. @ 1.15g (ksi)	References
Upper Trunnion (@ 300 °F)	Trunnion	SA-564 Gr. 630 PH	S_u = 135.0 S_y = 93.0 E = 27.05x10^3	5.9 @ Section A 7.3 @ Section B weld	[2.1] pages 9.7, 9.10
	Trunnion Sleeve	SA-182 F304N	S_u = 75.9 S_y = 25.0 E = 27.05x10^3	6.3 @ Section B 7.3 @ Section B weld	Ref. [2.1] pages 9.8, 9.10
Lifting Yoke (@ 70 °F)	Lifting Hook	SA-517 Gr. E	S_u = 105.0 S_y = 90.0 E = 29.7x10^3	(8.66+1.86)x1.15=12.10 (1) @ Section B	[2.2] page 12 in
	Lifting Beam	SA-517 Gr. E	S_u = 105.0 S_y = 90.0 E = 29.7x10^3	8.51x1.15 = 9.79 (1) @ Section a-a	[2.2] page 14 in

Note: (1) These stresses in Ref. [2.2] were calculated for 1g load; therefore, they are multiplied by a factor of 1.15.

Table 2. Data for NUHOMS® 32P Transfer Cask Trunnions and Lifting Yoke

TC Assembly	Components	Materials	Material properties (ksi)	Max. Stress @ 1.15g (ksi)	References
Upper Trunnion (@ 300 °F)	Trunnion	SA-564 Gr. 630 PH	S_u = 135.0 S_y = 93.0 E = 27.05x10^3	6.5 @ Section A 7.0 @ Section B	[2.7] pages 92, 93
	Trunnion Sleeve	SA-182 F304N	S_u = 75.9 S_y = 25.0 E = 27.05x10^3	3.73 @ Section C 5.53 @ Section B weld	[2.7] pages 94, 95
Lifting Yoke (@ 70 °F)	Lifting Hook	SA-517 GR. E	S_u = 105.0 S_y = 90.0 E = 29.7x10^3	22.2$^{(1)}$ x 1.088$^{(2)}$ = 24.15	[2.8] page 13
	Lifting Beam	SA-517 GR. E	S_u = 105.0 S_y = 90.0 E = 29.7x10^3	10.3$^{(1)}$ x 1.088$^{(2)}$ = 11.2 @ Section a-a	[2.8] page 20

Notes:
(1) The stress value was calculated for the design weight of 230,000 lbs [2.8]. The actual design weight of the transfer cask lifting with water and loaded DSC is 217,865 [2.7].
(2) The factor of 1.088 is calculated by: 217,865 x 1.15/230,000 = 1.088 so that 15% increase of design weight is taken into account.

Based on the stresses summarized in Tables 1 and 2, the fatigue evaluation of transfer cask upper trunnions and lifting yoke are performed according to the criteria from ASME code, Section III, Subsection NB-3222.4 [2.3] and fatigue curves from the ASME code Appendix I [2.4].

To determine the allowable fatigue cycles using fatigue curves in Appendix I [2.4], the alternating stress, S_a, is calculated by the following equation,

$$S_a = [S.I. \times K_F \times K_E]/2$$

where,

S.I. is maximum stress intensity

K_F is fatigue strength reduction factor, 4 [2.5];

K_E is correction factor for the Young's Modulus.

In the above formula, it is assumed that the mean stress for lifting cycle is S.I/2, with the minimum stress at zero-level and maximum stress at S.I level.

Once S_a is calculated, the corresponding allowable fatigue cycles can be determined [2.4].

The calculation results of the allowable fatigue cycles for TC normal lifting/handling load are summarized in Tables 3 and 4 for the NUHOMS® 24P and NUHOMS® 32P, respectively.

Table 3. Allowable Fatigue Cycles for NUHOMS® 24P systems

TC components	S.I. (ksi)	K_E	S_a (ksi)	Fatigue Curve ID [2.4]	Allowable Fatigue Cycles
Upper Trunnion	7.3	$28.3x10^3/27.05x10^3$ =1.046	15.27	I-9.2.2 curve B	> 10^{11}
Trunnion Sleeve	7.3	$30.0x10^3/27.05x10^3$ =1.109	16.19	I-9.1	> 10^6
Lifting Hook	12.10	$30.0x10^3/29.7x10^3$ =1.010	24.44	I-9.1	1.7×10^5
Lifting Beam	9.79	$30.0x10^3/29.7x10^3$ =1.010	19.78	I-9.1	> 10^6

Table 4. Allowable Fatigue Cycles for NUHOMS® 32P systems

TC components	S.I. (ksi)	K_E	S_a (ksi)	Fatigue Curve ID [2.4]	Allowable Fatigue Cycles
Upper Trunnion	7.0	$28.3x10^3/27.05x10^3$ =1.046	14.64	I-9.2.2 curve B	> 10^{11}
Trunnion Sleeve	5.53	$30.0x10^3/27.05x10^3$ =1.109	12.26	I-9.1	> 10^6
Lifting Hook	24.15	$30.0x10^3/29.7x10^3$ =1.010	48.78	I-9.1	5,147
Lifting Beam	11.2	$30.0x10^3/29.7x10^3$ =1.010	22.62	I-9.1	3.73×10^5

Tables 3 and 4 show that the limiting number of stress fatigue cycles for cask components (trunnions and trunnion sleeve) exceeds 10^6 cycles, and that limiting component is trunnion sleeve. The limiting number of stress fatigue cycles for yoke assembly components is 5,147, and the limiting component is lifting hook.

The limiting number of use cycles for cask components and yoke components will be assessed separately, below.

During a normal canister loading cycle, the transfer cask is subjected to mechanical load fluctuations including lifting and handling load fluctuations as follows:

1. At the start of the use cycle, the cask is lifted from the transport trailer to the plant's fuel building decon pit.
2. The cask and DSC filled with water is lifted from the decon pit and placed in the plant's fuel pool.
3. The cask and DSC filled with water, and loaded with SFAs is lifted from the fuel pool and placed in the decon pit.

4. The cask and dry DSC loaded with SFAs is lifted from the decon pit and is rotated to a horizontal position onto the support skid and transport trailer.

5. The cask with loaded DSC resting on the support skid is subjected to motion loads and braking loads during transport to the HSM along a predetermined route.

6. The cask is secured to the HSM and the DSC is transferred to the HSM with the hydraulic ram. At the end of this use cycle the cask is returned to its designated storage area or to begin the next use cycle.

Taking conservatively into account that each lifting may generate several stress cycles, say 3, and each operation specified above is associated with maximum lifting load, one can estimate that the limiting number for use cycles for trunnion sleeve can be assessed as $10^6/(3*6) = 55555$ cask transfer operations.

Since the limiting number of use cycles is much higher than the number of cycles expected during 60-year life-span of cask transfer operations, there is no expected potential risk of trunnion or trunnion sleeve in that time.

Similarly, limiting number for use cycles for lifting hook can be assessed as $5147/(3*6) = 286$ cask transfer operations.

Therefore, lifting hook can be used in 286 cask transfer operations without concerns of fatigue failure.

6.0 Conclusions

Fatigue evaluation presented above shows that cask transfer operations do not pose potential risk of fatigue failure of trunnion or trunnion sleeve throughout the planned 60-year service time.

In case of yoke assembly, the structural adequacy against fatigue failure can be secured for up to 286 transfer cask loading/unloading operations in planned 60-year service time.

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		Revision No.:	0
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DCR NO (if applicable) : NA	**PROJECT NAME:** License Renewal for CCNPP
PROJECT NO: 10955	**CLIENT:** CENG - Calvert Cliff Nuclear Power Plant (CCNPP)

CALCULATION TITLE:

DSC Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P

SUMMARY DESCRIPTION:

1) Calculation Summary

The DSC fatigue analysis is performed for the NUHOMS® 24P and NUHOMS® 32P.

2) Storage Media Description

N/A

If original issue, is licensing review per TIP 3.5 required?

Yes ☐ **No** ☒ **(explain below)** **Licensing Review No.:** _____

This calculation is prepared to support a Site Specific License Application by CCNPP that will be reviewed and approved by the NRC. Therefore, a 10CFR72.48 licensing review per TIP 3.5 is not applicable.

Software Utilized (subject to test requirements of TIP 3.3):	**Version:**
NONE	N/A

Calculation is complete:

Originator Name and Signature: Huan Li — Date: 03/30/2010

Calculation has been checked for consistency, completeness and correctness:

Checker Name and Signature: Jerzy Mackiewicz — Date: 03/30/2010

Calculation is approved for use:

Project Engineer Name and Signature: Kamran Tavassoli — Date: 03/30/10

REVISION SUMMARY

REV.	DESCRIPTION OF CHANGES	AFFECTED PAGES	AFFECTED Computational I/O
0	Initial Issue	All	All

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Figure 1. Design Fatigue Curve for Austenitic Steels, Nickel-Chromium-Iron Alloy, Nickel-Iron-Chromium-Iron Alloy, and Nickel-Copper Alloy for Temperature Not Exceeding 800 °F [2.5] 11

1.0 Purpose

The purpose of this calculation is to assess the fatigue effects on the NUHOMS® 24P and NUHOMS® 32P DSC designs for a total service life of 60 years.

2.0 References

2.1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsections NB, 1998 Edition including 1999 Addenda.

2.2. ASME Boiler and Pressure Vessel Code, Section II, Part D, 1998 Edition including 1999 Addenda.

2.3. TN Calculation No. 1095-34, Rev. 0, "NUHOMS 32P- DSC Structural Analysis".

2.4. Calvert Cliffs NPP Calculation CA04132, Rev. 3, "NUTECH Horizontal Module System (NUHOMS) 24P ISFSI Dry Shielded Canister (DSC) Structural Analysis for DSC Numbers R001-R024."

2.5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix I, 1998 Edition including 1999 Addenda.

2.6. TN Document NUH002.0103, Rev. 1A, Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, NRC Docket No. M-39.

2.7. Calvert Cliffs NPP Calculation CA04977, Rev. 1, "NUTECH Horizontal Module System (NUHOMS) 24P ISFSI Dry Shielded Canister (DSC) Structural Analysis for DSC Numbers R025 and Beyond."

2.8. Calvert Cliffs Independent Spent Fuel Storage Installation, Updated Safety Analysis Report, Rev 17.

3.0 Methodology and Materials

The fatigue analyses are performed to assess the rank of fatigue effects on the DSCs of NUHOMS® 24P and NUHOMS® 32P under designed service conditions, based on the ASME code in Section NB 3222.4 (d) [2.1] which sets six criteria to determine whether the designed operational conditions require detailed fatigue analysis, or whether the risk of fatigue failure can be qualified as remote and therefore the design can be exempted from the detailed analysis of fatigue.

It is conservatively assumed that the service temperature of the DSC for normal condition is 400 °F for NUHOMS® 24P [2.4, 2.7] and 460 °F for NUHOMS® 32P [2.3]. The mechanical properties of stainless steel SA 240 Type 304 for the DSCs are summarized in Table 1 below [2.2].

Table 1. Mechanical Properties of SA 240 Type 304

DSC Type	Temperature [°F]	S_m [ksi]	S_y [ksi]	S_u [ksi]	E ($x10^3$) [ksi]	CTE ($x10^{-6}$) [in/in/°F]
NUHOMS® 24P	400 [2.4, 2.7]	18.7	20.7	64.0	26.5	9.5
NUHOMS® 32P	460 [2.3]	18.1	20.1	63.7	26.2	9.6

4.0 Calculations and Results

The six criteria in NB-3222.4 (d) of ASME Code [2.1] that determine whether the design, with its service loading pattern and magnitude, can be exempted from detail fatigue failure analyses, are evaluated below:

1. Atmospheric to Service Pressure Cycle

In this requirement, the total specified number of times that the pressure will be cycled from atmospheric pressure to service pressure and back to atmospheric pressure during normal operating conditions (including startup and shutdown) shall not exceed the number of cycles on the applicable fatigue curves corresponding to the $S_a=3\times S_m$ for the material at service temperature.

For the DSCs of the NUHOMS® 24P and NUHOMS® 32P systems, the pressure established in the sealed DSCs is maintained unchanged during the normal storage in the HSM. Therefore, this condition is satisfied since the pressure in the DSC is not cycled during its design life of 60 years.

2. Normal Service Pressure Fluctuation

Per the second requirement, the specified full range of pressure fluctuations during the normal service condition shall not exceed the quantity:

$$1/3 \times \text{design pressure} \times (S_a/S_m)$$

Where,

S_a - is the value obtained from the applicable design fatigue curve for the total specified number of Significant Pressure Fluctuations.

S_m - is the allowable stress intensity for the material at service temperature;

Significant Pressure Fluctuations are those for which the total pressure excursion exceeds the pressure range equal to:

$$1/3 \text{ x design pressure x } (S/S_m)$$

Where,

S - equals to S_a, determined at 10^6 cycles of applicable ASME code fatigue curve [2.5], if the total specified number of service cycles is 10^6 or less.

According to fatigue curve shown in Figure 1 (Figure I-9.2.1 in Ref. [2.5]), S = 28.2 KSI. For the normal condition, the maximum normal operating pressure of DSC is 10.0 psi for NUHOMS® 24P [2.4, 2.7], and 30 psi for NUHOMS® 32P [2.3]. Using the data in Table 1 and the formula above, the Significant Pressure Fluctuation is 1/3 × 10 × (28.2/18.7) = 5.0 psi for NUHOMS® 24P and 1/3 × 30× (28.2/18.0) = 15.7 psi for NUHOMS® 32P. Such pressure fluctuations may occur due to the seasonal ambient temperature changes during normal storage.

According to Reference [2.6], it is assumed that measurable pressure fluctuations, caused by ambient temperature cycles, occur 5 times per year for DSCs. Therefore for the designed operational service life of 60 years, the number of significant pressure cycles is expected to be 600 for DSCs. The corresponding value of S_a associated with this number of cycles is approximately 150 ksi.

Hence, per the second requirement, the pressure fluctuations for NUHOMS® 24P DSCs shall not exceed a value of 1/3 × 10.0 × 150/18.7 = 26.7 psig; while for NUHOMS® 32P DSCs, the pressure fluctuations shall not exceed a value of 1/3 × 30 × 150/18.1 = 82.0 psig. Table 2 summaries the calculated data of pressure fluctuations for both NUHOMS®24P and NUHOMS®32P. It is shown in Table 2 that the allowable values of pressure fluctuations will not be exceeded during the pressure fluctuation of DSCs for both NUHOMS® 24P and NUHOMS® 32P systems. Therefore, the second requirement is satisfied.

Table 2. Data of Pressure Fluctuation for NUHOMS®24P and NUHOMS®32P

DSC	Actual pressure fluctuation (psi)	Allowable pressure fluctuation (psi)
NUHOMS® 24P	5.0	26.7
NUHOMS® 32P	15.7	82.0

3. Temperature Difference---Startup and Shutdown

This requirement states that the temperature difference between any two adjacent points of the component during normal service shall not exceed $S_a/2E\alpha$. The S_a is the value obtained from the applicable design fatigue curve for the specified number of startup-shutdown cycles, α is the value of the instantaneous coefficient of thermal expansion at the mean value of the temperatures at the two points, and E is the Young's Modulus at the mean value of the temperature at the two points.

A normal operational cycle of startup-shutdown for DSC occurs only once for a whole designed service life. According to Fig. I-9.2.1 in Ref. [2.5], the value of S_a is very large (>800 ksi), and therefore the value of $S_a/2E\alpha$ is very high (> 1500 °F). This value is far greater than temperature difference between any two points on the DSCs. Therefore, this condition is met for both NUHOMS® 24P DSCs and NUHOMS® 32P DSCs.

4. *Temperature Difference---Normal Service*

Under this condition, the temperature difference between any two adjacent points shall not change by more than the value of $S_a/2E\alpha$ during the normal service, where S_a is the value obtained from applicable fatigue curve for total number of Significant Temperature-Difference Fluctuations.

A temperature-difference fluctuation is considered significant if its total algebraic range exceeds the quantity $S/2E\alpha$, where S is the value of S_a obtained from the applicable fatigue curve at 10^6 cycles (when the total numbers of thermal cycles equal or less than 10^6).

Small fluctuations of thermal gradients in DSCs during normal storage in HSM occur as a result of seasonal ambient temperature changes [2.6], the ambient temperature cycles causing a measurable thermal gradient fluctuation are assumed to occur 5 times per year [2.6]. Therefore, for an operational life of 60 years, the total number of ambient temperature cycles is 600, which is less than 10^6. Hence, a value of S = 28.2 ksi is taken according to Ref. [2.5].

The value of $S_a/2E\alpha$ for the DSCs of NUHOMS® 24P and NUHOMS® 32P is calculated using the material properties from Table 1, which is 54 °F for both NUHOMS® 24P and NUHOMS® 32P. It is shown in Refs. [2.6] that for NUHOMS® 24P the most significant temperature fluctuation occurs during a change in ambient temperature from 0 °F to 100 °F, which results in a change of temperature difference of 32.3 °F. This temperature difference is less than 54 °F. Therefore, this condition is satisfied for NUHOMS® 24P.

It is known that the temperature fluctuation is proportional to the heat load, the maximum expected temperature fluctuation for the NUHOMS® 32P DSC would be

$$32.3 \times Q_{32p}/Q_{24p} = 32.3 \times 21.12/15.84 = 43.1\ °F$$

Where,

Q_{32p} is the maximum heat load for NUHOMS® 32P, which is 21.12 KW [2.8];

Q_{24p} is the maximum heat load for NUHOMS® 24P, which is 15.84 KW [2.8].

Clearly, these temperature fluctuations for both NUHOMS® 24P (32.3 °F) and NUHOMS® 32P (43.1 °F) are less than 54 °F. Therefore, this condition is satisfied.

5. *Temperature Difference---Dissimilar materials*

This condition states that for the components with different moduli of elasticity or CTE, the total algebraic range of temperature fluctuation experienced by the component during normal service does not exceed the value of $S_a/2(E_1\alpha_1 - E_2\alpha_2)$.

It is known that the DSC is made of homogeneous material (i.e., stainless steel), therefore, this condition is not applicable to the DSC.

6. *Mechanical Loads*

The sixth criterion states that specified full range of mechanical loads, shall not result in load stresses whose range exceeds the S_a value obtained from the applicable design fatigue curve for the total specified number of Significant Load Fluctuations.

A load fluctuation shall be considered significant only if the total excursion of load stress exceeds the quantity S, where S is defined below:

- S is the value of S_a corresponding to the 10^6 cycles from the applicable fatigue curve if the total specified number of service cycle is 10^6 or less.
- S is the value of S_a obtained from the applicable fatigue curve corresponding to the maximum cycles defined on the curve if the total specified number of service cycle exceeds 10^6.

The mechanical loads for the DSC are those associated with handling operations and the seismic events. One handling load cycle and a major seismic event are postulated during the design life of the DSC. The DSC stresses resulting from these mechanical load fluctuations are small since the structural capacity of the DSC is designed for the extreme accident loads such postulated cask drops.

In Ref. [2.6], the number of Significant Load cycles expected to occur during handling operations and seismic events was conservatively taken to be 1000 for the operational life of 50 years.

For the operational life of 60 years, 1000 cycles is still sufficiently conservative – but 1200 cycles are assumed here for the life extension analysis.

From fatigue curve shown in Figure 1 (Fig. I-9.2.1 in Ref. [2.5]) - the S_a value corresponding to the 1200 cycles is S_a=110 ksi (approximation). According to the ASME code the allowable stress for the DSC is $3S_m$ for service levels A & B. Therefore, the allowable stresses for the DSC of NUHOMS® 24P and NUHOMS® 32P systems are calculated to be 3x18.7 = 56.1 ksi and 3x18.0 = 54.0 ksi, respectively. These values are less than the allowed S_a limit of 110 ksi. Visibly, the sixth requirement is satisfied.

5.0 Conclusions

ASME code criteria listed in Section NB 3222.4 (d) are utilized to determine whether the service loads for the NUHOMS® 24P and NUHOMS® 32P DSCs create potential risk of the design fatigue failure. The criteria evaluation shows that DSC service loads of the NUHOMS® 24P and NUHOMS® 32P systems do not create any potential risk of the DSC design fatigue failure, nor that detail fatigue evaluation is necessary.

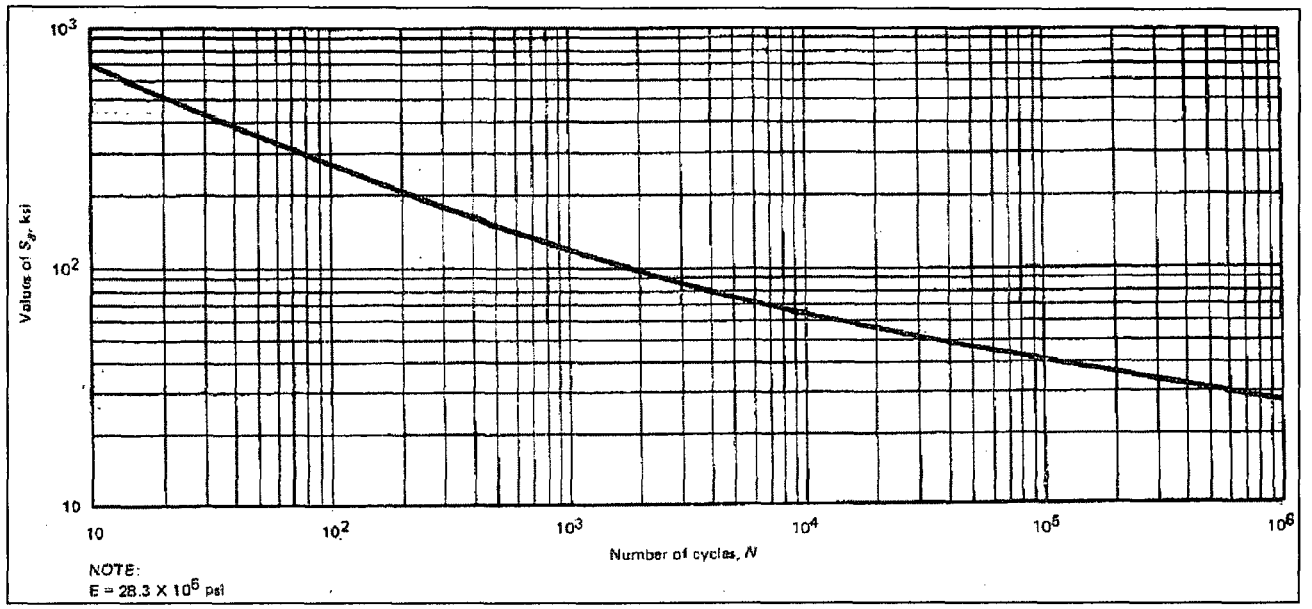

Figure 1. Design Fatigue Curve for Austenitic Steels, Nickel-Chromium-Iron Alloy, Nickel-Iron-Chromium-Iron Alloy, and Nickel-Copper Alloy for Temperature Not Exceeding 800 °F [2.5]

AREVA TRANSNUCLEAR INC.	**Form 3.2-1** **Calculation Cover Sheet** **TIP 3.2 (Revision 4)**	**Calculation No.:**	10955-0203
		Revision No.:	0
		Page: 1 of 12	

DCR NO (if applicable) : NA	**PROJECT NAME:** License Renewal for CCNPP
PROJECT NO: 10955	**CLIENT:** CENG - Calvert Cliff Nuclear Power Plant (CCNPP)

CALCULATION TITLE:

Transfer Cask Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P.

SUMMARY DESCRIPTION:

1) Calculation Summary

Calculation presents an evaluation of the loading cycles of transfer cask for the NUHOMS® 24P and NUHOMS® 32P systems to show that fatigue analysis exemption criteria associated with NC-3219.2 of ASME code are met for 60-year service life of transfer cask.

2) Storage Media Description:

NA

If original issue, is licensing review per TIP 3.5 required?

Yes ☐ **No** ☒ **(explain below)** **Licensing Review No.:** _____

This calculation is prepared to support a Site Specific License Application by CCNPP that will be reviewed and approved by the NRC. Therefore, a 10CFR72.48 licensing review per TIP 3.5 is not applicable.

Software Utilized (subject to test requirements of TIP 3.3):	**Version:**
NONE	N/A

Calculation is complete: Originator Name and Signature: Jerzy Mackiewicz	03/30/2010 Date:
Calculation has been checked for consistency, completeness and correctness: Rahed Haroon for Gurmit Dhaliwal Checker Name and Signature: Dhaliwal Gurmit	3/30/2010 Date:
Calculation is approved for use: Project Engineer Name and Signature: Kamran Tavassoli	03/30/10 Date:

REVISION SUMMARY

REV.	DESCRIPTION OF CHANGES	AFFECTED PAGES	AFFECTED Computational I/O
0	Initial Issue	All	All

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1.0 Purpose

Calculation presents an evaluation of the loading cycles of transfer cask for the NUHOMS® 32P and NUHOMS® 24P systems to show that fatigue analysis exemption criteria associated with NC-3219.2 of ASME code are met for 60-year service life of the transfer cask.

2.0 References

1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, 1992 Edition.
2. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix I, 1992 Edition.
3. ASME Boiler and Pressure Vessel Code, Section II, Part D, 1992 Edition.
4. TN Calculation No. 1095-35, Rev. 2, "NUHOMS 32P Transfer Cask Structural Analysis."
5. TN Document NUH002.0103, Rev. 1A, Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, NRC Docket No. M-39.
6. TN Calculation No. 1095-6, Rev. 1, NUHOMS-32P, Transfer Thermal Analysis 103 °F Ambient.
7. TN Calculation No. 1095-16, Rev. 0, NUHOMS-32P, Transfer Thermal Analysis, -3 °F Ambient.
8. BGE Drawing BGE-01-3002, Rev. 1, "ISFSI Onsite Transfer Cask Inner & Outer Shell Assy."
9. BGE Drawing BGE-01-3003, Rev. 2, "ISFSI Onsite Transfer Cask Inner Main Assembly."
10. BGE Drawing BGE-01-3001, Rev. 0; "ISFSI Onsite Transfer Structural Shell Assy."

3.0 Assumptions

1. Fatigue effects evaluation is based on the assumed 600 use cycles of transfer cask in 60-year transfer cask service time.
2. Material properties evaluation is done for the temperature 400 °F. This temperature bounds transfer cask operational temperature [4] and secures conservatism of criteria estimation.
3. Limiting, lower number of loading cycles of the transfer cask can be obtained by assuming that transfer cask loads have structural and thermal characteristics of NUHOMS 32P systems. Thermal gradients for 32P systems are more severe than thermal gradients for 24P systems, and 32P DSC designs are heavier than 24P designs.

4.0 Materials

The mechanical properties of transfer cask materials are evaluated at temperature 400 °F. This temperature bounds transfer cask operational temperature and was chosen to secure conservatism of criteria estimation Mechanical properties of steel components of transfer cask at 400 °F are summarized in Table 1, below:

Table 1 TC Material Specifications [8], [9], [10]

Components	Material Specification
Structural Shell	SA-516 Type 70
Inner Shell	SA-240 Type 304
Top Cover Plate	SA-240 Type 304
Top Flange	SA-182 Type F304N
Bottom Support Ring	SA-182 Type F304N
Bottom Cover Plate	SA-240 Type 304
RAM Access Penetration Ring	SA-182 Type F304N
Bottom End Plate	SA-240 Type 304
Trunnions	SA-564 Type 630 PH
Trunnion Sleeves	SA-182 Type F304N

Table 2 TC Material Properties at 400 °F

Material Specification		S_m [psi]	S_u [psi]	E [psi]	α *(instantaneous)* [in/in/°F]
SA-240 Type 304	(18cr-8ni)	18700	64400	26.5×10^6	9.8×10^{-6}
SA-182 Type F304N	(18cr-8ni-n)	20300	73200	26.5×10^6	9.8×10^{-6}
SA-516 Type 70	(C-Mn-Si)	21700	70000	27.7×10^6	7.7×10^{-6}
SA-564 Type 630 PH	(17Cr-4Ni-4Cu)	43800	131400	26.5×10^6	5.9×10^{-6}

5.0 Criteria and Methodology

Criteria are based on subsection NC-3219.2 of ASME code [1] - Rules to Determine Need for Fatigue Analysis of Integral Parts of Vessels. Fatigue analysis need not be made provided all of criteria contained in Condition A or all criteria contained in condition B of NC-3219.2 are met. If neither Condition A nor Condition B is met, a detailed fatigue analysis is required in accordance with the rules of Appendices XIII and XIV for those parts, which do not satisfy the conditions.

The calculation is addressing criteria contained in condition B, given in Section NC-3219.2 of ASME code, to assess the need of fatigue analysis for cask loading operation cycles.

Fatigue evaluation is based on structural and thermal loads characteristics of 32P systems that bound loads associated with 24P systems. Thermal gradients for 32P systems are more severe than thermal gradients for 24P systems, and 32P DSC designs are heavier than 24P designs.

6.0 Calculations and Results

The evaluation below addresses the six criteria (a) through (f) contained in Condition B. A summary of the six criteria and their application to the transfer cask are presented in the paragraphs that follow:

(a) *The first criterion states that the transfer cask is adequate for fatigue effects provided that the expected total number of atmospheric-to-operating pressure cycles during normal operation does not exceed the number of cycles on the applicable fatigue curve of Appendix XIV corresponding to an S_a value of three times the S_m value in Tables 2A, 2B and 4, Section II, Part D, Subpart 1 for the material at operating temperature*

The transfer cask is not a pressure-retaining boundary; therefore, the first criterion is not applicable.

(b) *The second criterion states that transfer cask is adequate for fatigue effects provided that the specified full range of pressure fluctuations during normal operation does not exceed the quantity*

$$(1/3 \times \text{design pressure} \times S_a/S_m), \text{ where}$$

S_a - is the value obtained from the applicable fatigue curve from Appendix XIV for the total specified number of significant pressure fluctuations, and

S_m - is the allowable stress intensity for the material value in Tables 2A, 2B and 4, Section II, Part D, Subpart 1 for the material at operating temperature.

Significant pressure fluctuations are those for which the total excursion exceeds the quantity:

(1/3) x design pressure x S/S_m, where

S equals the value of S_a for 10^6 cycles in case of total number of service cycles is 10^6 or less.

Since the transfer cask is not pressure retaining boundary the second criterion is not applicable.

(c) *The third criterion states that the transfer cask is adequate for fatigue effects provided that the temperature differences between any two adjacent points on the transfer cask during normal operation do not exceed the magnitude:*

$$S_a / (2 \times E \times \alpha), \text{ where}$$

S_a - is the value obtained from the applicable fatigue curve for the specified number of startup-shutdown cycles,
α - is the instantaneous coefficient of thermal expansion at the mean value of the temperatures at the two points (given by Tables TE, Section II, Part D, Subpart 2, and
E - is the modulus of elasticity taken from Tables TM, Section II, part D, Subpart 2 at the mean value of the temperatures at the two points.

The temperature difference is a maximum for a point on the inner surface of the cask and the corresponding point on the outer surface of the cask, through the cask wall thickness.

It is conservatively postulated that the transfer cask will be used approximately 600 times during its designed life to transfer DSCs to and from the HSM, The conservative value of S_a associated with this number of cycles is 90 ksi (Figure I-9.1 [2], S_u < 80 ksi).

Therefore, using conservatively data of Table 2, and taking $E=26.5\times10^6$ psi and $\alpha = 9.8\times10^{-6}$, the lowest bound of the quantity $S_a/2E\alpha$ is $90000 / (2\times26.5\times10^6\times9.8\times10^{-6}) = 173.3$ °F.

The maximum temperature difference between any two adjacent points on the transfer cask, for NUHOMS 32P systems, can be conservatively estimated as 32 °F, based on temperature distribution plot from Ref. [7], page 18 (Figure 2). Such conclusion is based on the observation that no area on the temperature distribution plot indicates the local temperature change bigger than contour plot temperature increment 32 °F.

Therefore, the third criterion is satisfied for the transfer cask.

(d) *The fourth criterion states that the transfer cask is adequate for fatigue effects provided that the temperature difference between any two adjacent points on the transfer cask does not change during normal operation by more than the quantity*

$$S_a / (2 \times E \times \alpha), \text{ where}$$

S_a - *is the value obtained from the applicable fatigue curve for the total specified number of significant temperature-difference fluctuations.*

A temperature-difference fluctuation is considered to be significant if its total algebraic range exceeds the quantity;

$$S / (2 \times E \times \alpha), \text{ where}$$

S equals the value of S_a *for* 10^6 *cycles in case of total number of service cycles is* 10^6 *or less*

During a normal use cycle, the transfer cask is subjected to temperature fluctuations relative to the prevailing ambient temperature at the start of a use cycle. A normal transfer cask thermal use cycle for DSC loading and transfer to the HSM would results in a temperature fluctuation from the following operational conditions:

1. At the start of the use cycle, the cask will have a temperature equal to the prevailing normal ambient temperature.
2. A temperature fluctuation due to placement of the cask in the plant's fuel pool at the prevailing plant fuel pool temperature would result.
3. A temperature fluctuation due to removal of the cask from the plant's fuel pool with the decay heat power of the SFAs in water transferred to the cask would result.
4. A temperature fluctuation due to draining of the DSC/cask with the decay heat power of the SFAs in air transferred to the cask would result.
5. A temperature fluctuation due to drying of the DSC with the decay heat power of the SFAs in a vacuum transferred to the cask would result.
6. A temperature fluctuation due to backfilling of the DSC with the decay heat power of the SFAs in helium transferred to the cask would result,
7. Temperature fluctuations may result from transport of the transfer cask and DSC to the HSM with a solar heat flux and the decay heat power of the SFAs in helium transferred to the cask,
8. At the end of the cycle the cask is returned to a temperature equal to the prevailing ambient temperature.

The temperature fluctuations for a DSC unloading use cycle would be similar. This temperature fluctuation does not have however significant effect on the transfer cask thermal gradient fluctuations.

Taking the bounding value of S = 12500 psi for carbon steel with S_u value less than 80 ksi (Figure I-9.1 [2], S_u < 80 ksi)., and pair of E = 27.7 x 10^6 psi, and α= 7.7 x 10^{-6} in./in./°F, the bounding value of S/2Eα is 12500 / ($2\times27.7\times10^6\times7.7\times10^{-6}$) = 29.3 °F.

The most significant expected fluctuation in temperature gradient occurs during a change in ambient temperature from -3 °F to 103 °F. [6], [7]. The assessment of temperature distribution results, documented in Reference [6] and [7] (Figure 1 and Figure 2), reveal that the local change of temperature due to the change of ambient temperatures from -3 °F to 103 °F, for the transfer cask components, is below 16 °F temperature range bound.

The fourth criterion, therefore, is satisfied for the transfer cask.

(e) *The fifth criterion states that for components fabricated from materials of differing moduli of elasticity or coefficients of thermal expansion, the total algebraic range of temperature fluctuation experienced by the component during normal operation must not exceed the magnitude*

$$S_a / (2 \times (E_1 \times \alpha_1 - E_2 \times \alpha_2)), \text{ where}$$

S_a *- is the value obtained from the applicable fatigue curve for the total specified number of significant temperature fluctuations,*

E_1, *and* E_2 *- are the moduli of elasticity, and*

α_1, *and* α_2 *- are the values of the instantaneous coefficients of thermal expansion at the mean temperature value involved for the two materials of construction.*

A temperature fluctuation is considered to be significant if its total excursion exceeds the quantity $Sa/2(E_1\alpha_1 - E_2\alpha_2)$,), *where S is the value of* S_a *obtained from the applicable fatigue curve for* 10^6 *cycles. If the two materials used have different applicable design fatigue curves, the lower value of* S_a *shall be used.*

The critical area for the transfer cask is the interface between the trunnion sleeve (SA-182 Type F304N) and the trunnion (SA-564 Type 630PH). Assuming 600 use cycles, the value of S_a = 90 ksi. The maximum value of $Sa/2(E_1\alpha_1 - E_2\alpha_2)$, for materials presented in Table 2 is 90000 / ($2\times(26.5\times10^6\times9.8\times10^{-6} - 26.5\times10^6\times5.9\times10^{-6})$) = 435 °F. The maximum temperature range experienced by the transfer cask during normal operation is clearly below 435 °F [6], [7]. Therefore, this fifth criterion is satisfied for the transfer cask.

(f) *The sixth criterion states that the transfer cask is adequate for fatigue effects provided that the specified full range of mechanical loads do not result in stress ranges which exceed the* S_a *value obtained from the applicable fatigue curve for the total specified number of significant load fluctuations. If the total specified number of significant load fluctuations does not exceed* 10^6, *the S is value of* S_a *at* $N = 10^6$ *cycles.*

During a normal use cycle, the transfer cask is subjected to mechanical load fluctuations including handling and transport load fluctuations as follows:

1. At the start of the use cycle, the cask is lifted from the transport trailer to the plant's fuel building decon pit.
2. The cask and DSC filled with water is lifted from the decon pit and placed in the plant's fuel pool.
3. The cask and DSC filled with water, and loaded with SFAs is lifted from the fuel pool and placed in the decon pit.
4. The cask and dry DSC loaded with SFAs is lifted from the decon pit and is rotated to a horizontal position onto the support skid and transport trailer.
5. The cask with loaded DSC resting on the support skid is subjected to motion loads and braking loads during transport to the HSM along a predetermined route.
6. The cask is secured to the HSM and the DSC is transferred to the HSM with the hydraulic ram. At the end of this use cycle the cask is returned to its designated storage area or to begin the next use cycle.

The maximum total range of mechanical load fluctuations for this normal use cycle is 30.5 ksi ([4], pages 44,110). The mechanical loads fluctuations for DSC unloading use cycle would be similar. For the assumed 600 use cycles, and assumed conservative ratio of stress cycles per use cycle equal to 5, the value of S_a corresponding to 3000 stress cycles is about 56 ksi (Figure I-9.1 [2]). Such magnitude exceeds significantly maximum total stress range 30.5 ksi.

The sixth criterion, therefore, is satisfied for the transfer cask.

The evaluation presented in the preceding paragraphs demonstrates that the six criteria (a) through (f) contained in NC-3219.2 are satisfied for all components of the transfer cask.

7.0 Conclusions

ASME code criteria listed in Section NC-3219.2 were evaluated to determine whether the transfer cask service loads of NUHOMS® 24P and NUHOMS® 32P systems create potential risk of the design fatigue failure. The criteria evaluation, documented in the calculation, shows that transfer cask service loads of NUHOMS® 24P and NUHOMS® 32P systems do not create potential risk of the TC design fatigue failure, for 600 cycles in 60-year transfer cask life, and that detailed fatigue evaluation is not necessary.

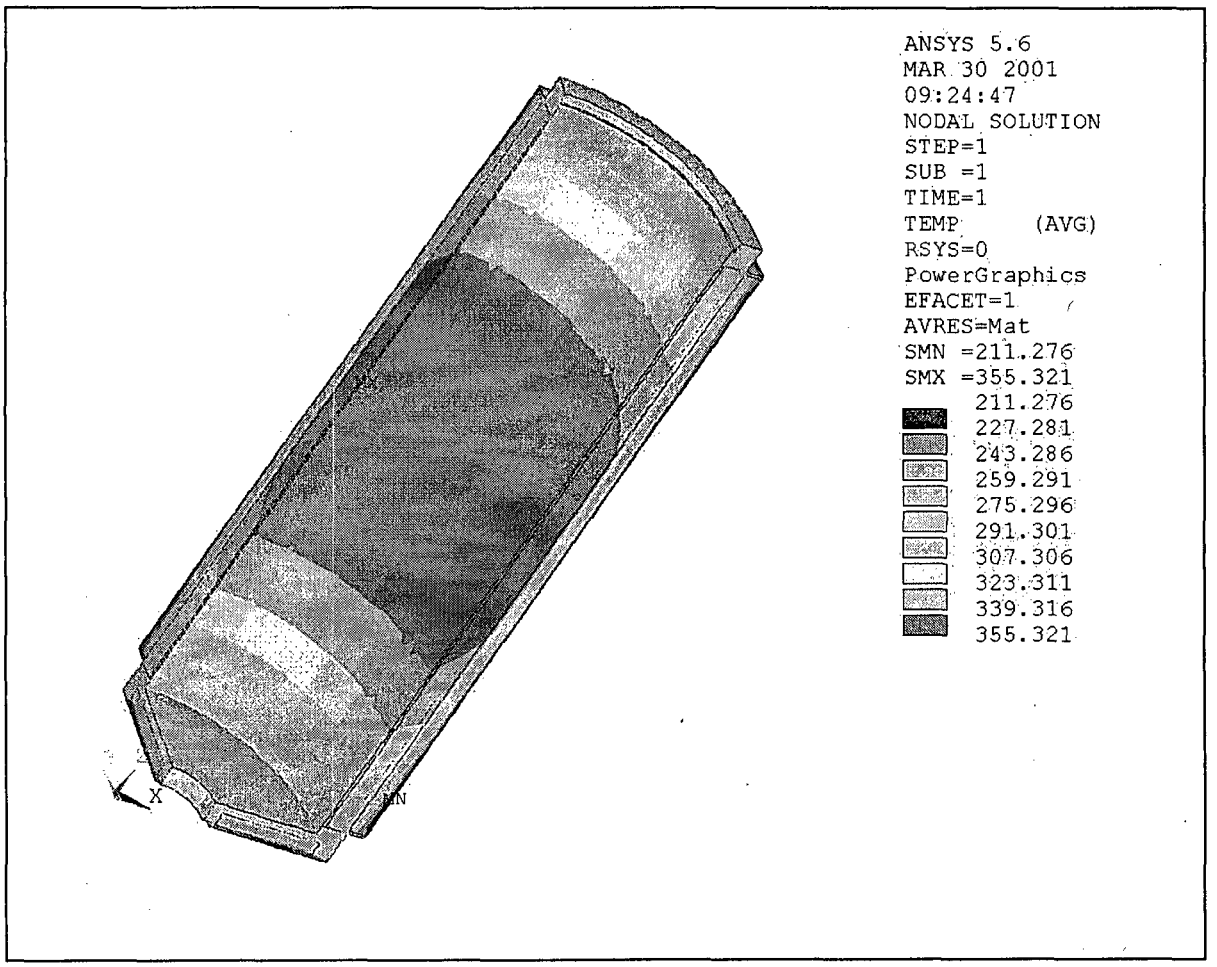

Figure 1 Cask Temperature Profile for Ambient Temperature 103 °F ([6], page 23)

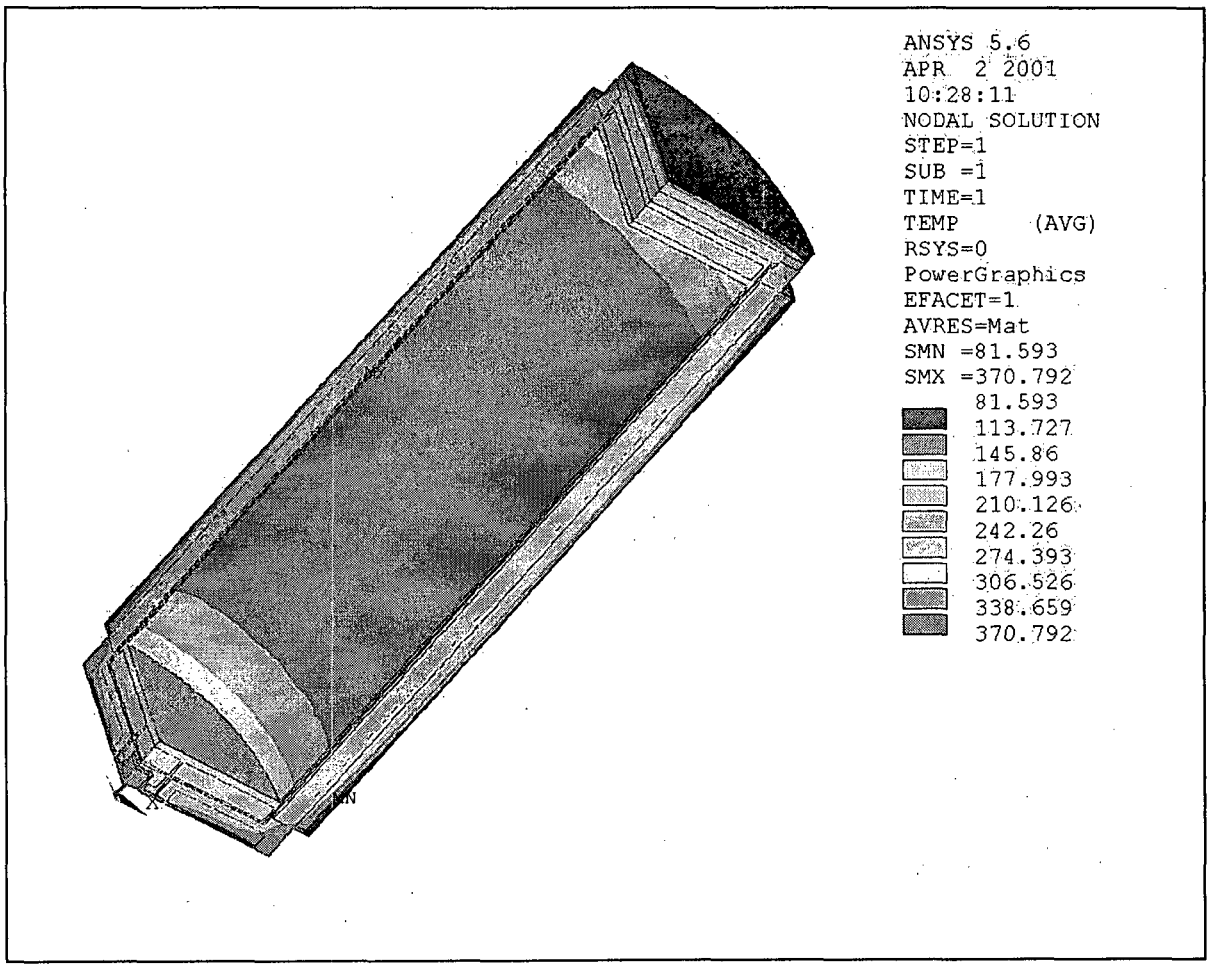

Figure 2 Cask Temperature Profile for Ambient Temperature -3 °F ([7], page 18)

Appendix C

ISFSI Updated Safety Analysis Report Supplement and Changes

C1.0 INTRODUCTION

This appendix provides a supplement and identifies pertinent changes to the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) Updated Safety Analysis Report (USAR). Section C2.0 of this appendix contains proposed changes to the existing ISFSI USAR. Section C3.0 of this appendix contains a proposed new section for the ISFSI USAR to be added under Chapter 9, Conduct of Operations. The new Section 9.6, Aging Management Programs, provides a summarized description of the activities for managing the effects of aging of ISFSI systems, structures, and components. This proposed new ISFSI USAR section will also present the evaluations of time-limited aging analyses (TLAAs) for the renewed license period.

C2.0 CHANGES TO EXISTING ISFSI USAR INFORMATION

The following is a listing of changes to the ISFSI USAR to support the renewal of Material License SNM-2505.

- Add the following text to the last paragraph is Section 1.1

  ", or for a maximum of 40 years under the renewed license period (60 years, total)"

The revised paragraph will read as follows (The text to be added is shown in bold underline font):

> "The fuel assemblies to be stored in the ISFSI are located in the Calvert Cliffs spent fuel pool and were irradiated only in the Calvert Cliffs reactors. Twenty-four fuel assemblies are stored in each NUHOMS-24P DSC, 32 fuel assemblies are stored in each NUHOMS -32P DSC, and one DSC is stored in each concrete module. The license allows construction and operation of a total of 120 modules. These modules will be built incrementally, as needed, to match BGE's requirements for additional storage. Operation of the facility will continue for up to 20 years under the initial license and continue under license renewal as necessary until a permanent facility is available for spent fuel disposal**, or for a maximum of 40 years under the renewed license period (60 years, total)**. As defined in Table 1.2-2 of Reference 1.2, the minimum design life of the facility is 50 years."

- Delete the following text in the second paragraph of Subsection 1.2.1

  "Despite Department of Energy's obligations under the Nuclear Waste Policy Act of 1982, as amended, to begin accepting fuel on January 31, 1998, BGE's current best estimate for the earliest date to ship spent fuel for permanent disposal is the year 2013."

  and

  "The provision for 120 HSMs will provide the minimum storage capacity needed to carry Calvert Cliffs to approximately 2026."

- Revise the only remaining sentence in the second paragraph in Subsection 1.2.1 to read

  "The initial license limits spent fuel storage capacity to 1,111.68 TeU of spent fuel assemblies."

The revised paragraph will read as follows:

> "The initial license limits spent fuel storage capacity to 1,111.68 TeU of spent fuel assemblies."

- The Chapter 9 LIST OF EFFECTIVE PAGES will be revised to list the new Section 9.6 pages and to note the page number changes for Section 9.6 now Section 9.7 and Section 9.7 now Section 9.8.

- The Chapter 9 TABLE OF CONTENTS will be revised to list the new Section 9.6 subsections and to revise Section 9.6 now Section 9.7 and Section 9.7 now Section 9.8.

- The following acronyms will be added to the Chapter 9 LIST OF ACRONYMS

  "CLB Current Licensing Basis"
  "LRA CCNPP License Renewal Application"
  "SER Safety Evaluation Report"
  "TLAAs Time-Limited Aging Analyses"

- A new Section 9.6 AGING MANAGEMENT PROGRAMS will be added to the Calvert Cliffs ISFSI USAR. See Section C3.0 of this appendix for the new text.

- Section 9.6 DECOMMISSIONING PLAN will be renumbered as Section 9.7 and the page numbers will be revised.

- Section 9.7 REFERENCES will be renumbered as Section 9.8 and the page numbers will be revised.

- The following references will be added to the Chapter 9 reference section. The section will be numbered as 9.8 (was Section 9.7 REFERENCES)

  9.14 AREVA Technical Report 10955-0101, Revision 1, May 21, 2010, ISFSI Time-Limited Aging Analysis Report

  9.15 U.S. Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, May 1988, Radiation Embrittlement of Reactor Vessel Materials

  9.16 NUH-002, Revision 2, March 1990, Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS®-24P, Submitted to the United States Nuclear Regulatory Commission by Pacific Nuclear Fuel Services, Inc., San Jose, California

  9.17 Letter from R. J. Lewis (NRC) to G. Vanderheyden (CCNPP), dated June 10, 2005, Amendment 6 to Material License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation, Safety Evaluation Report

  9.18 American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel Code, Section III, Division I, Subsection NB, July 1, 2009, Class 1 Components, Rules for Construction of Nuclear Facility Components

C3.0 NEW ISFSI USAR SECTION

The following text will be integrated into the ISFSI USAR Chapter 9 to document aging management programs credited in the ISFSI license renewal review, and TLAAs evaluated to demonstrate acceptability during the period of extended operation. The following text will be numbered sequentially within the new ISFSI USAR Section 9.6, Aging Management Programs.

9.6 AGING MANAGEMENT PROGRAMS

An aging management assessment of the HSMs, DSCs [Nutech Horizontal Modular Storage] (NUHOMS-24P and NUHOMS-32P), transfer cask, transfer cask lifting yoke, and the cask support platform was performed. The inspection and monitoring activities identified existing activities necessary to provide

reasonable assurance that ISFSI and transfer cask components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the renewal period. This section describes these aging management programs.

9.6.1 HSM AGING MANAGEMENT PROGRAM

The HSM Aging Management Program credits the Calvert Cliffs Part 50 programs credited for managing the effects of aging in the Calvert Cliffs Nuclear Power Plant as described in Chapter 16 of Reference 9-1. The HSM programs involve monitoring the exterior surfaces of the HSMs, including visual inspection of the accessible concrete; any exposed steel subcomponents, embedments, and attachments; and the lightning protection system. Interior inspections are conducted upon loading of a cask. Exterior inspections are conducted annually.

The HMS monitored conditions include, but not limited, to the following:

- Concrete - spalling, cracking, delaminations, honey combs, leaching, discoloration, loss of material, or any other property that would be noted by visual inspection
- Structural Steel - corrosion, peeling paint, deflection, lost or missing anchors /fasteners, missing or degraded grout under base plates, twisted beams, cracked welds
- Equipment Foundations - settlement, cracked concrete
- Equipment Supports - cracked concrete, loose connections, corroded steel
- Roof Systems - structural integrity, deteriorated penetrations (i.e., drains, vents, etc.), signs of water infiltration, cracks, ponding, and flashing degradation
- Seismic Gaps - gaps or loss of joint filler material
- Lightning Protection System (above grade) - corrosion

9.6.2 TRANSFER CASK AGING MANAGEMENT PROGRAM

The transfer cask aging management program credits periodic inspections performed on the transfer cask. The procedure includes visual inspections of the carbon steel subcomponents. Monitored conditions include corrosion.

9.6.3 TRANSFER CASK LIFTING YOKE AGING MANAGEMENT PROGRAM

The transfer cask lifting yoke aging management program credits the transfer cask lifting yoke annual inspection. This procedure includes visual inspections of the transfer cask lifting yoke carbon steel subcomponents. Monitored conditions include corrosion.

9.6.4 CASK SUPPORT PLATFORM AGING MANAGEMENT PROGRAM

The cask support platform aging management program credits the Calvert Cliffs power plant Chemistry Control Program, as described in Chapter 16 of Reference 9-1. Loss of material and cracking are prevented through control of specified limits on chloride in the spent fuel pool water.

9.6.5 TIME-LIMITED AGING ANALYSIS

This section discusses the results for each of the TLAAs evaluated for license renewal. The evaluations have demonstrated that the analyses have been projected to the end of the renewed license period.

9.6.5.1 DSC Time-Limited Aging Analysis

AREVA Technical Report 10955-0101 (Reference 9.14) was prepared to identify and evaluate the effect of TLAA, to demonstrate safe operation over the extended service life of the ISFSI. The report evaluated the DSCs stored in the HSMs. This section describes the finding of the TLAA.

DSC Materials:

Stainless Steel

The NUHOMS-24P and NUHOMS-32P DSC construction uses Stainless Steel 304 material and compatible weld metal. Since the DSC is filled with the inert helium gas there is no significant corrosion of the DSC shell and other components. Neutron fluence can affect mechanical properties of steels. However, studies on fast neutron damage in stainless steel and low alloy steels rarely evaluate damage at fluence levels below 10^{17} neutrons/cm^2 because they are not significant (Reference 9.15). For the DSC, the neutron fluence (10^{14} neutrons/cm^2) is much less than this level for the intended storage period and hence, a TLAA is not required.

The DSC and weld stresses due to temperature and pressure inside the DSC are an important aspect of the design. Per Reference 9.14, the NUHOMS-24P DSC normal operating design temperature and pressure are 400°F and 10 psig and the accident temperature and pressure are 460°F and 50 psig.

Per Reference 9.14, the NUHOMS-32P DSC normal operating design temperature is 460°F for the DSC shell and 380°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 30 psig. The design temperatures for the accident conditions for the NUHOMS-32P DSC shell is 575°F and 475°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 100 psig. The accident pressure values are established using 100% cladding breach.

The design temperatures are calculated at the beginning of NUHOMS storage. This calculation is bounding since the DSC temperatures are shown to monotonically decrease as a function of time.

Therefore, the heating effect (and hence, the internal pressure effect) on the DSC for the future 40 years of service will be much less severe than that during the past 20 years of service. Hence, the stresses in the DSC components will be acceptable for the extension period.

The service life for the NUHOMS-24P DSC system is documented in Table 1.2-2, Reference 9.16 as 50 years. Sufficient clearances are provided in both the radial and axial direction between the DSC internal components to permit free thermal expansion for NUHOMS-24P and NUHOMS-32P DSCs. This design feature acts to minimize the thermal cycling and fatigue on the DSC. There will be more room for free thermal expansion as the decay heat from the fuel decreases causing the DSC internal component temperatures to decrease as the storage time is increased from 50 years to 60 years. Therefore, thermal cycling and fatigue on the DSC will not be impacted when the storage period is increased from 50 years to 60 years.

Lead in the Shield Plugs

The DSC uses lead in the shield plugs. The lead shielding function is not affected by the radiation level in the DSC.

Poison Plates (NUHOMS-32P DSC Only)

Since the NUHOMS-32P DSC uses fixed neutron absorbers in the DSC basket, calculations were performed to assure criticality safety. Time dependency for criticality calculations may result due to depletion of boron in the poison plates utilized in the NUHOMS-32P DSC baskets.

Reference 9.17, Section 6.3, states that "The continued efficacy of the neutron absorber materials over a 20-year storage period is assured by the design of the NUHOMS-32P canister which ensures that the neutron absorbers will remain in place during accident conditions. Additionally, the neutron flux from the irradiated fuel will result in negligible depletion of the Boron 10 content in the neutron absorber materials over the life of the storage system."

Per Reference 9.14, the total neutron activity in the NUHOMS-32P DSC is 4.175×10^8 n/s/assembly. To estimate the total flux a conservative final assembly surface area of 25,000 cm^2 is considered in Reference 9.14. The total scalar flux is estimated to be 8.65×10^4 n/cm^2-s. Using the thermal cross section for Boron 10, (3,837 barn), the fraction of the original Boron 10 depleted after 60 years is 2.3E-6, which is negligible. The actual neutron flux is mostly fast and epithermal, and will be declining with time, so the actual depletion during dry storage will be less than the depletion calculated in Reference 9.14. Therefore, continued efficacy of the neutron poison is assured for an additional 40 years of fuel storage.

DSC Fatigue Evaluation

The fatigue evaluation of the NUHOMS-24P DSC is documented in Appendix C.4.1 of Reference 9.16 for a 50 year operational life of the ISFSI. The fatigue effects on the DSC were addressed using the criteria contained in Section III NB-3222.4 of the ASME Code (Reference 9.18, The 1983 Edition). The analysis evaluated the DSC under six criteria and concluded that the DSC and other components satisfy these criteria and that no consideration of fatigue is required for a service life of 50 years.

In order to extend the operational life by another 10 years, it is necessary to re-evaluate the DSC against these six criteria using an approach that is consistent with that utilized in Reference 9.16. A fatigue analysis for the NUHOMS-24P and NUHOMS-32P DSCs was performed for a 60 year service life. This analysis uses the six criteria in NB-3222.4(d) of ASME Code (Reference 9.18, the 1998 Edition including the 1999 Addenda) and determines that the DSC service loads of the NUHOMS-24P DSC and NUHOMS-32P DSC systems do not create any potential risk of the DSC design fatigue failure and that no detailed fatigue evaluation is necessary.

9.6.5.2 HSM Time-Limited Aging Analysis

Reference 9.14 was prepared to identify and evaluate the effect of TLAA, to demonstrate safe operation over the extended service life of the ISFSI. The report evaluated the HSMs. This section describes the finding of the TLAA.

HSM Concrete

The HSM is a reinforced concrete structure. The effect of radiation on the HSM concrete is evaluated in Section 8.1.1.5.D and Section 12.8.1.1.5.D. These evaluations demonstrate that the magnitude of the neutron fluence incident on the concrete is low enough to not affect the properties of the concrete. These evaluations also demonstrate that the magnitude of the gamma-ray energy deposition on the HSM concrete is not sufficient to cause any radiation heating in the concrete of the HSM. Therefore, the thermal analyses documented in Sections 8.1.1.5D and 12.8.1.1.5D implicitly considered the radiation heat effects adequately for the HSM concrete.

NUHOMS-24P DSC

The maximum predicted temperature of concrete at the beginning of storage was estimated to be below 150°F in Reference 9.14 using a bounding decay heat at the beginning of storage life.

NUHOMS-32P DSC

The maximum temperature of the concrete at beginning of storage with a NUHOMS-32P DSC is 157°F per Reference 9.14.

The maximum concrete temperatures for the additional 40 years of service (as shown in Figure 8.1-27 of Reference 9.16) will be lower because the decay heat reduces monotonically as a function of time. Hence, the heating effect on the HSM concrete, for an additional 40 years of service, will be much less severe than the past 20 years of service.

The environmental degradation of reinforced concrete will not be significant, as proper concrete cover has been provided to the reinforcing bars made of carbon steel.

Reference 9.16, Section 8.2.10.6 documents the analysis of thermal cycling of the HSM based on the 50 year storage life. The number of cycles will increase from 18,250 to 21,900 when the design life is extended from 50 years of storage to 60 years of storage. These are still significantly below the limit of 10,000,000 (See Section 8.2.10.6 of Reference 9.16). Therefore, thermal cycling will have negligible impact on the HSM reinforced concrete for an additional 40 years of service.

DSC Support Rail Steel in the HSMs

The DSC support structure inside the HSM is designed to support the DSC during normal, off-normal and accident conditions. Since the DSC support rails are fabricated from Nitronic 60 austenitic stainless steel, it is expected that there would be no corrosion of the rail material and is expected to maintain its function for the additional 40 years of service.

DSC Support Rail Lubricant in the HSMs

The HSM and transfer cask support rails are coated with a dry film lubricant Perma-Slik to minimize friction during insertion and retrieval of the DSC. The material specification of the lubricant indicates that it is suitable for very high and cryogenic temperature applications. The presence of a non-corrosive environment due to the absence of a formal sea breeze and relatively milder temperature fluctuations at ISFSI site ensure that the lubricant does not degrade with age. The effect of radiation on these lubricants is not specified, however, it is expected that it is minimal since these are inorganic and consist entirely of graphite, a moderating material. As stated above once the DSC is in place within the HSM, the lubricant performs no function during storage of the DSC.

NUHOMS-24P DSC

The coefficient of friction associated with these lubricants is below 0.05 while the design basis calculations employed a coefficient of friction of 0.25 (Section 8.1.1.1 D of Reference 9.16). The mechanical system to be used for DSC transfer is capable of exerting a force equal to the loaded weight of a DSC and this condition has been evaluated in Section 8.1.2.1 of Reference 9.16 for the NUHOMS-24P DSC. A coefficient of friction of 1.0 has been used (for these "jammed DSC" analyses) without relying on the solid film lubricant. The support structure is designed for this loading. Hence, no further analysis is required.

NUHOMS-32P DSC

The coefficient of friction associated with these lubricants is below 0.05 while the design basis calculations employed a coefficient of friction of 0.25 (Reference 9.14). The mechanical system to be used for DSC transfer is capable of exerting a force equal to the loaded weight of a DSC and this condition has been evaluated for the NUHOMS-32P DSC. A coefficient of friction of 1.0 has been used (for these "jammed DSC" analyses) without relying on the solid film lubricant. The support structure is designed for this loading. Hence, no further analysis is required.

9.6.5.3 Transfer Cask Fatigue Evaluation

The fatigue evaluation for the transfer cask is performed in accordance with ASME code criteria listed in Section NC-3219.2 to determine whether the transfer cask service loads of NUHOMS-24P DSC and NUHOMS-32P DSC systems create potential risk of the design fatigue failure. The criteria evaluation shows that transfer cask service loads of NUHOMS-24P and NUHOMS-32P DSC systems do not create potential risk of the transfer cask design fatigue failure, for 600 cycles in 60-year transfer cask life, and that detailed fatigue evaluation is not necessary.

Transfer Cask Trunnions Fatigue Evaluation

The fatigue evaluation of transfer cask trunnions shows that the transfer cask operations do not pose potential risk of fatigue failure of trunnion or trunnion sleeve throughout the planned 60-year service time.

NS-3 in Transfer Cask

The transfer cask contains 3 inches of NS-3 neutron shielding sandwiched between the cask outer shell material and neutron shield jacket. Per Reference 9.14, the gamma and neutron dose at 1 inch from the cask surface for the accident conditions is 135 mrem/hr and 1000 mrem/hr, respectively for the NUHOMS-24P DSC and 85 mrem/hr and 1433 mrem/hr, respectively for the NUHOMS-32P DSC. Also, the dose rates at 1 inch from the transfer cask surface for the accident conditions with the NS-3 at the side of the transfer cask replaced with air bounds the dose rates at the inner surface of NS-3 in the transfer cask during normal conditions and that the transfer cask is only subjected to this gamma exposure when a fuel-loaded DSC is in the transfer cask during loading and transfer operations which are short term durations. This results in a gamma dose of approximately 3.0×10^5 Rads over the service life of 60 years. This is based on an assumption that 1 Rad = 1 Rem and is considered reasonable for gamma radiation for hydrogenous materials. This is significantly below the exposure limit of 1.5×10^{10} Rads for the material as stated in Reference 9.14.

Per Reference 9.14, to estimate the neutron fluence, a neutron dose to flux factor of 1 mrem/hr = 100 n/cm^2-s is used. The dose to flux factor for neutrons is based on dose rate spectra results from various NUHOMS ISFSI evaluations. The integrated fluence is estimated to be approximately 3.16×10^{14} neutrons/cm^2 over the service life of 60 years for the NS-3 in the transfer cask. Reference 9.14 noted that the thermal neutron exposure limit 1.5×10^{19} neutrons/cm^2 for the NS-3 material. Therefore, it is concluded that there is no significant degradation to the NS-3 material for the additional 40 years of operations of the transfer cask.

The exposure to radiation sources for an additional 40 years of service is shown to have no significant impact on the shielding capability of the NS-3 in the transfer cask. No significant hydrogen loss in the NS-3 material is expected due to radiation exposure.

9.6.5.3 Time-Limited Aging Analysis of the Transfer Cask Lifting Yoke

Lifting Yoke Fatigue Evaluation

The fatigue evaluation of transfer cask trunnions and lifting yoke system shows that the transfer cask operations do not pose potential risk of fatigue failure of trunnion or trunnion sleeve throughout the planned 60-year service time.

In the case of the transfer cask lifting yoke assembly, the structural adequacy against fatigue failure is secured for up to 286 transfer cask loading/unloading operations in the planned 60-year service time.

Appendix D

TECHNICAL SPECIFICATIONS CHANGES

TECHNICAL SPECIFICATION CHANGES

The Nuclear Regulatory Commission preliminary guidance for renewal of site-specific 10 CFR Part 72 licenses (Reference D1) states that an application for an Independent Spent Fuel Storage Installation (ISFSI) license renewal will include any Technical Specification changes or additions that are necessary to manage the effects of aging during the license renewal period. Review of the information provided in this ISFSI license renewal application and in the ISFSI Technical Specifications has confirmed that no changes to the current ISFSI Technical Specifications are needed.

REFERENCE

D1 Letter from E. W. Brach (NRC) to W. R. Matthews (Virginia Electric Power Company), dated March 29, 2001, Preliminary NRC Staff Guidance for 10 CFR Part 72 License Renewal

Appendix E

Environmental Report Supplement

Calvert Cliffs Nuclear Power Plant, LLC
September 17, 2010

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E1.0 INTRODUCTION

Independent Spent Fuel Storage Installations (ISFSIs) for storing spent fuel and associated radioactive materials are licensed by the U.S. Nuclear Regulatory Commission (NRC) in accordance with the Atomic Energy Act of 1954 (42 United States Code [USC] 2011, et seq.) and NRC implementing regulations (10 Code of Federal Regulations [CFR] Part 72). Nuclear Regulatory Commission issued Calvert Cliffs Nuclear Power Plant (Calvert Cliffs) a 20-year site-specific ISFSI operating license in 1992 for the temporary storage of spent fuel until a federal repository was available for permanent storage.

As a result of various events, a federal repository remains unavailable at the time of this application. In the Nuclear Waste Policy Act of 1982, Congress directed the U.S. Department of Energy (DOE) to construct and operate a geologic repository for the permanent disposal of commercial spent nuclear fuel. Under the Nuclear Waste Policy Act, DOE was to begin accepting spent nuclear fuel for disposal on or before January 1998. In the Amendments of 1987, Congress directed DOE to study Yucca Mountain located in Nevada as a potential location for a federal repository for spent nuclear fuel. The studies were delayed primarily by the State of Nevada by withholding the issuance of the requisite environmental permits, until 1994 when extensive subsurface studies were initiated. In 2002, based on the results of the DOE comprehensive studies, Congress and the President approved Yucca Mountain as the site for the federal repository. In June of 2008, DOE submitted a license application to the NRC that proposed initial loading of the facility may begin as early as the 2020 timeframe. In 2009, the President announced plans to terminate the Yucca Mountain Program, and in January 2010, the Secretary of Energy announced the formation of a blue ribbon commission on America's Nuclear Future to study alternatives for developing a safe, long-term solution to managing the nation's used nuclear fuel and high-level waste from defense programs. The final report from the blue ribbon commission is to be issued within 24 months or early 2012. Based upon this experience, there is significant uncertainty regarding when DOE will be able to take possession of the fuel.

Given the delays in developing a permanent repository, commercial nuclear power reactor owners and operators, including Calvert Cliffs Nuclear Power Plant, LLC – the owner and operator of Calvert Cliffs, have utilized their existing spent fuel pools to the maximum extent possible and provided separate dry storage facilities for additional space in order to maintain safe storage of the spent fuel until its acceptance by DOE for transportation offsite.

E1.1 PROPOSED ACTION

The proposed action is the renewal of the operating license for the ISFSI. The current operating license (SNM-2505) expires on November 30, 2012. By this application, Calvert Cliffs requests NRC to renew the operating license for an additional 40 years (i.e., until November 30, 2052). Pursuant to 10 CFR 72.7, Calvert Cliffs also requests exemption from the requirement for a 20-year renewal term, and proposes a license renewal period of 40 years to continue to store spent fuel until November 30, 2052. Based upon the experience with the proposed Yucca Mountain repository reviewed above, the time required to select, develop, and license facilities necessary to safely disposition spent nuclear fuel may require over 20 years. A 40-year term would provide storage capability while DOE develops and implements a viable long-term spent fuel disposition strategy.

E1.2 PURPOSE AND NEED

The purpose and need for the proposed action (renewal of an operating license) is to provide for continued temporary dry storage of spent nuclear fuel assemblies generated from operation of Calvert Cliffs until facilities are available for permanent disposition of spent fuel from Calvert Cliffs.

E1.3 ENVIRONMENTAL REPORT SUPPLEMENT SCOPE AND METHODOLOGY

This supplemental report was prepared by Calvert Cliffs as part of its application to the NRC for license renewal as provided by the following NRC regulations:

- 10 CFR Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste, 72.42, Duration of License; Renewal, and 72.34, Environmental Report
- 10 CFR Part 51, Environmental Protection Requirements for Domestic Licensing and Regulatory Functions, 51.60, Environmental Report-Materials Licenses

Title 10 CFR 72.42 lists the requirements that a licensee must meet for renewal of an ISFSI license, which includes the requirement for an Environmental Report Supplement to be submitted as part of the application. The NRC regulation 10 CFR 51.60 requires that an applicant for renewal of a license for storage of spent fuel in an ISFSI submit with its application a supplemental Environmental Report. In determining what information to include in the ISFSI Environmental Report supplement, Calvert Cliffs has relied on guidance provided in Reference E9.1. NRC guidance indicates that license renewal is not an exercise in re-licensing and is not intended to impose requirements beyond those that were met by the facility when it was initially licensed by the NRC. In addition, NRC indicates that the supplemental report may be limited to incorporating by reference, updating or supplementing the information previously submitted to reflect any significant environmental change. Therefore, Calvert Cliffs assembled a team to review the Environmental Report submitted with the original ISFSI license application and NRC's subsequent environmental analysis, to identify areas that require updating to meet the expectations of Reference E9.1 and to evaluate whether any significant changes have occurred during the initial licensing period and whether any significant changes are anticipated during the renewal term. To perform this review, Calvert Cliffs utilized staff that are knowledgeable of the site and ISFSI operations, who were involved with license renewal of the Units 1 and 2, and who are involved with current licensing activities for Calvert Cliffs Unit 3 (see Section E10.0 for a list of preparers).

As a result of this review, Calvert Cliffs determined that information related to the topics listed in Table E1.3-1 has not changed and the information and analyses provided in the original ISFSI Environmental Report remains valid, and therefore, will not be addressed in this supplemental report. For each topic, Table E1.3-1 provides the specific reference to the discussion in the original ISFSI Environmental Report.

To develop the Environmental Report supplement, Calvert Cliffs references material in the original ISFSI Environmental Report that remains valid throughout this document. In addition, Calvert Cliffs relies on information developed in support of Calvert Cliffs Unit 3, and references the Calvert Cliffs Unit 3 Environmental Report (Reference E9.2), as well as the NRC Draft Environmental Impact Statement (Reference E9.3) for detailed characterizations of the surrounding environment.

Table E1.3-1, Environmental Report Supplement Content Incorporated By Reference

Topic	Environmental Report Section	Remarks
Geology and Soils	2.5	Original information and analysis remains valid.
Water Resources	2.4, 5.4, 5.6	Original information and analysis remains valid.
Meteorology and Climatology	2.3, 5.6.2	Original information and analysis remains valid.
Ecology	2.2, 5.6.3	Original information and analysis remains valid, and threatened and endangered species are addressed in Sections E3.3 and E4.2 of this report.

E1.4 REGULATORY COMPLIANCE

The ISFSI facility does not require any additional permits, licenses, or approvals to operate other than the renewal of the NRC operating license.

Table E1.4-1 lists the environmental authorizations and consultations that would be required prior to the NRC renewal of the ISFSI operating license. The remainder of this section discusses each of these in more detail.

Table E1.4-1, Environmental Authorizations for Calvert Cliffs ISFSI License Renewal[a]

Agency	Authority	Requirement	Remarks
NRC	Atomic Energy Act (42 USC 2011 et seq.)	License renewal	Environmental report supplement submitted in support of license renewal application.
Fish and Wildlife Service	Endangered Species Act Section 7 (16 USC 1536)	Consultation	Requires Federal agency issuing a license to consult with Fish and Wildlife Service. (see Enclosures 1 and 2)
Maryland Historic Trust	National Historic Preservation Act Section 106 (16 USC 470f)	Consultation	Requires Federal agency issuing a license to consider cultural impacts and consult with State Historic Preservation Officer. (see Enclosure 3)
Maryland Department of Environment Coastal Zone Consistency	Federal Coastal Zone Management Act (16 USC 1451 et seq.)	Certification	Requires an applicant to provide certification to the Federal agency issuing the license that license renewal would be consistent with the federally approved state coastal zone management program. Based on its review of the proposed activity, the State must concur with or object to the applicant's certification. (see Enclosure 4)

[a] No renewal-related requirements identified for local or other agencies.

E1.4.1 THREATENED AND ENDANGERED SPECIES CONSULTATION

Section 7 of the Endangered Species Act (16 USC 1531 et seq.) requires Federal agencies to ensure that agency action is not likely to jeopardize any species that is listed or threatened. Depending on the action involved, the Act requires consultation with the Fish and Wildlife Service regarding effects on non-marine species, or the National Marine Fisheries Service, for marine species, or both. Fish and Wildlife Service and National Marine Fisheries Service have issued joint procedural regulations in 50 CFR Part 402, Subpart B, which address consultation, and Fish and Wildlife Service maintains the joint list of threatened and endangered species at 50 CFR Part 17. Given continued operations of the ISFSI does not impact the Chesapeake Bay, consultation with the National Marine Fisheries Service is not warranted.

As discussed in Section E3.3, several Federal threatened and endangered species and state species of special concern are found on the Calvert Cliffs site. In preparation for the Federal consultation process, Calvert Cliffs invited comment from the Fish and Wildlife Service and the Maryland Heritage Program regarding potential effects that the ISFSI license renewal might have on threatened and endangered species. Enclosures 1 and 2 include copies of Calvert Cliffs correspondence with these agencies.

E1.4.2 HISTORIC PRESERVATION CONSULTATION

Section 106 of the National Historic Preservation Act (16 USC 470 et seq.) requires Federal agencies having the authority to license any undertaking, prior to issuing the license, to take into account the effect of the undertaking on historic properties and to afford the Advisory Committee on Historic Preservation an opportunity to comment on the undertaking. Committee regulations provide for establishing an agreement with any State Historic Preservation Officer to substitute state review for Council review (36 CFR 800.2). Although not required by Federal law or NRC regulation, Calvert Cliffs has chosen to invite comment by the Maryland State Historic Preservation Officer. Enclosure 3 includes copies of Calvert Cliffs correspondence with the State Historic Preservation Officer.

E1.4.3 COASTAL ZONE MANAGEMENT PROGRAM COMPLIANCE

The Federal Coastal Zone Management Act (16 USC 1451 et seq.) imposes requirements on applicants for a Federal license to conduct an activity that could affect a state's coastal zone. The Act requires the applicant to certify to the licensing agency that the proposed activity would be consistent with the state's federally approved coastal zone management plan [16 USC 1456(c)(3)(A)]. The National Oceanic and Atmospheric Administration has promulgated implementing regulations that indicate that the requirement is applicable to renewal of Federal licenses for activities not previously reviewed by the state [15 CFR 930.51(b)(1)]. The regulations require that the license applicant provide its certification to the Federal licensing agency and a copy to the applicable state agency [15 CFR 930.57(a)].

The Maryland's Coastal Program was established through an Executive Order and approved by the National Oceanic and Atmospheric Administration in 1978. The Coastal Program is a networked program and is comprised of several state planning and regulatory programs. The Maryland Department of Natural Resources is the lead agency. Maryland's coastal boundary follows the inland boundary of the counties (and Baltimore City) bordering the Atlantic Ocean, Chesapeake Bay, and the Potomac River (Reference E9.4). The ISFSI is located within the Maryland Coastal Zone, and Enclosure 4 contains a copy of the Calvert Cliffs coastal zone management program certification for the ISFSI license renewal. Calvert Cliffs' submitted project descriptive material and a certification to the State. Concurrent with submitting the Environmental Report supplement to the NRC, Calvert Cliffs will submit a copy of this report to the State of Maryland in fulfillment of the regulatory requirement.

E2.0 ALTERNATIVES

The original ISFSI Environmental Report (Reference E9.5) analyzed six alternatives. In addition to the no-action alternative, the following alternatives were evaluated:

- Shipment to another nuclear reactor site
- Shipment to monitored retrievable storage or repository
- Construction of a new independent storage pool
- Other dry storage technologies
- Fuel consolidation
- Shipment to a reprocessing facility

Since the time the original ISFSI Environmental Report was prepared, no progress has been made in the industry to develop options for monitored retrievable storage or a spent fuel reprocessing facility. Without these options or a permanent federal repository, other utilities are also facing challenges for spent fuel storage and do not provide the storage capacity to received spent fuel from other utilities. Therefore, these three alternatives are no longer viable alternatives. As discussed in the original ISFSI Environmental Report, fuel consolidation remains a non-viable alternative.

E2.1 NO-ACTION ALTERNATIVE

The no-action alternative would result in the expiration of the ISFSI operating license as of November 30, 2012. Spent fuel currently stored at the facility would need to be removed. However, there is no federal repository or other legitimate disposition pathway available for this waste. Therefore, the no-action alternative would result in the lack of an appropriate storage or disposal pathway for the spent fuel, and therefore, is not a viable alternative.

E2.2 PROPOSED ACTION

The proposed action is renewal of the ISFSI operating license to authorize operation for an additional 40 years. The ISFSI is located on the Calvert Cliffs Nuclear Power Plant site near Lusby, Maryland. Calvert Cliffs owns and operates two 2,737 megawatt-thermal nuclear generating units on the site. Spent fuel generated from these two units is initially stored in the spent fuel pool, and then transferred to the ISFSI for dry storage.

The ISFSI is located outside the Calvert Cliffs protected area, but within the owner controlled area approximately 2300 feet southwest of the plant, and occupies approximately 6 acres. The ISFSI is about 86 feet above the probable maximum flood elevation. The area around the ISFSI is sloped to direct surface drainage to collection ditches for channeling rain water away from the site. Local intense rainfall is not a problem since the resulting flood water would need to rise at least 18 inches above yard grade to block the horizontal storage module (HSM) air inlets. Adequate surface drainage exists at the ISFSI yard to assure that water will not collect to a depth of any concern.

The current license authorizes the construction and operation of 120 HSMs, which are built incrementally to meet storage requirements. The ISFSI currently consists of 72 HSMs. The facility was initially licensed to use Nutech Horizontal Modular Storage (NUHOMS)-24P dry storage system. The principal components are an HSM, comprised of concrete and structural steel, and a steel dry shielded canister (DSC) with an internal basket that holds the spent fuel. The exterior walls and roof of the HSM are three feet thick, and the interior walls are two feet thick. Each HSM contains one DSC and each DSC contains 24 fuel assemblies.

The HSM provides structural support for the DSC, protects the DSC against extreme natural hazards such as tornado missiles, and provides radiation shielding. The concrete walls form interconnected sub-units of modules which are six wide and two back-to-back to form a 2x6 array. The modules are designed to provide surface dose rates at or below the design criteria of 20 mrem/hr on the HSM side and 100 mrem/hr at the HSM door. The HSM dissipates decay heat from the spent fuel by a combination of radiation, conduction, and convection. Natural convection air flow enters at the bottom of the HSM, circulates around the DSC, and exits through the flow channels in the HSM roof slab. A thermal radiation shield is used to reduce the HSM roof temperature to within acceptable limits for all conditions (Reference E9.6).

Since the initial license was issued, Calvert Cliffs received a license amendment that permits use of the Transnuclear NUHOMS-32P DSCs to optimize its dry spent fuel storage capacity (References E9.7 and E9.14). The NUHOMS-32P DSC system stores eight more spent fuel assemblies than the NUHOMS-24P DSC using the same external and internal shell dimensions. The NUHOMS-32P DSC storage capacity is optimized by reducing the space between the locations of each fuel assembly and by slightly reducing the size of the storage locations. The ISFSI currently consists of 48 HSMs for the NUHOMS-24P DSC and 24 HSMs for the NUHOMS-32P DSC. All 48 of the NUHOMS-24P HSMs are loaded. Only 15 of the NUHOMS-32P HSMs are loaded.

In June 2009, Calvert Cliffs submitted a license amendment request to allow increased burnup fuel to be loaded into the NUHOM-32P DSCs (Reference E9.8). The supply of fuel assemblies meeting the current license Technical Specifications was exhausted late in 2009. The current ISFSI Technical Specification burnup limit is 47 GWd/MTU, which is insufficient to support transfer of fuel currently in the Calvert Cliffs spent fuel pool and loading it into the remaining HSMs. No physical changes to the DSC or the HSM are required. However, Technical Specification changes are required to support fuel with higher burnup to be stored, consist of:

- expansion of the gamma and neutron source allowed for selected assemblies,
- increase burnup limit for fuel assemblies, and
- increase the allowable air temperature rise in the HSM.

The details of these proposed changes are provided in Calvert Cliffs license amendment request (Reference E9.8). Increasing the burnup limit to 52 GWd/MTU would sufficiently bound the range of discharge burnup values for fuel currently stored in the Calvert Cliffs spent fuel pool. Therefore, future transfers require approval of the request.

In addition to these primary components, the ISFSI also requires transfer equipment to move the DSCs from the spent fuel pool (where they are loaded with spent fuel) to the HSMs (where they are stored). This transfer system consists of a transfer cask, a hydraulic ram, a truck, a trailer, and a cask skid. This transfer system interfaces with the existing Calvert Cliffs spent fuel pool, the cask handling crane, and the site layout (i.e., roads and topography).

No electrical systems are required for the HSM or DSC during long-term storage, other than for lighting and security system power. Electrical power is used during DSC closure operations in the plant's Auxiliary Building and during DSC transfer operations to the HSM at the ISFSI. The required electrical power in the Auxiliary Building is obtained from the existing plant system. Power at the ISFSI is supplied from the site power distribution system which is powered from 11 and/or 21 13 kV plant busses (Reference E9.6).

Compressed helium or filtered plant air is used to force water from the DSC during closure operations. There are no steam systems required. Borated water is used to fill the DSC cavity prior to insertion into the spent fuel pool. Demineralized water is needed for filling the DSC/cask annulus and for washdown operations. This water is supplied by the existing Auxiliary Building demineralized water supply. No sewage treatment system is required for the ISFSI. No communication systems are required for the safe operation of the ISFSI. The existing plant page and telephone system has been extended to the ISFSI for convenience during transfer operations. Calvert Cliffs personnel currently perform the daily activities associated with the operation of the ISFSI facility including routine maintenance. No additional staff or offsite personnel will be required for routine operations and maintenance during the license renewal term. The passive nature of the facility requires little maintenance beyond periodic surveillance of the air inlet and outlet vents.

No gaseous, liquid, or solid wastes are generated at the ISFSI. The small volumes of these wastes, which are generated in the Auxiliary Building, has no significant impact on the ability of existing plant systems to process them and this condition is expected to remain unchanged in the license renewal term.

No radioactive wastes are generated during the storage life of the DSCs. Radioactive wastes generated during loading operations are treated using existing Calvert Cliffs facilities and procedures. No radioactive off-gas is produced at the ISFSI. Potentially contaminated air and helium purged from the DSC during evacuation are directed to the Auxiliary Building ventilation system. No liquid waste is produced at the ISFSI. Contaminated pool water removed from loaded DSCs is normally drained back into the spent fuel pool with no additional processing. A small amount (<15 ft^3/DSC) of liquid waste results from transfer cask decontamination. The decontamination procedure results in a small amount of a detergent/demineralized water mixture being collected in the cask washdown pit. Liquid wastes collected in the cask washdown pit are directed to the power plant liquid waste processing system.

No solid waste is produced at the ISFSI. A small quantity (<2 ft^3/DSC) of low-level solid waste is generated as a result of DSC loading operations and transfer cask decontamination (e.g., disposable Anti-C garments, tape, decon cloths, etc.). The solid waste generated is processed by compaction using the solid waste processing system.

No construction or refurbishment is currently planned during the ISFSI license renewal term. However, due to the delay in licensing of a federal repository for spent nuclear fuel, the need for long-term spent fuel storage onsite is significant and will require complete build out and loading of the facility (i.e., 120 HSMs). The ISFSI facility is subject to aging management activities to ensure the integrity of the HSMs during the ISFSI license renewal term. The aging management programs are summarized in Appendix A of this license renewal application.

A complete description of ISFSI is provided in the ISFSI Updated Safety Analysis Report (USAR) (Reference E9.6).

E2.3 OTHER ALTERNATIVES

No other alternatives than those analyzed in the original ISFSI Environmental Report have been identified. Of those originally evaluated, only construction of a new spent fuel pool and construction of a new ISFSI are potential options to the proposed action.

Construction of a new independent spent fuel pool, similar to the existing pool, would involve high commissioning costs and significant operating and maintenance cost, as compared to continued operation of the existing facility. In addition to construction impacts, higher occupational exposure would also

occur as a result of transferring the inventory of spent fuel currently stored in the ISFSI to the new pool, as well as additional exposure from maintenance and surveillance activities required to support operation of a spent fuel storage pool.

Another alternative is to license and construct and operate a new ISFSI facility on the Calvert Cliffs site. This would involve environmental impacts from construction, significant cost, and increased personal exposure associated with the transfer of fuel from the existing facility with no environmental benefit over the continued operation of the existing facility.

For these reasons, Calvert Cliffs concludes that none of these options are reasonable alternatives to the continued operation of the existing ISFSI.

E3.0 AFFECTED ENVIRONMENT

The ISFSI encompasses 6 acres and is located wholly within the Calvert Cliffs property, a 962-acre site in Calvert County, Maryland where Calvert Cliffs owns and operates two nuclear units. The site is located approximately 60 miles south of Baltimore, 40 miles southeast of Washington, D.C., 10.5 miles southeast of Prince Frederick, and 7.5 miles north of Solomons, Maryland (Figures 2.2-1 and 2.2-3 of Reference E9.5). Located on the Calvert Peninsula, the site is bordered on the east by the Chesapeake Bay. Maryland State Route 2/4 lies to the west and Flag Ponds Nature Park to the immediate north of the site. On a parcel adjacent to Calvert Cliffs to the south, UniStar Nuclear Services, a subsidiary of Constellation Energy, has proposed to construct and operate a single nuclear unit, Calvert Cliffs Unit 3. The area around the Calvert Cliffs site is primarily rural, featuring the landscape dominated by rolling hills and deciduous forests interspersed with pasture, cropland, and residential development (Reference E9.3, Section 2.0).

E3.1 LAND USE

A map depicting the land use within the Calvert Cliffs site is presented in Figure 2-4 of Reference E9.3. The combined Calvert Cliffs and the proposed Calvert Cliffs Unit 3 site encompass 2070 acres. Approximately 331 acres of the combined area is developed to accommodate the two existing nuclear units; auxiliary facilities, including the ISFSI; a barge slip; and onsite transmission line corridors. Forest covers approximately 1619 acres and 106 acres are open land that was previously used for agriculture. The Calvert Cliffs site is located within an unincorporated area of Calvert County (Reference E9.3, Figure 2.2-1).

To facilitate the development of the third nuclear unit (Calvert Cliffs Unit 3), the original Calvert Cliffs site was divided into a north parcel and a south parcel. Calvert Cliffs, which owns Units 1 and 2, retains ownership of the 962-acre north parcel (i.e., current Calvert Cliffs site). A 726-acre portion of the South Parcel upon which the new unit is to be constructed was conveyed to Calvert Cliffs Unit 3. The remaining portion of the South Parcel, consisting of approximately 382 acres, was conveyed to a sister entity of Calvert Cliffs Unit 3 (Reference E9.2, Section 2.2.1).

With its location on the Chesapeake Bay, the Calvert Cliffs site is located within the Maryland Coastal Zone, and a portion of the site is within the Chesapeake Bay Critical Area. The Chesapeake Bay Critical Area is designated as the first 1,000 ft of land inshore of the Bay and is subject to State and County regulations and ordinances that implement the Chesapeake Bay Critical Area Protection Act (Reference E9.2). The ISFSI is located well outside the Chesapeake Bay Critical Area. The Maryland Coastal Zone follows the inland boundaries of all counties bordering the Atlantic Ocean, Chesapeake Bay, and the Potomac River, which includes Calvert County (Reference E9.2). Therefore, a consistency

determination with the state of Maryland's federally approved Coastal Zone Management Program is required, and is presented in Enclosure 4.

A detailed breakdown of land use within the two Maryland counties (Calvert County and St. Mary's County), where the majority of current Calvert Cliffs Units 1 and 2 employees reside, is presented in Tables 2.2-8 and 2.2-9 of Reference E9.2. In summary, the four land use classifications, water, forest, wetlands and barren land or undeveloped land, together account for 70.8 percent of the land use in Calvert County and 79.9 percent of the land use in St. Mary's County. The major land use classifications for developed land are agricultural and residential. Agriculture accounts for 12.5 percent and 11.2 percent of the land use in Calvert County and St. Mary's County, respectively. Residential accounts for 14.6 percent and 7.0 percent of the land use in Calvert County and St. Mary's County, respectively (Reference E9.2, Section 2.2.3).

There are no known claims by Native Americans on lands within the site boundary or within the vicinity of the Calvert Cliffs site. Public recreational lands of note in Calvert County include the federally owned U.S. Naval Recreation Center at Solomons in the southern portion of the county, and Maryland's Calvert Cliffs State Park in Calvert County and Greenwell State Park, located just across the Patuxent River in St. Mary's County. Calvert County recreational facilities, Flag Ponds Nature Park, Jefferson Patterson Park and Museum, and Cove Point Park, are located within the vicinity of the Calvert Cliffs site. The American Chestnut Land Trust's Parkers Creek Watershed Nature Preserve, the largest trust property concentrated in one general location within the county, is located approximately 8 miles north of the Calvert Cliffs site (Reference E9.2, Section 2.2.1).

Other features on note in the vicinity include the Dominion Cove Point Liquified Natural Gas (LNG) import facility (a little over 100 acres of industrial land use) is about 3.5 miles southeast of the Calvert Cliffs site within Calvert County. Associated with this facility is the Dominion Cove Point LNG pipeline that is located south of the Calvert Cliffs site and continues in a northern direction west of Maryland State Route 2/4. Just beyond the 8 mile radius is the Patuxent River Naval Air Station (approximately 6500 acres) to the south of the Calvert Cliffs site in St. Mary's County (Reference E9.3, Section 2.2.3).

There are five small concentrations of population, either cities or unincorporated communities, within 10 miles of the Calvert Cliffs site. They are California, Calvert Beach-Long Beach, Chesapeake Ranch Estates-Drum Point, Lusby, and Prince Frederick (Reference E9.2, Section 2.5).

E3.2 TRANSPORTATION

A description of transportation was not included in the original ISFSI Environmental Report; therefore, this section has been provided to be consistent with the requirements of Reference E9.1. Calvert County has one main four-lane road (Maryland State Route 2/4) bisecting the County north to south with smaller roads running from the main road to the water on each side. Very few of the smaller roads off Maryland State Route 2/4 connect with each other; therefore, this highway services the bulk of the traffic for the length of the County. This highway runs adjacent to the Calvert Cliffs site and provides the only access to the site. A detailed characterization of the transportation system in the area is included in Reference E9.3.

E3.3 ECOLOGY

As noted in Table E1.3-1, information on the ecology of the Calvert Cliffs site was provided in the original ISFSI Environmental Report and the conditions of the site relative to the general ecology have not appreciably changed. Detailed information summarizing the results of recent studies done in

association with the development of the proposed Calvert Cliffs Unit 3 project can be found in Reference E9.3, Section 2.4.

This section presents the most recent information available on the occurrence of special-status species at the Calvert Cliffs site. In June 2006, UniStar requested an environmental review by the Maryland Department of Natural Resources for the proposed Calvert Cliffs Unit 3 project. The project area considered in that review includes the ISFSI site. Surveys were conducted on the Calvert Cliffs project area to locate any state and federal species that were identified as having the potential to occur on the site by Maryland Department of Natural Resources in their July 2006 response letter. Table E3.3-1 lists the state and federal special-status species that may occur at the Calvert Cliffs site. No critical habitat for threatened or endangered species occurs on the Calvert Cliffs site (Reference E9.3, Section 2.4). The surveys and habitat characterizations done in association with the Calvert Cliffs Unit 3 project were evaluated and found to be accurately representative of current site conditions. Prior to the submittal of this application, a letter requesting an environmental review for the ISFSI license renewal project was sent to the Maryland Department of Natural Resources and the Fish and Wildlife Service. Enclosure 1 provides copies of correspondence with these agencies.

While surveys conducted in support of Calvert Cliffs Unit 3 established the potential for or known occurrence of special status species listed in Table E3.3-1 on the combined Calvert Cliffs and Calvert Cliffs Unit 3 area, none of the species are known to occur within the six-acre ISFSI facility or areas of operations.

E3.4 AIR QUALITY

A description of air quality was not included in the original ISFSI Environmental Report; therefore, this section has been provided to be consistent with the requirements of Reference E9.1. Meteorology and climatology were addressed in the original ISFSI Environmental Report, and provide an accurate characterization of the conditions in the area. Based on Environmental Protection Agency data, Calvert County, Maryland, is in attainment for all the National Ambient Air Quality Standards except for the eight-hour ozone standard as of June 16, 2010 (Reference E9.9). Non-attainment of the eight-hour ozone standard is due to Calvert County's proximity to Washington, D.C.

E3.5 NOISE

In support of the NRC Construction and Operating License Application for Calvert Cliffs Unit 3, a survey was conducted in November 2006 to measure ambient environmental community noise levels. The survey results document existing conditions for a typical and representative period during the leaf-off season. During leaf-on season, fully leafed trees would attenuate or reduce traffic noise from Maryland State Route 2/4 and any existing plant emissions, both factors tending to decrease residual sound levels. The results of the survey indicate that the residual sound level in the Calvert Cliffs site area is dominated by traffic noise, and further indicate that noise from simultaneous operation of the two existing nuclear units is not audible, day or night, at all off-site survey locations. All measured ambient sound levels are attributable to normal, current environmental sources such as traffic noise, and are not related to Calvert Cliffs Units 1 and 2, which includes the ongoing operation of the ISFSI.

E3.6 HISTORICAL AND CULTURAL

Research conducted recently in association with UniStar's Calvert Cliffs Unit 3 project identified 1,029 previously surveyed, inventoried, and recorded cultural resources within a 10 mile radius of the existing Calvert Cliffs site. The resources identified are located in the Maryland counties of St. Mary's, Calvert, and Dorchester. In addition, Phase I and II investigations conducted as a part of the UniStar project

identified four architectural resources and one archaeological site eligible for listing on the National Register of Historic Places on the Calvert Cliffs site (Reference E9.2, Section 2.5). These four sites are located outside of the 6-acre ISFSI facility and areas of ISFSI operations and are shielded from views of the facility by the dense forest surrounding the ISFSI.

Table E3.3-1, Endangered, Threatened, or Special Concern Animal and Plant Species that may occur at Calvert Cliffs Site[a]

Scientific Name	Common Name	Federal Status[b]	State Status[b]
Insects			
Cicindela dorsalis dorsalis	Northeastern Beach Tiger Beetle	T	E
Cicindela puritana	Puritan Tiger Beetle	T	E
Plants			
Centrosema virginianum	Spurred butterfly pea	-	R
Lemma trisulca	Star duckweed	-	E
Potamogeton foliosus	Leafy Pondweed	-	E
Potamogeton	Claspingleaf Pondweed	-	R
Potamogeton spirillus	Spiral Pondweed	-	HR
Quercus shumardii	Shumard's Oak	-	T
Sesuvium maritimum	Sea purslane	-	E
Solidago speciosa	Showy Goldenrod	-	T
Zizaniopsis miliacea	Southern Wild Rice	-	E
Fish			
Acipenser brevirostrum	Shortnose Sturgeon	E	E
Acipenser oxyrinchus	Atlantic Sturgeon	C	-
Alosus aestivalis	Blueback Herring	SC	-
Alosus pseudoharengus	Alewife	SC	-
Fundulus luciae	Spotfin killifish	-	R
Reptiles			
Caretta caretta	Atlantic Loggerhead Turtle	E	T
Chelonia mydas	Green Turtle	T	T
Dermochelys coriacea	Leatherback Turtle	E	E
Amphibians			
Gastrophryne carolinensis	Eastern Narrow-mouthed Toad	-	E

[a] Sources: Reference E9.3.
[b] C = Candidate; E = endangered; HR = Highly Rare; R = Rare; SC = Species of Concern; T = threatened.

E3.7 VISUAL AND SCENIC

The rolling topography of the area and the dense forest cover preclude views of the Calvert Cliffs physical structures from off-site (Reference E9.3, Section 2.5.2.4). Although there are limited views of some Calvert Cliffs structures such as the intakes from the Chesapeake Bay, the ISFSI facility is over

2,000 feet landward of the Bay and at an elevation of 114 feet above mean sea level (Reference E9.5, Section 2.4).

E3.8 DEMOGRAPHY AND SOCIOECONOMIC

E3.8.1 DEMOGRAPHY

The following sections describe the current demographic and economic characteristics for the 50 mile and 10 mile geographic areas, as well as the two-county region of influence. Most demographic data generated by the U.S. Census Bureau and used in this analysis are from the year 2000 unless otherwise noted. Information presented in this section was developed by UniStar for the proposed Calvert Cliffs Unit 3 Combined License application (Reference E9.2, Section 2.5) unless noted otherwise.

The two-county region of Calvert County and St. Mary's County is considered for the socioeconomic analyses given the majority of the existing workforce for Calvert Cliffs Units 1 and 2 reside in these counties. As of May 2010 a total of 942 employees work at the Calvert Cliffs site. Of this total, 885 are Constellation Energy Nuclear Group employees and 57 are contractors. More than 89 percent of the current permanent workforce at Calvert Cliffs resides in Calvert County (63 percent) or St. Mary's County (26 percent).

The estimated resident population distribution within 50 miles of Calvert Cliffs site for the years 2000, 2010, 2020, 2030, 2040, and 2050 is listed in Tables E3.8-1 through E3.8-6. Figure 2.2-1 of Reference E9.5 illustrates the locations of the sectors identified in these tables.

The nearest major population centers within about 50 miles of the Calvert Cliffs site are Washington, D.C., located approximately 55 driving miles to the northwest and Annapolis, Maryland, 50 driving miles to the north. Smaller cities and towns within 50 driving miles include Glenarden, 50 driving miles away, North Beach, 26 driving miles, La Plata at 36 driving miles, Leonardtown which is 20 driving miles, and Seat Pleasant at 49 driving miles. Calvert County is part of the Washington-Arlington-Alexandria, DC-VA-MD-WV Metropolitan Statistical Area and shares a high degree of economic and social integration with the metropolitan area. St. Mary's County is a part of the much smaller Lexington Park, Maryland Micro Area.

The estimated resident population distribution within 10 miles of Calvert Cliffs for the years 2000, 2010, 2020, 2030, 2040, and 2050 is listed in Tables E3.8-7 through E3.8-12. Figure 2.1-3 of Reference E9.5 illustrates the locations of the sectors identified in these tables.

The area within a 10 mile radius of the Calvert Cliffs site is predominately rural, dominated by farmland and forests, clusters of residential communities, and by the waters of the Chesapeake Bay. Cities and recognizable unincorporated but named communities within a 10 mile driving distance of the Calvert Cliffs site include California, Calvert Beach-Long Beach, Chesapeake Ranch Estates-Drum Point, Lusby, and Prince Frederick.

Table E3.8-13 presents the demographic characteristics for the residential population within each of the five 10 mile circles radiating from the Calvert Cliffs site. These demographic characteristics – age and sex distributions, racial and ethnical distributions, and household income figures – present a statistical profile of the 50 mile geographic area in 2000.

The area surrounding the Calvert Cliffs site has experienced significant growth for the last three and one half decades. The projections for the total population within the 50 mile radius of the site are 29 percent higher than that projected in the original ISFSI Environmental Report for 2010 and 46 percent higher

within the 10 mile radius of the site. Population densities have increased noticeably in both Calvert and St. Mary's counties from 2000 to 2005. The year 2000 population densities were 377 people per square mile in Calvert County and 239 people per square mile in St. Mary's County. In comparison, the 2005 population density in Calvert County was 409 people per square mile and the population density in St. Mary's County was 267.4 people per square mile. Nationally, the average population density was 83.8 people per square mile in 2005.

From 2000 to 2010, the population of Calvert County was expected to grow an annual average of 2.5 percent, about two times the annual average population growth rate for the State of Maryland (1.08 percent). During that same period, the population of St. Mary's County was expected to grow an annual average of 2.25 percent, also substantially higher than the average growth rate in Maryland. The combined two-county population is expected to grow from 160,774 in 2000 to 353,444 in 2050.

E3.8.2 COMMUTERS AND TRANSIENT POPULATION

Out-commuting represents a significant change to the population base in the Calvert and St. Mary's Counties. Based on 2000 Census Bureau County-to-County Worker Flow survey data, these counties experienced a net loss of 20,931 persons during the work week/work day/work hour period. Recreational use is considered the primary contributor to transient populations in the area. The Southern Region of Maryland, which includes Calvert County, St. Mary's County, and Charles County, had 541,791 visitors in 2004. Major parks within the 10 mile radius include Calvert Cliffs State Park to the south and Flag Ponds Park to the north of the Calvert Cliffs site.

E3.8.3 MINORITY AND LOW-INCOME POPULATIONS

This section presents the analysis of the minority and low-income populations that was recently done as part of the environmental assessment for the proposed Calvert Cliffs Unit 3 project. Use of this information is appropriate as the geographic area is essentially the same for both actions and the analysis used Census 2000 data.

Using ArcView® GIS software and U.S. Census Bureau's 2000 census data, all census block groups within the 50 mile radius were identified. A census block group was included in the 50 mile comparative geographic area if its boundaries were fully contained in the area, or if any part of the census block group was contained in the area. The ArcView® GIS software and U.S. Census Bureau's 2000 census data were then used to determine the minority and low-income characteristics, by census block group, within 50 miles of the Calvert Cliffs site.

The 50 mile radius contains a total of 2,177 census block groups; 1,116 in Maryland, 605 in Virginia, 433 in Washington D.C., and 23 in Delaware.

E3.8.3.1 Minority Populations

A "minority" racial population is defined as: American Indian or Alaskan Native; Asian, Native Hawaiian, or other Pacific Islander; Black (African-American) races; and multi-racial, or "some other race," and Hispanic ethnicity (Reference E9.10). The racial population is expressed in terms of the number and/or percentage of people that are minorities in an area. The sum of these racial minority populations is referred to as the aggregate racial minority population.

The NRC guidance indicates that a minority population exists if either of the following two criteria is met:

- The minority population of the census block group or environmental impact area (in this case the 50 mile geographic area) exceeds 50 percent; or
- The minority population percentage of the environmental impact area is significantly greater (typically at least 20 percentage points) than the minority population percentage in the geographic area chosen for comparative analysis (in this case the 50 mile geographic area).

For each of the 2,177 census block groups within the 50 mile radius, the percent of the census block group's population represented by each minority classification (each race, aggregate minority population, and Hispanic/Latino origin) was calculated and compared to the two criteria listed above. If any census block group minority percentage exceeded 50 percent, then the block group was identified as containing a minority population. If any census block group percentage exceeded the applicable percentage in the 50 mile geographical area by more than 20 percentage points, then the census block group was identified as containing a minority population. Table 2.5-45 and Figure 2.5-4 through Figure 2.5-8 of Reference E9.2 identify the various minority block groups. Within the 50 mile geographic area there are a total of 891 census block groups that are classified as having minority populations. Maryland has 463 minority census block groups, Virginia has 113 blocks, Washington D.C. has 312 blocks, and Delaware has three minority census block groups.

There are no federally recognized Native American tribes within the 50 mile geographic area or within the State of Maryland. However, non-recognized Native American tribes and communities include the Piscataway-Conroy Confederacy based in La Plata, in Charles County. There are established Amish and Mennonite communities in the northwestern section of St. Mary's County, Maryland.

E3.8.3.1 Low-Income Populations

The Census Bureau's definition of a low-income household is based on governmental statistical poverty thresholds. For the purposes of conducting this analysis, a block group is considered to be low income if either of the following two criteria are met:

- The number of low-income households in the census block group or the environmental impact site (in this case the 50 mile geographic area) exceeds 50 percent; or
- The percentage of households below the poverty level in an environmental impact area is significantly greater (typically at least 20 percentage points) than the low-income population percentage in the geographic area chosen for comparative analysis (in this case, the 50 mile geographic area).

Using 2000 Census data, low-income households in each census block group were divided by the total households for that census block group to obtain the percentage of low-income households per block group. If any census block group low-income percentage exceeded 50 percent, then the block group was identified as containing a low-income population. If any census block group percentage exceeded the applicable percentage in the geographical area by more than 20 percentage points, then the census block group was identified as containing a low-income population.

Low-income census block group information displayed in Tables 2.5-44 and 2.5-46, and Figure 2.5-9 of Reference E9.2 shows the locations of the low-income block groups. There are very few concentrations of low-income populations within 50 mile radius of the Calvert Cliffs site. Only 67 census block groups are classified as having low-income populations with 27 in Maryland, 3 in Virginia, 35 in Washington

D.C., and 2 in Delaware. There are no low-income census block groups in Calvert County and only 1 in St. Mary's County.

Table E3.8-1, Estimated Population Distribution in 2000 within 50 miles of Calvert Cliffs[a]

Sector	0-10 Miles	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles	50 Mile Total
N	0	0	7,848	91,036	144,479	243,363
NNE	0	403	6,479	13,477	9,394	29,753
NE	1	1,020	8,948	19,513	14,160	43,642
ENE	198	1,668	17,492	9,015	29,169	57,542
E	35	236	468	4,739	77,460	82,938
ESE	0	709	594	3,635	15,217	20,155
SE	188	183	0	1,030	7,158	8,559
SSE	8,378	477	795	1,136	1,855	12,641
S	7,631	20,464	2,277	5,420	7,210	43,002
SSW	5,915	16,134	4,340	8,751	6,820	41,960
SW	2,931	8487	2,985	3,412	5,020	22,835
WSW	3,909	7558	4,213	13,953	7,842	37,475
W	2,613	11,560	8,962	8,346	25,052	56,533
WNW	3,050	11,857	54,835	67,423	346,300	483,465
NW	5,178	11,561	19,014	272,660	1,285,806	1,594,219
NNW	718	20,524	22,756	95,361	284,819	424,178
Total	**40,745**	**112,841**	**162,006**	**618,907**	**2,267,761**	**3,202,260**

a Source: Reference E9.2.

Table E3.8-2, Estimated Population Distribution in 2010 within 50 miles of Calvert Cliffs[a]

Sector	0-10 Miles	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles	50 Mile Total
N	0	0	8,916	103,420	164,125	276,461
NNE	0	461	7,356	15,310	10,664	33,791
NE	1	1,157	10,155	22,165	16,076	49,554
ENE	224	1,895	19,871	10,231	33,109	65,330
E	40	268	532	5,378	87,967	94,185
ESE	0	804	675	4,127	17,284	22,890
SE	214	207	0	1,172	8,128	9,721
SSE	9,518	541	902	1,284	2,107	14,352
S	8,658	23,249	2,586	6,140	8,177	48,810
SSW	6,719	18,326	4,920	9,943	7,747	47,655
SW	3,329	9,636	3,383	3,872	5,703	25,923
WSW	4,435	8,584	4,778	15,845	8,907	42,549
W	2,972	13,135	10,176	9,480	28,458	64,221
WNW	3,466	13,469	62,305	76,586	393,439	549,265
NW	5,882	13,127	21,594	309,789	1,460,774	1,811,166
NNW	814	23,311	25,842	108,344	323,581	481,892
Total	**46,272**	**128,170**	**183,991**	**703,086**	**2,576,246**	**3,637,765**

a Source: Reference E9.2.

Table E3.8-3, Estimated Population Distribution in 2020 within 50 miles of Calvert Cliffs[a]

Sector	0-10 Miles	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles	50 Mile Total
N	0	0	9,844	114,165	181,180	305,189
NNE	0	506	8,141	16,906	11,795	37,348
NE	1	1,283	11,239	24,489	17,798	54,810
ENE	250	2,094	21,944	11,333	36,634	72,255
E	44	297	590	5,949	97,178	104,058
ESE	0	891	745	4,568	19,093	25,297
SE	239	231	0	1,293	8,985	10,748
SSE	10,512	599	1,001	1,435	2,335	15,882
S	9,580	25,666	2,864	6,823	9,081	54,014
SSW	7,421	20,228	5,454	10,980	8,568	52,651
SW	3,679	10,656	3,753	4,284	6,296	28,668
WSW	4,900	9,476	5,295	17,523	9,836	47,030
W	3,278	14,505	11,255	10,466	31,418	70,922
WNW	3,826	14,875	68,765	84,554	434,233	606,253
NW	6,496	14,498	23,845	341,869	1,612,230	1,998,938
NNW	900	25,737	28,544	119,564	357,146	531,891
Total	**51,126**	**141,542**	**203,279**	**776,201**	**2,843,806**	**4,015,954**

[a] Source: Reference E9.2.

Table E3.8-4, Estimated Population Distribution in 2030 within 50 miles of Calvert Cliffs[a]

Sector	0-10 Miles	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles	50 Mile Total
N	0	0	10,636	123,437	195,922	329,995
NNE	0	545	8,783	18,277	12,732	40,337
NE	1	1,381	12,130	26,455	19,197	59,164
ENE	268	2,258	23,722	12,217	39,535	78,000
E	48	319	634	6,419	105,036	112,456
ESE	0	960	806	4,925	20,628	27,319
SE	253	248	0	1,392	9,699	11,592
SSE	11,364	647	1,076	1,539	2,512	17,138
S	10,348	27,746	3,091	7,345	9,770	58,300
SSW	8,021	21,882	5,875	11,866	9,244	56,888
SW	3,974	11,510	4,044	4,625	6,803	30,956
WSW	5,306	10,242	5,705	18,910	10,630	50,793
W	3,543	15,672	12,155	11,308	33,967	76,645
WNW	4,139	16,079	74,356	91,429	469,619	655,622
NW	7,020	15,677	25,784	369,758	1,743,683	2,161,922
NNW	971	27,822	30,850	129,306	386,236	575,185
Total	**55,256**	**152,988**	**219,647**	**839,208**	**3,075,213**	**4,342,312**

[a] Source: Reference E9.2.

Table E3.8-5, Estimated Population Distribution in 2040 within 50 miles of Calvert Cliffs[a]

Sector	0-10 Miles	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles	50 Mile Total
N	0	0	11,883	137,816	218,681	368,380
NNE	0	609	9,815	20,403	14,241	45,068
NE	2	1,549	13,560	29,555	21,463	66,129
ENE	301	2,531	26,491	13,675	44,180	87,178
E	54	358	,713	7,187	117,302	125,614
ESE	0	1,076	901	5,512	23,047	30,536
SE	289	277	0	1,560	10,849	12,975
SSE	12,682	721	1,210	1,731	2,817	19,161
S	11,557	30,976	3,455	8,223	10,956	65,167
SSW	8,958	24,428	6,588	13,252	10,338	63,564
SW	4,437	12,858	4,531	5,172	7,602	34,600
WSW	5,922	11,448	6,399	21,133	11,885	56,787
W	3,958	17,505	13,580	12,642	37,923	85,608
WNW	4,620	17,947	82,998	102,053	524,107	731,725
NW	7,845	17,503	28,786	412,679	1946,050	2,412,863
NNW	1,091	31,063	34,449	144,322	431,074	641,999
Total	**61,716**	**170,849**	**245,359**	**936,915**	**3,432,515**	**4,847,354**

[a] Source: Reference E9.2.

Table E3.8-6, Estimated Population Distribution in 2050 within 50 miles of Calvert Cliffs[a]

Sector	0-10 Miles	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles	50 Mile Total
N	0	0	12,850	149,109	236,615	398,574
NNE	0	662	10,609	22,073	15,381	48,725
NE	2	1,669	14,655	31,949	23,183	71,458
ENE	324	2,735	28,653	14,757	47,771	94,240
E	57	387	767	7,767	126,849	135,827
ESE	0	1,164	975	5,952	24,924	33,015
SE	306	299	0	1,692	11,725	14,022
SSE	13,719	780	1,303	1,859	3,036	20,697
S	12,493	33,525	3,731	8,872	11,813	70,434
SSW	9,689	26,419	7,106	14,337	11,171	68,722
SW	4,801	13,901	4,892	5,592	8,223	37,409
WSW	6,402	12,377	6,899	22,858	12,841	61,377
W	4,279	18,932	14,672	13,669	41,032	92,584
WNW	4,995	19,416	89,805	110,420	567,169	791,805
NW	8,479	18,932	31,138	446,573	2,105,862	2,610,984
NNW	1,177	33,613	37,266	156,196	466,477	694,729
Total	**66,723**	**184,811**	**265,321**	**1,013,675**	**3,714,072**	**5,244,602**

[a] Source: Reference E9.2.

Table E3.8-7, Estimated Population Distribution in 2000 within 10 miles of Calvert Cliffs[a]

Sector	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles	10 Mile Total
N	0	0	0	0	0	0	0
NNE	0	0	0	0	0	0	0
NE	0	0	0	0	0	1	1
ENE	0	0	0	0	0	198	198
E	0	0	0	0	0	35	35
ESE	0	0	0	0	0	0	0
SE	0	0	0	0	188	0	188
SSE	0	0	33	439	3,242	4,664	8,378
S	0	67	28	189	1,504	5,843	7,631
SSW	0	43	207	143	204	5,318	5,915
SW	0	329	0	165	57	2,380	2,931
WSW	0	857	612	65	85	2,290	3,909
W	30	432	289	175	357	1,330	2,613
WNW	0	55	59	85	506	2,345	3,050
NW	0	432	1,006	1,037	287	2,416	5,178
NNW	0	0	0	0	0	718	718
Total	**30**	**2,215**	**2,234**	**2,298**	**6,430**	**27,538**	**40,745**

[a] Source: Reference E9.2.

Table E3.8-8, Projected Population Distribution in 2010 within 10 miles of Calvert Cliffs[a]

Sector	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles	10 Mile Total
N	0	0	0	0	0	0	0
NNE	0	0	0	0	0	0	0
NE	0	0	0	0	0	1	1
ENE	0	0	0	0	0	224	224
E	0	0	0	0	0	40	40
ESE	0	0	0	0	0	0	0
SE	0	0	0	0	214	0	214
SSE	0	0	37	498	3,681	5,302	9,518
S	0	76	32	215	1,705	6,630	8,658
SSW	0	49	235	162	232	6,041	6,719
SW	0	374	0	187	65	2,703	3,329
WSW	0	972	695	74	96	2,598	4,435
W	34	492	329	199	406	1,512	2,972
WNW	0	62	67	97	575	2,665	3,466
NW	0	491	1,144	1,177	328	2,742	5,882
NNW	0	0	0	0	0	814	814
Total	**34**	**2,516**	**2,539**	**2,609**	**7,302**	**31,272**	**46,272**

[a] Source: Reference E9.2.

Table E3.8-9, Projected Population Distribution in 2020 within 10 miles of Calvert Cliffs[a]

Sector	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles	10 Mile Total
N	0	0	0	0	0	0	0
NNE	0	0	0	0	0	0	0
NE	0	0	0	0	0	1	1
ENE	0	0	0	0	0	250	250
E	0	0	0	0	0	44	44
ESE	0	0	0	0	0	0	0
SE	0	0	0	0	239	0	239
SSE	0	0	41	551	4,065	5,855	10,512
S	0	84	35	236	1,886	7,339	9,580
SSW	0	55	259	180	257	6,670	7,421
SW	0	414	0	207	73	2,985	3,679
WSW	0	1,074	767	81	106	2,872	4,900
W	38	542	362	219	448	1,669	3,278
WNW	0	69	74	107	635	2,941	3,826
NW	0	542	1,260	1,302	361	3,031	6,496
NNW	0	0	0	0	0	900	900
Total	**38**	**2,780**	**2,798**	**2,883**	**8,070**	**34,557**	**51,126**

[a] Source: Reference E9.2.

Table E3.8-10, Projected Population Distribution in 2030 within 10 miles of Calvert Cliffs[a]

Sector	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles	10 Mile Total
N	0	0	0	0	0	0	0
NNE	0	0	0	0	0	0	0
NE	0	0	0	0	0	1	1
ENE	0	0	0	0	0	268	268
E	0	0	0	0	0	48	48
ESE	0	0	0	0	0	0	0
SE	0	0	0	0	253	0	253
SSE	0	0	45	596	4,398	6,325	11,364
S	0	91	38	255	2,040	7,924	10,348
SSW	0	58	281	194	276	7,212	8,021
SW	0	446	0	224	76	3,228	3,974
WSW	0	1,165	830	88	115	3,108	5,306
W	41	585	391	237	485	1,804	3,543
WNW	0	74	81	115	687	3,182	4,139
NW	0	586	1,362	1,407	390	3,275	7,020
NNW	0	0	0	0	0	971	971
Total	**41**	**3,005**	**3,028**	**3,116**	**8,720**	**37,346**	**55,256**

[a] Source: Reference E9.2.

Table E3.8-11, Projected Population Distribution in 2040 within 10 miles of Calvert Cliffs[a]

Sector	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles	10 Mile Total
N	0	0	0	0	0	0	0
NNE	0	0	0	0	0	0	0
NE	0	0	0	0	0	2	2
ENE	0	0	0	0	0	301	301
E	0	0	0	0	0	54	54
ESE	0	0	0	0	0	0	0
SE	0	0	0	0	289	0	289
SSE	0	0	50	664	4,909	7,059	12,682
S	0	102	42	286	2,280	8,847	11,557
SSW	0	65	312	217	309	8,055	8,958
SW	0	497	0	250	86	3,604	4,437
WSW	0	1,297	927	98	129	3,471	5,922
W	45	654	439	265	541	2,014	3,958
WNW	0	84	90	129	766	3,551	4,620
NW	0	654	1,524	1,570	435	3,662	7,845
NNW	0	0	0	0	0	1,091	1,091
Total	**45**	**3,353**	**3,384**	**3,479**	**9,744**	**41,711**	**61,716**

[a] Source: Reference E9.2.

Table E3.8-12, Projected Population Distribution in 2050 within 10 miles of Calvert Cliffs[a]

Sector	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles	10 Mile Total
N	0	0	0	0	0	0	0
NNE	0	0	0	0	0	0	0
NE	0	0	0	0	0	2	2
ENE	0	0	0	0	0	324	324
E	0	0	0	0	0	57	57
ESE	0	0	0	0	0	0	0
SE	0	0	0	0	306	0	306
SSE	0	0	54	718	5,307	7,640	13,719
S	0	110	46	310	2,462	9,565	12,493
SSW	0	70	339	234	334	8,712	9,689
SW	0	538	0	269	94	3,900	4,801
WSW	0	1,403	1,002	106	140	3,751	6,402
W	49	707	475	287	584	2,177	4,279
WNW	0	90	96	139	828	3,842	4,995
NW	0	708	1,646	1,697	470	3,958	8,479
NNW	0	0	0	0	0	1,177	1,177
Total	**49**	**3,626**	**3,658**	**3,760**	**10,525**	**45,105**	**66,723**

[a] Source: Reference E9.2.

Table E3.8-13, Select Demographic and Economic Characteristics of Residential Populations, by Distance from the Calvert Cliffs Site, 2000[a]

Demographic and Economic Characteristics	Radial Distance, mile (km)					
	0-10 mile (0-16 km)	10-20 mile (16-32 km)	20-30 mile (32-48 km)	30-40 mile (48-60 km)	40-50 mile (60-80 km)	50 Mile Total (80-km Total)
Total Population[b]	40,745	111,659	163,358	618,846	2,259,157	3,195,170
Age Composition:						
Persons under 5 yrs old	2,992	7,588	10,873	41,578	148,788	211,819
Persons 18 yrs and over	29,458	80,295	120,226	456,584	1,738,152	2,424,715
Persons 65 yrs and older	4,203	9,721	18,951	61,657	218,766	313,298
Gender Composition:						
Females	21,169	55,925	83,981	322,859	1,161,278	1,645,212
Ethnic Composition:						
Caucasians[c]	35,454	91,113	116,465	265,801	1,170,147	1,678,980
African-American[c]	5,219	15,657	40,378	322,496	767,075	1,150,825
Hispanic-Latino[d]	782	1,885	2,578	14,135	241,685	261,065
Income Characteristics:						
Median Household Income, 1999 [e]	$61,369	$59,241	$57,945	$60,221	$57,464	$57,464

[a] Source: Reference E9.2, Section 2.5.
[b] Resident population excludes transient populations.
[c] Persons describing themselves as of one race only.
[d] Persons of Hispanic and Latino origin may of any race.
[e] Median Household Income is the median income for the cumulative households from the Calvert Cliffs site; for example, Median Household Income column labeled 30-40 miles is the median for all households within 0-40 miles from the plant site.

E4.0 ENVIRONMENTAL IMPACTS

No refurbishment or construction is required to support operations of the ISFSI during the 40-year license renewal period. Therefore, there will be no impacts from refurbishment or construction. The following sections discuss environmental consequences associated with continued operations of the ISFSI.

E4.1 OCCUPATIONAL AND PUBLIC HEALTH

There are no normal or off-normal releases or effluents expected from the double-sealed storage canisters; therefore, only direct and reflected radiation from the ISFSI is considered in the occupational and public dose assessments.

E4.1.1 OCCUPATIONAL DOSE

Radiation protection and doses from the ISFSI operations are discussed in Chapter 7 of Reference E9.6. Calvert Cliffs is committed to a program of keeping occupational radiation exposure as low as reasonably achievable. Calvert Cliffs follows appropriate NRC Regulatory Guides as well as 10 CFR Part 20. Section 2.4 of the ISFSI Technical Specifications establish the following dose rate limits:

- 20 mrem/hr contact dose rate from the HSM sides
- 100 mrem/hr contact dose rate at the HSM door and vents.

The license amendment for using a NUHOMS-32P DSC (Reference E9.7) confirmed that there is no significant increase in individual or cumulative occupational radiation exposure and that the corresponding dose is bounded by the original Technical Specification limits. Calculations submitted in

support of the higher burnup fuel license amendment request (Reference E9.8) confirm that these limits would be maintained for that higher burnup fuel. Section 7.4 of Reference E9.6 provides a detailed discussion.

E4.1.2 PUBLIC DOSE

Doses to any offsite receptor are only from direct and scatter gamma radiation from the HSM. The estimated dose from the storage modules to any receptor beyond the site boundary is well below regulatory limits even when combined with station doses. The maximum exposed member of the public maintaining continuous occupancy in the nearest residence, which is 4,075 feet from the ISFSI facility, is less than 2 mrem/year when all 120 HSMs are filled to capacity and less than 13.5 mrem/year for the remaining uranium fuel cycle activities in the area. The collective dose for persons within 0-2 miles of the ISFSI is conservatively estimated to be 24.3 mrem/year over 2,771 people. This is less than 1 percent of the collective dose from the remaining fuel cycle operations. Therefore, radiation exposure due to the ISFSI, combined with all other fuel cycle operations, would not exceed the regulatory requirements of 25 mrem/year in 10 CFR 72.104 and 40 CFR Part 190. Section 7.6 of Reference E9.6 provides a detailed discussion of public dose.

E4.2 OTHER IMPACTS

Continued operations of the ISFSI during the 40-year license renewal term would have little to no impact on the following resources:

- Land Use
- Transportation
- Geology and soils
- Water resources
- Ecological resources
- Air quality
- Noise
- Historic and cultural resources
- Visual/scenic resources
- Socioeconomics
- Environmental Justice
- Waste Management

The ISFSI is a totally passive installation that is designed to provide shielding and safe confinement of spent nuclear fuel. There are no liquid discharges from the facility, so no geologic, water, or aquatic ecological resources would be affected. There are no air emissions from the facility, so air quality would be unaffected.

As noted, operation, maintenance and surveillance activities would continue to be performed by Calvert Cliffs personnel, and no additional staff would be required to conduct these activities during the license renewal term. Therefore, there would be no significant adverse effect on transportation or regional socioeconomics during the license renewal term. Recent population increases indicate spent fuel storage has not been a significant deterrent to immigration or growth, so continued operations of the facility would have little effect, if any, on land use patterns in the region.

The ISFSI and associated operations are located in an intensively developed area, and as discussed in Section E3.3, special-status species are not known to exist within the 6-acre ISFSI facility or areas on or affected by operation. In addition, Calvert Cliffs is not aware of any activities associated with continued

operations of the ISFSI facility that would adversely impact federal or state threatened or endangered species or other state-listed species with special status designations.

As discussed in Section E3.6, extensive cultural and archeological studies performed recently in support of the proposed Calvert Cliffs Unit 3 project identified four architectural resources and one archaeological site eligible for listing on the National Historic Register of Historic Places on the combined Calvert Cliffs and Calvert Cliffs Unit 3 site areas. Calvert Cliffs is not aware of any activities associated with operation of the ISFSI that would adversely impact these resources. Furthermore, views of the ISFSI facility from these resources and from all off-site locations, including the Chesapeake Bay, are shielded by the surrounding vegetation and topography. In addition, acoustic surveys indicate that operational noise is not detectable offsite. Therefore, no adverse effects to historic, cultural, archaeological, scenic, or aesthetic resource would result from renewal of the ISFSI license for another 40 years.

Based on the Calvert Cliffs review, the ISFSI license renewal and continued operations would result in no significant impact. No member of the public would be substantially affected and, as a consequence, there would be no disproportionately high and adverse impacts on any member of the public, including minority and low-income populations.

E4.3 IMPACTS FROM POTENTIAL ACCIDENTS

Section 8.2 of Reference E9.6 addresses the following design events and other accidents which could impact the safe operation of the ISFSI.

- Loss of air outlet shielding
- Tornado winds/tornado missiles
- Earthquake
- Flood
- Transfer cask drop
- Lightning
- Blockage of air inlets and outlets
- DSC leakage
- Accidental pressurization of DSC
- Forest fire
- LNG plant or pipeline spill or explosion

Two of these postulated accidents, loss of air outlet shielding and floods were deemed not applicable to the ISFSI. The following discussion summarizes the results of the accident analyses in terms of dose consequences, with reference to the specific USAR section that details the accident analysis.

Tornado winds/tornado missiles – analysis concludes all components of the ISFSI are capable of safely withstanding tornado wind loads and tornado generated missiles. Therefore, there is no accident dose associated with the design basis tornado (see Reference E9.6, Section 8.2.2). Results bound both DSC designs and fuel with burnup rates up to 52 GWd/MTU.

Earthquake – analysis concludes that major components of the ISFSI have been designed and evaluated to withstand the forces generated by the design basis earthquake; hence, there are no dose consequences (see Reference E9.6, Section 8.2.3).

Transfer cask drop – analyses have shown that transfer cask, NUHOMS-32P DSC, its internal basket assembly, and contained fuel will maintain their structural integrity through a cask drop. However,

for conservatism, it was assumed that the entire neutron shielding was lost as a result of the design drop accident. This would result in an increase in the cask surface contact dose 1,518 mrem/hour on contact, and 127.6 mrem/hour at 15 feet. An onsite worker at an average distance of 15 feet for the 8-hour recovery time would receive an additional dose of 1,021 mrem (see Reference E9.6, Section 12.8.2.5). Analysis for the proposed NUHOMS-32P high burnup fuel results in a dose of 1,751.7 mrem/hour at contact and 145.5 mrem/hour at 15 feet. Accordingly, the onsite worker is estimated to receive a dose of 1164 mrem at 15 feet over the 8-hour recovery period (Reference E9.8, Attachment 5). Both the accident dose rate and the recovery dose meet the 10 CFR 72.106 limit.

Lightning – the analysis concludes that no off-normal operating condition would develop as a result of lightning striking in the vicinity of the ISFSI; therefore, no dose consequences would occur (see Reference E9.6, Section 8.2.6). This conclusion is applicable to both DSC designs, as well as for the DSCs loaded with the higher burnup fuel.

Blockage of air inlets and outlets – the analysis concludes there are no off-site dose consequences, and the only possible dose increase is related to a recovery operation where the on-site worker could receive an additional 584 mrem during an estimated 8-hour debris removal period for the NUHOMS-24P (see Reference E9.6, Section 8.2.7), and 544 mrem for the NUHOMS-32P for the same recovery period (see Reference E9.6, Section 12.8.2.7). The results for the proposed NUHOMS-32P high burnup fuel are bounded by these results (Reference E9.8).

Dry shielded canister leakage – the passive nature of the Calvert Cliffs NUHOMS system and the various design features preclude any credible event that could result in the rupture of all fuel rods concurrent with DSC leakage; however, for conservatism, this accident assumes that the fuel rods and the DSC pressure boundary are ruptured during the hypothetical event. Whole body and maximum organ doses are calculated for the hypothetical individual assumed to be present at the nearest controlled area boundary location (approximately 3,900 feet from the facility) for the duration of the event. The resulting calculated doses for the NUHOMS-24P are 0.1 mrem and 17.8 mrem for the maximum off-site total body and skin doses, respectively. For the NUHOMS-32P using the same methodology except a 24-month operating cycle, the calculated doses are 0.36 mrem for total body dose and 60.1 mrem for off-site skin dose. The calculated doses for the NUHOMS-32P high burnup fuel are 109.6 and 0.65 mrem for skin and total body doses, respectively (Reference E9.8). The results for both designs and the high burnup fuel are well within the 10 CFR 72.106 limit of 5,000 mrem (see Reference E9.6, Sections 8.2.8 and 12.8.2.8).

Accidental pressurization of dry shielded canister – analysis indicates that the maximum DSC accident pressure is within the design basis limits; therefore, there is no dose consequence would result (see Reference E9.6, Section 8.2.9). This conclusion is applicable to both DSC designs, as well as for the DSCs loaded with the higher burnup fuel.

Forest fire – analysis concludes that there are no dose consequences associated with the postulated forest fire accident (see Reference E9.6, Section 8.2.10). This conclusion is applicable to both DSC designs, as well as for the DSCs loaded with the higher burnup fuel.

LNG plant or pipeline spill or explosion – analysis concludes that operation of the nearby Cove Point LNG facility would not present any undue hazards to Calvert Cliffs or the ISFSI (see Reference E9.6, Section 8.2.11).

In summary, the accident dose rates for the DSC with the proposed neutron and gamma source terms associated with the high burn up fuel meet the 10 CFR 72.106 regulatory limit for all scenarios analyzed. The accident analyses summarized above would bound operations through the ISFSI 40-year license renewal period.

E5.0 MITIGATION MEASURES

Based upon the assessments provided in Section E4.0, the impacts of license renewal and continued operation of the ISFSI are small and would not require additional mitigation. Current operations include mitigation activities that would continue during the term of the renewed license. Calvert Cliffs performs routine monitoring activities and any associated mitigation to ensure the safety of workers, the public and the environment. These monitoring activities include the Radiological Environmental Monitoring Program (REMP) conducted for Calvert Cliffs Units 1 and 2 and the ISFSI, periodic monitoring of the HSMs and preventative maintenance as necessary, and monitoring and maintenance of the perimeter and security fences.

E6.0 ENVIRONMENTAL MEASUREMENT AND MONITORING

Current monitoring requirements for the ISFSI facility are defined in ISFSI Technical Specification 6.2, and comprise the REMP. The objectives of the ISFSI REMP program are:

- To satisfy the community concern regarding the impact of the ISFSI on the environment,
- To verify that radioactivity and ambient radiation levels attributable to operation of the ISFSI are within the limits specified in the Environmental Radiation Protection Standards as stated in 40 CFR Part 190,
- To detect any measurable buildup of long-lived radionuclides in the environment due to the ISFSI,
- To monitor and evaluate ambient radiation levels around the ISFSI, and
- To determine whether any statistically significant increase occurs in the concentration of radionuclides near the ISFSI.

Details of this program and results can be found in Reference E9.11.

There are no other physical, chemical, or ecological monitoring requirements beyond that described above to support operations of the facility. The proposed action does not involve any changes to the Technical Specifications, refurbishment or changes in operation that would impact the design of the REMP; therefore, this program would continue through the license renewal period, and no additional environmental measurement or monitoring would be required beyond that of the current REMP described above.

E7.0 COST – BENEFIT ANALYSIS

The largest direct benefit resulting from the proposed action, renewal of the ISFSI for an additional 40 years of operation, is maintaining continued capability for safe, secure storage of spent fuel. The incremental increase in operating costs due to aging management activities is insignificant compared to the construction costs of a new dry fuel storage installation on the current site property or the construction and maintenance costs associated with a new spent fuel pool. As explained in Section E2.0, these are the only two viable alternatives available. Both of these alternatives would also involve additional personnel radiation dose from transferring fuel, and maintenance requirements of a new spent fuel pool which would involve added radiation dose for the operational term.

A quantitative evaluation of environmental costs of alternatives is not necessary to recognize that significant environmental impacts would be avoided by continued operation of the ISFSI versus the

limited available options for storage. As summarized in the environmental information submitted for the Oconee ISFSI license renewal application, the NRC has extensive experience evaluating environmental impacts from ISFSI facilities, and the NRC has consistently issued findings of no significant impact for the operation of the facilities (Reference E9.12, Section 1.2).

Based upon these considerations, it is reasonable to conclude the proposed license renewal of the ISFSI for an additional 40 years of operation would provide an economic advantage over other storage alternatives. The proposed license renewal involves a cost-effective utilization of an existing asset, with relatively minor environmental impact, making it the preferred means of securing long-term spent fuel storage capability.

E8.0 SUMMARY OF CONSEQUENCES

All impacts of license renewal of the ISFSI are small and would not require additional mitigation.

E8.1 UNAVOIDABLE ADVERSE IMPACTS

License renewal and continued operations of the ISFSI would incur no unavoidable adverse impacts as a result of normal operations. No credible accidents would increase the dose to the public beyond a minimal amount, and occupational doses from a conservative analysis of a credible accident would be a maximum of 1,164 mrem during the 8-hour period to remove debris from a blocked inlet or outlet vent.

E8.2 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENT OF RESOURCES

The continued operation of ISFSI for the license renewal term will result in irreversible and irretrievable resource commitments including:

- Land required to permanently store or dispose of the spent nuclear fuel and low-level radioactive wastes generated from facility operations,
- Elemental materials that will become radioactive, and
- Materials used for the normal industrial operations of the facility that cannot be recovered or recycled or that are consumed or reduced to unrecoverable forms.

E8.3 SHORT-TERM USES, MAINTENANCE, AND ENHANCEMENT OF LONG-TERM PRODUCTIVITY

The current balance between short-term use and long-term productivity of the environment at the ISFSI site has remained relatively constant since the facility began operating in 1992. This balance is described in the original Environmental Report (Reference E9.5) and the NRC Environmental Assessment (Reference E9.13), which noted the conversion of approximately six acres of land to facilities for spent fuel storage. Much of the land in the vicinity of ISFSI had once been farmed, and the ISFSI was originally located in an area that was previously disturbed during the construction of Calvert Cliffs Units 1 and 2.

Calvert Cliffs notes that the current balance is now well established and can be expected to remain essentially unchanged by renewal of the ISFSI operating license and extended operation of the facility. Extended operation of the ISFSI would postpone restoration of the site and its potential availability for uses other than fuel storage for up to an additional 40 years.

E9.0 REFERENCES

E9.1 NUREG-1748, August 2003, Environmental Review Guidance for Licensing Actions Associated with NMSS Programs

E9.2 UNS (UniStar Nuclear Services, LLC), Revision 6, September 2009, Calvert Cliffs Nuclear Power Plant Unit 3 Combined License Application Part 3: Environmental Report

E9.3 NUREG-1936, Volume 1, April 2010, Environmental Impact Statement for Combined License (COL) for Calvert Cliffs Nuclear Power Plant Unit 3 (Draft Report for Comment)

E9.4 National Oceanic and Atmospheric Administration 2010, Ocean and Coastal Management in Maryland, Accessed online: http://coastalmanagement.noaa.gov/mystate/md.html. date accessed: 5/24/2010

E9.5 Letter from Mr. G. C. Creel (BGE) to Director, Division of Industrial and Medical Nuclear Safety; Office of Nuclear Material Safety and Safeguards (NRC), dated December 12, 1989, Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) Application, Chapter 12 - Environmental Report

E9.6 Calvert Cliffs Independent Spent Fuel Storage Installation Updated Safety Analysis Report, Revision 17, September 8, 2008

E9.7 Letter from R. J. Lewis (NRC) to G. Vanderheyden (CCNPP), dated June 10, 2005, Amendment 6 to Material License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation

E9.8 Letter from J. A. Spina (CCNPP) to Document Control Desk (NRC), dated June 15, 2009, License Amendment Request: Allow Increased Burnup Fuel to be Loaded into NUHOM-32P Dry Shielded Canister

E9.9 EPA (U.S. Environmental Protection Agency) 2010, Maryland, Delaware, Washington, D.C. 8-hour Ozone Nonattainment Areas, http://www.epa.gov/air/oagps/grconbk/mddedc8.html accessed July 8, 2010

E9.10 LIC-203, Revision 1, May 24, 2004, Procedural Guidance for Preparing Environmental Assessments and Considering Environmental Issues, Office of Nuclear Reactor Regulation

E9.11 Letter from J. S. Gaines (CCNPP) to Document Control Desk (NRC), dated May 15, 2009, Annual Radiological Environmental Operating Report

E9.12 Letter from D. A. Baxter (Duke) to Document Control Desk (NRC), dated January 30, 2008, Site-Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application, Environmental Report Supplement

E9.13 Letter from C. J. Haughney (NRC) to G. C. Creel (BGE), dated March 22, 1991, Environmental Assessment Related to Construction and Operation of the Calvert Cliffs Independent Spent Fuel Storage Installation

E9.14 Letter from R. A. Nelson (NRC) to G. Vanderheyden (CCNPP), dated November 2, 2005, Amendment 7 to Materials License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation

E10.0 LIST OF PREPARERS

This section presents a list of the individuals who contributed to the technical content of this environmental report supplement. Some of the individuals listed below prepared specific sections in accordance with their technical qualifications. Others provided input to those sections through in-depth review and data validation, while others provided overall technical and management reviews for their respective organization.

NAME:	**YVONNE F. ABERNETHY**
AFFILIATION:	CPG Environmental Services
EDUCATION:	M.S. Forest Economics and Management, Louisiana State University, 1984 B.S., Forest Management, University of Tennessee, 1979
TECHNICAL EXPERIENCE:	27 years experience in environmental impact and regulatory compliance analysis; NEPA analysis and documentation; natural resources management, cultural resource assessment, and socioeconomic analysis, including 9 years of experience supporting natural resource and cultural resource management for Calvert Cliffs. Cultural and natural resource lead for the proposed Calvert Cliffs Unit 3.
ERS RESPONSIBILITY:	Prepared land use, ecology, and socioeconomic sections.
NAME:	**JULEA B. HOVEY**
AFFILIATION:	CPG Environmental Services
EDUCATION:	M.S.P.H., Environmental Science, University of South Carolina, 1996 B.S., Biology, Georgia Southern University, 1987
TECHNICAL EXPERIENCE:	22 years experience in environmental management programs, environmental impact and regulatory compliance analysis; NEPA analysis and documentation. 16 years experience providing environmental technical support to Calvert Cliffs including serving as the License Renewal Calvert Cliffs Units 1 and 2 Environmental Project Manager and advisor to the Calvert Cliffs Unit 3 Environmental Review and Permitting.
ERS RESPONSIBILITY:	ERS lead; prepared Chapters 1, 2, 5, 6, 7 and 8 and transportation, air quality, noise, and public and occupational health sections.
NAME:	**GREGORY C. DECAMP**
AFFILIATION:	CPG Environmental Services
EDUCATION:	M.S., Biology, Bowling Green State University B.S., Chemistry, Xavier University
TECHNICAL EXPERIENCE:	36 years experience in NEPA analysis and documentation; ecological studies and assessments; RCRA compliance; facility siting and alternatives analysis; environmental permitting and regulatory analysis. Experience includes conduct of environmental reviews and development of licensing documentation for five nuclear power plant license renewal projects and two power uprate projects.
ERS RESPONSIBILITY:	Technical and editorial review.

NAME:	**BRENDA NUSE**
AFFILIATION:	Calvert Cliffs Licensing Department
EDUCATION:	B.S., Biology, St. Mary's College of Maryland
TECHNICAL EXPERIENCE:	29 years experience at Calvert Cliffs responsible for environmental management, environmental regulatory analysis and permitting, including 18 years experience in wildlife habitat and natural resource management.
ERS RESPONSIBILITY:	Technical review and validation.

ENCLOSURES

1. Letter from Mr. D. E. Lauver (CCNPP) to Mr. L. Miranda (FWS), dated June 3, 2010, Nuclear Regulatory Commission Informal Consultation Preparation

2. Letter from Mr. D. E. Lauver (CCNPP) to Ms. Lori Byrne (MDNR), dated June 3, 2010, Environmental Review

3. Letter from Mr. D. E. Lauver (CCNPP) to Mr. J. R. Little (MD of Planning), dated June 3, 2010, Nuclear Regulatory Commission Informal Consultation Preparation

4. Letter from Mr. D. E. Lauver (CCNPP) to Mr. E. Ghigiarelli (MDOE), dated June 15, 2010, Coastal Zone Management Consistency Determination

Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

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CALVERT CLIFFS
NUCLEAR POWER PLANT

June 3, 2010

U.S. Fish and Wildlife Services
Chesapeake Bay Field Office
177 Admiral Cochrane Drive
Annapolis, MD 21401

ATTENTION: Mr. Leopoldo Miranda

SUBJECT: Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation
Nuclear Regulatory Commission Informal Consultation Preparation

Calvert Cliffs Nuclear Power Plant, LLC is preparing an application to renew the operating license for the Calvert Cliffs Nuclear Power Plant (CCNPP) Independent Spent Fuel Storage Installation (ISFSI), and we intend the application to be consistent with your agency's interests and the priorities of our community. As part of the license renewal process, the Nuclear Regulatory Commission requires the applicant identify adverse impacts to threatened and endangered species resulting from continued operation of the facility.

Calvert Cliffs believes that continued operation of the CCNPP ISFSI will have no adverse impact on any protected species. There are no planned operational or refurbishment activities for the period of extended operations that would invalidate this conclusion. The Nuclear Regulatory Commission may request an informal consultation with your agency on this matter.

To assist you in making your determination, figures are enclosed which depict the CCNPP ISFSI location and the associated vicinity. The proposed action involves the renewal of operating license SNM-2505 for an additional 40 years of operation. The current operating license expires November 30, 2012. The CCNPP ISFSI is collocated with the CCNPP Units 1 and 2 generating facility, and began operation in 1992. The six-acre facility provides temporary storage of spent fuel generated by the operation of Units 1 and 2. Calvert Cliffs intends to continue to utilize the facility for temporary storage until a federal repository is available and the U.S. Department of Energy takes possession of the fuel for permanent disposal, as provided in the Nuclear Waste Policy Act of 1982.

As a result of an aging management review of the facilities systems, structures, and components, as well as a review of time limited analyses associated with ISFSI operations, continued operations will not involve any changes to operations or refurbishment to the facility.

It is our intent that, by contacting you at this point in the process, we can identify any concerns or data needed so that those areas identified can be addressed to ensure that the consultation process proceeds smoothly and efficiently. We would appreciate it if you would provide your comments and any additional information or action that might be required from Calvert Cliffs to expedite the upcoming consultation process.

Should you have questions regarding this matter, please contact me at (410) 495-5219 or Ms. Brenda D. Nuse at (410) 495-4913.

Very truly yours,

Douglas E. Lauver
Director – Licensing

DEL/RDW/bjd

Enclosure: Location of the Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation and Surrounding Area

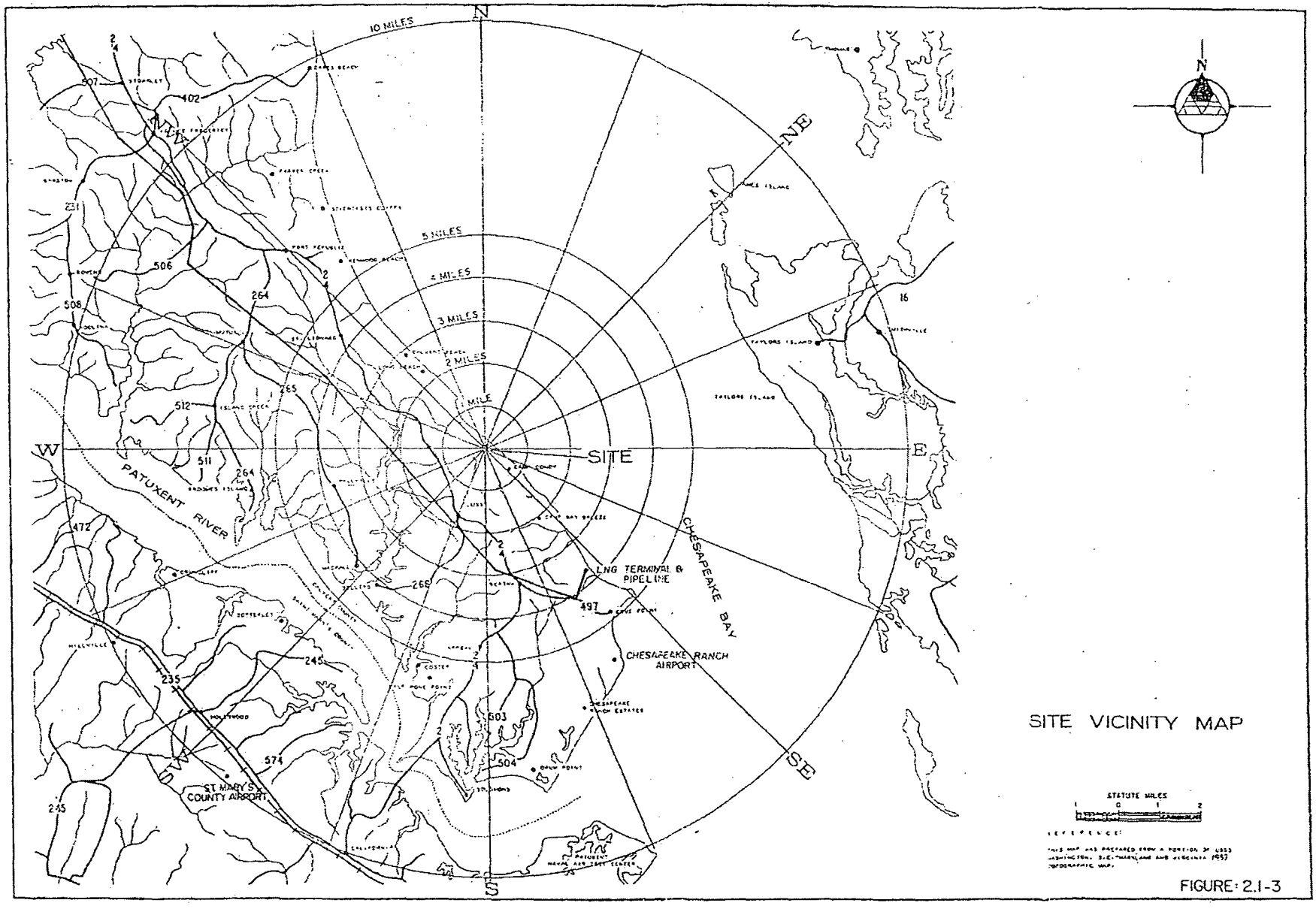
cc: Document Control Desk, NRC

ENCLOSURE

Location of the Calvert Cliffs Nuclear Power Plant

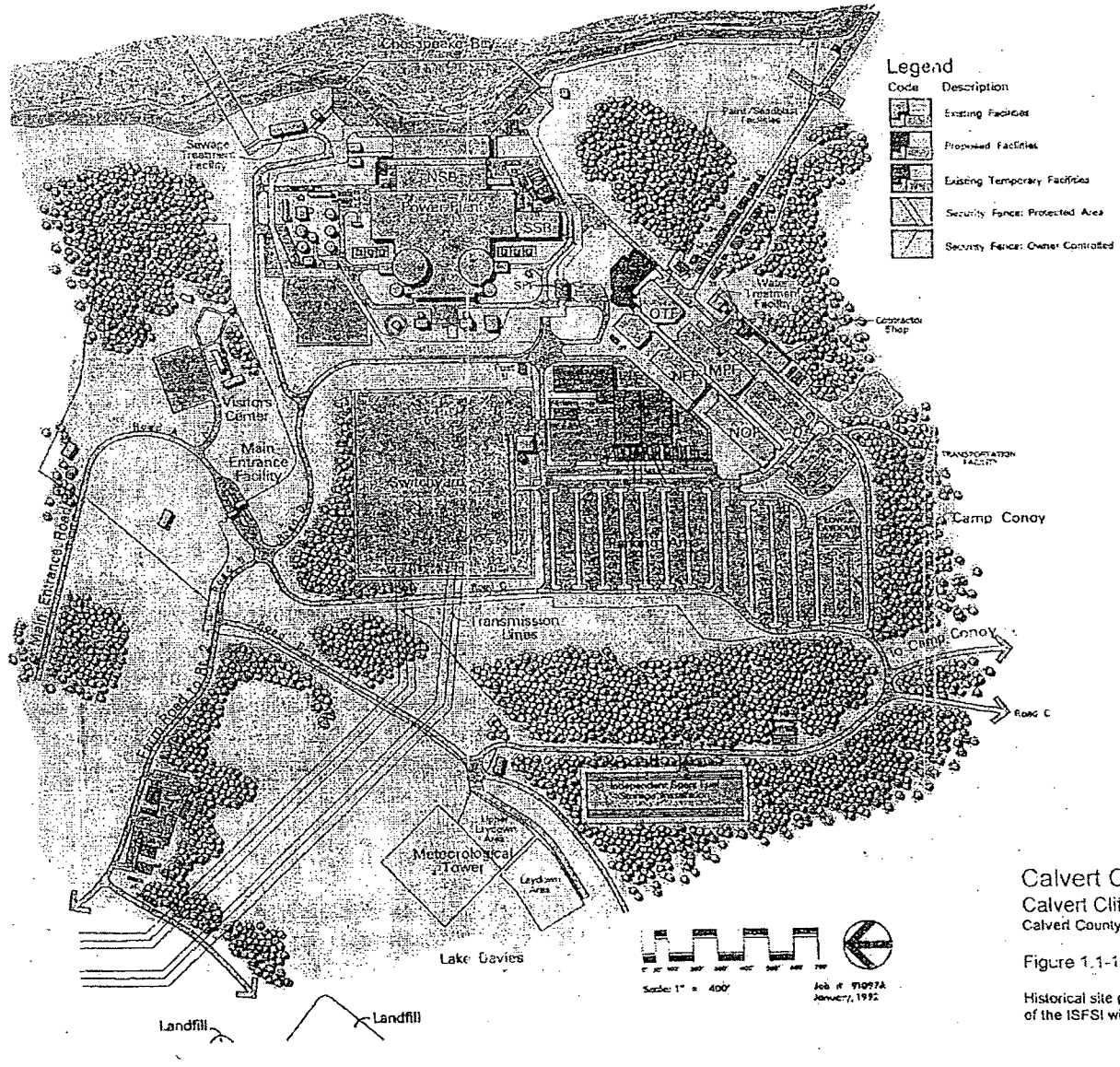
Independent Spent Fuel Storage Installation and Surrounding Area

Calvert Cliffs Nuclear Power Plant, LLC
June 3, 2010



SITE VICINITY MAP

FIGURE: 2.1-3



Calvert Cliffs Site Plan

Calvert Cliffs Nuclear Power Plant

Calvert County, Maryland

Figure 1.1-1

Historical site plan showing the relative location of the ISFSI with respect to the power plant.

Revision 3

Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

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CALVERT CLIFFS
NUCLEAR POWER PLANT

June 3, 2010

Maryland Department of Natural Resources
Wildlife and Heritage Service
Tawes State Office Building, E-1
580 Taylor Avenue
Annapolis, MD 21401

ATTENTION: Ms. Lori Byrne

SUBJECT: Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation
Environmental Review

Calvert Cliffs Nuclear Power Plant, LLC requests an environmental review in support of the Section 7 Consultation under the Endangered Species Act be performed by your office. Calvert Cliffs is preparing an application to renew the operating license for the Calvert Cliffs Nuclear Power Plant (CCNPP) Independent Spent Fuel Storage Installation (ISFSI) for an additional 40 years. The current operating license expires November 30, 2012. The facility is collocated with the CCNPP Units 1 and 2 generating facility near Lusby, Calvert County Maryland, and began operation in 1992. The six-acre facility provides temporary storage of spent fuel generated by the operation of Units 1 and 2 and is situated approximately 3,000 feet west of the Chesapeake Bay shoreline on the CCNPP site. Figures are enclosed, which depict the CCNPP ISFSI location and the associated vicinity.

Calvert Cliffs intends to continue to utilize the facility for temporary storage until a federal repository is available and the U.S. Department of Energy takes possession of the fuel for permanent disposal, as provided in the Nuclear Waste Policy Act of 1982. As a result of an aging management review of the facilities systems, structures, and components, as well as a review of time limited analyses associated with ISFSI operations, continued operations will not involve any changes to operations or refurbishment to the facility. As part of the license renewal process, the Nuclear Regulatory Commission requires the applicant identify adverse impacts to threatened and endangered species resulting from continued operation of the facility. It is our intent, that by contacting you at this point in the process, we can identify any concerns or data needed to support your environmental review.

Should you have questions regarding this matter, please contact me at (410) 495-5219 or Ms. Brenda D. Nuse at (410) 495-4913.

Very truly yours,

Douglas E. Lauver
Director – Licensing

DEL/RDW/bjd

Enclosure: Location of the Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation and Surrounding Area

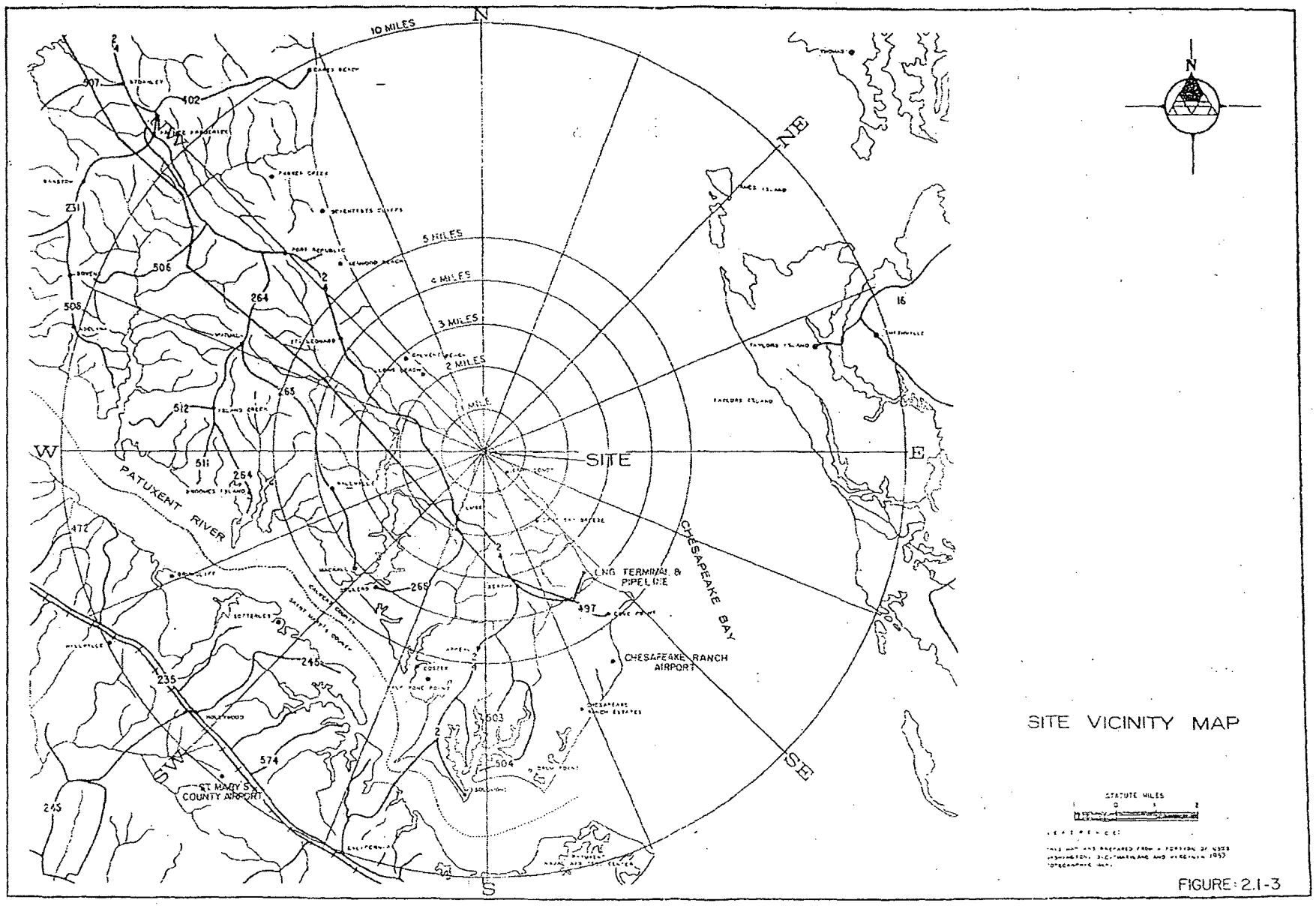
cc: Document Control Desk, NRC

ENCLOSURE

Location of the Calvert Cliffs Nuclear Power Plant

Independent Spent Fuel Storage Installation and Surrounding Area

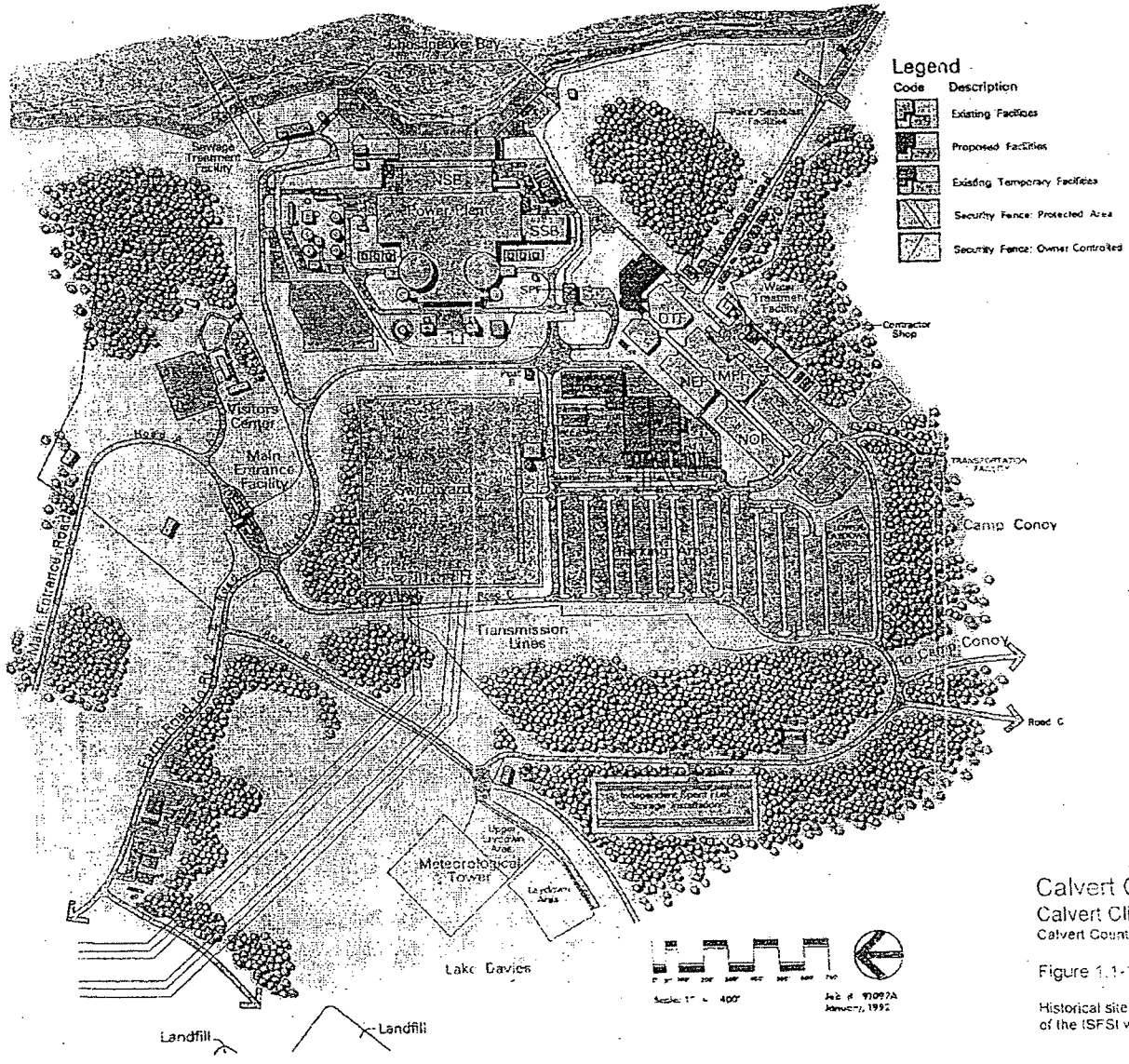
Calvert Cliffs Nuclear Power Plant, LLC
June 3, 2010



SITE VICINITY MAP

STATUTE MILES

FIGURE: 2.1-3



Calvert Cliffs Site Plan

Calvert Cliffs Nuclear Power Plant

Calvert County, Maryland

Figure 1.1-1

Historical site plan showing the relative location of the ISFSI with respect to the power plant.

Revision 3

Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

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CALVERT CLIFFS
NUCLEAR POWER PLANT

June 3, 2010

Maryland Department of Planning
Maryland Historical Trust
100 Community Place
Crownsville, MD 21032

ATTENTION: Mr. J. Rodney Little

SUBJECT: Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation
Nuclear Regulatory Commission Informal Consultation Preparation

Calvert Cliffs Nuclear Power Plant, LLC is preparing an application to renew the operating license for the Calvert Cliffs Nuclear Power Plant (CCNPP) Independent Spent Fuel Storage Installation (ISFSI), and we intend the application to be consistent with your agency's interests and the priorities of our community. As part of the license renewal process, the Nuclear Regulatory Commission requires the applicant identify adverse impacts to cultural resources resulting from continued operation of the facility.

Calvert Cliffs believes that continued operation of the CCNPP ISFSI will have no adverse impact on any cultural resources. There are no planned operational or refurbishment activities for the period of extended operations that would invalidate this conclusion. The Nuclear Regulatory Commission may request an informal consultation with your agency on this matter.

To assist you in conducting your evaluation, figures are enclosed which depict the CCNPP ISFSI location and the associated vicinity. The proposed action involves the renewal of operating license SNM-2505 for an additional 40 years of operation. The current operating license expires November 30, 2012. The CCNPP ISFSI is collocated with the CCNPP Units 1 and 2 generating facility, and began operation in 1992. The six-acre facility provides temporary storage of spent fuel generated by the operation of Units 1 and 2. Calvert Cliffs intends to continue to utilize the facility for temporary storage until a federal repository is available and the U.S. Department of Energy takes possession of the fuel for permanent disposal, as provided in the Nuclear Waste Policy Act of 1982.

As a result of an aging management review of the facilities systems, structures, and components, as well as a review of time limited analyses associated with ISFSI operations, continued operations will not involve any changes to operations or refurbishment to the facility.

It is our intent that, by contacting you at this point in the process, we can identify any concerns or data needed so that those areas identified can be addressed to ensure that the consultation process proceeds smoothly and efficiently. We would appreciate it if you would provide your comments and any additional information or action that might be required from Calvert Cliffs to expedite the upcoming consultation process.

Should you have questions regarding this matter, please contact me at (410) 495-5219 or Ms. Brenda D. Nuse at (410) 495-4913.

Very truly yours,

Douglas E. Lauver
Director – Licensing

DEL/RDW/bjd

Enclosure: Location of the Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation and Surrounding Area

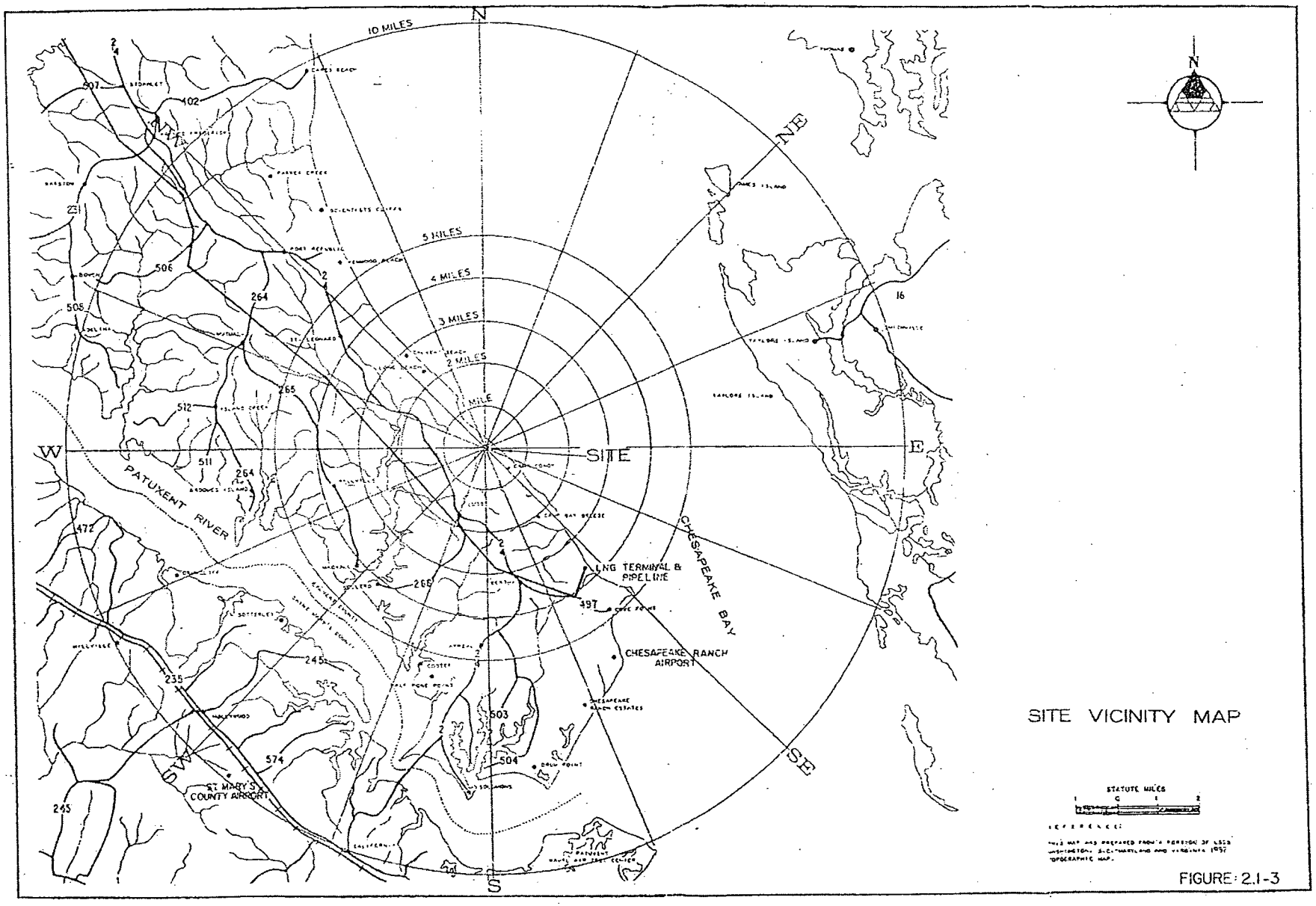
cc: Document Control Desk, NRC

ENCLOSURE

Location of the Calvert Cliffs Nuclear Power Plant

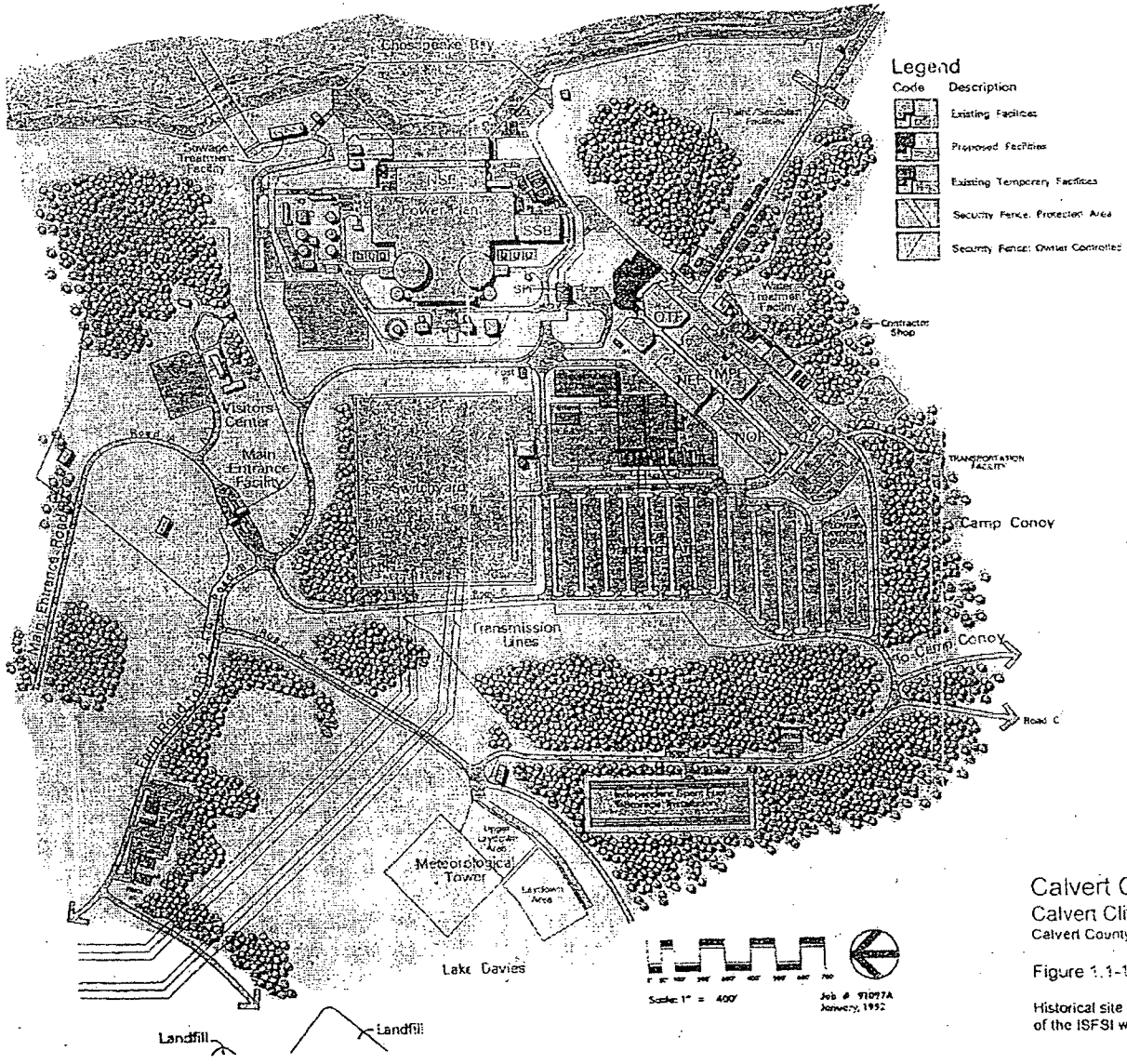
Independent Spent Fuel Storage Installation and Surrounding Area

Calvert Cliffs Nuclear Power Plant, LLC
June 3, 2010



SITE VICINITY MAP

FIGURE: 2.1-3



Calvert Cliffs Site Plan

Calvert Cliffs Nuclear Power Plant

Calvert County, Maryland

Figure 1.1-1

Historical site plan showing the relative location of the ISFSI with respect to the power plant.

Revision 3

Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
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a joint venture of
Constellation Energy EDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

June 15, 2010

Maryland Department of the Environment
Wetlands and Waterways Program
1800 Washington Boulevard
Baltimore, MD 21230

ATTENTION: Mr. Elder Ghigiarelli

SUBJECT: Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation
Coastal Zone Management Consistency Determination

Calvert Cliffs Nuclear Power Plant, LLC is preparing an application to renew the U.S. Nuclear Regulatory Commission (NRC) operating license for the Calvert Cliffs Nuclear Power Plant (CCNPP) Independent Spent Fuel Storage Installation (ISFSI), and has performed a review for consistency with the Maryland Coastal Zone Management Program. Calvert Cliffs is requesting concurrence that the license renewal action described below is consistent with the Maryland Coastal Zone Management Program Consistency Certification presented in Attachment (1).

In accordance with NRC regulations for license renewal (10 CFR Part 72), Calvert Cliffs will include an Environmental Report Supplement with its license renewal application. In conjunction with the application to the NRC, Calvert Cliffs must also submit a certification to the federal licensing agency (NRC) and the state pursuant to the requirements of the Coastal Zone Management Act. The Environmental Report Supplement will include a description of the proposed action and provide updates to the information and analyses included in the original environmental report. The report will also include a complete list of licenses, permits, and other approvals from federal, state, and local authorities for current CCNPP ISFSI operations, as well as approvals and consultations that would be required for the approval of the license for extended operation. We will provide you with a copy of the entire application when it is submitted to the NRC.

The proposed action involves the license renewal of operating license SNM-2505 for an additional 40 years of operation. The current operating license expires November 30, 2012. The CCNPP ISFSI is collocated with the CCNPP Units 1 and 2 generating facility, and began operation in 1992. The six-acre facility provides temporary storage of spent fuel generated by the operation of Units 1 and 2, and is

situated approximately 3,000 feet west of the Chesapeake Bay shoreline on the CCNPP site. Calvert Cliffs intends to continue to utilize the facility for temporary storage until a federal repository is available and the U.S. Department of Energy takes possession of the fuel for permanent disposal, as provided in the Nuclear Waste Policy Act of 1982. Figures are enclosed which depict the CCNPP ISFSI location and the associated vicinity.

As a result of an aging management review of the facilities systems, structures, and components, as well as a review of time limited analyses associated with ISFSI operations, continued operations will not involve any changes to operations or refurbishment to the facility. Calvert Cliffs also reviewed the information and analyses included in the original Environmental Report submitted with the initial license application to the NRC in 1990. The results of this review indicate that the operation of the ISFSI will continue to have small, if any, impact on the environment; and therefore, impacts to the coastal zone will remain small. Calvert Cliffs ISFSI license renewal and continued operation and its effects, are all consistent with the Maryland Coastal Zone Management Program.

Calvert Cliffs has initiated correspondence with agencies to support NRC's consultations under the Endangered Species Act and the Historic Preservation Act.

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219 or Ms. Brenda D. Nuse at (410) 495-4913.

Very truly yours,

Douglas E. Lauver
Director – Licensing

DEL/RDW/bjd

Attachment: (1) Maryland Coastal Zone Management Consistency Certification

Enclosure: Location of the Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation and Surrounding Area

ATTACHMENT (1)

MARYLAND COASTAL ZONE MANAGEMENT CONSISTENCY CERTIFICATION

Calvert Cliffs Nuclear Power Plant, LLC
June 15, 2010

ATTACHMENT (1)

MARYLAND COASTAL ZONE MANAGEMENT CONSISTENCY CERTIFICATION

Calvert Cliffs Nuclear Power Plant, LLC hereby certifies that the CCNPP ISFSI license renewal is consistent with the Maryland Coastal Zone Management Program.

The regulation [15 CFR 930.63(a)] provides the State six months from the receipt of this letter in which to concur or object to the Calvert Cliffs certification. However, the regulation [15 CFR 930.62(b)] also provides that if the State has not issued a decision within three months following the commencement of state agency review, it shall notify the contacts listed below of the status of the matter and the basis for further delay. The State's concurrence, objections, or notifications of review status should be sent to the following contacts:

Mr. John M. Goshen
Division of Spent Fuel Storage and Transportation
U.S. Nuclear Regulatory Commission
Mail Stop E3 D2M
Washington, DC 20555
301-492-3325

George H. Gellrich, Vice President
Calvert Cliffs Nuclear Power Plant, LLC
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
(410) 495-5200

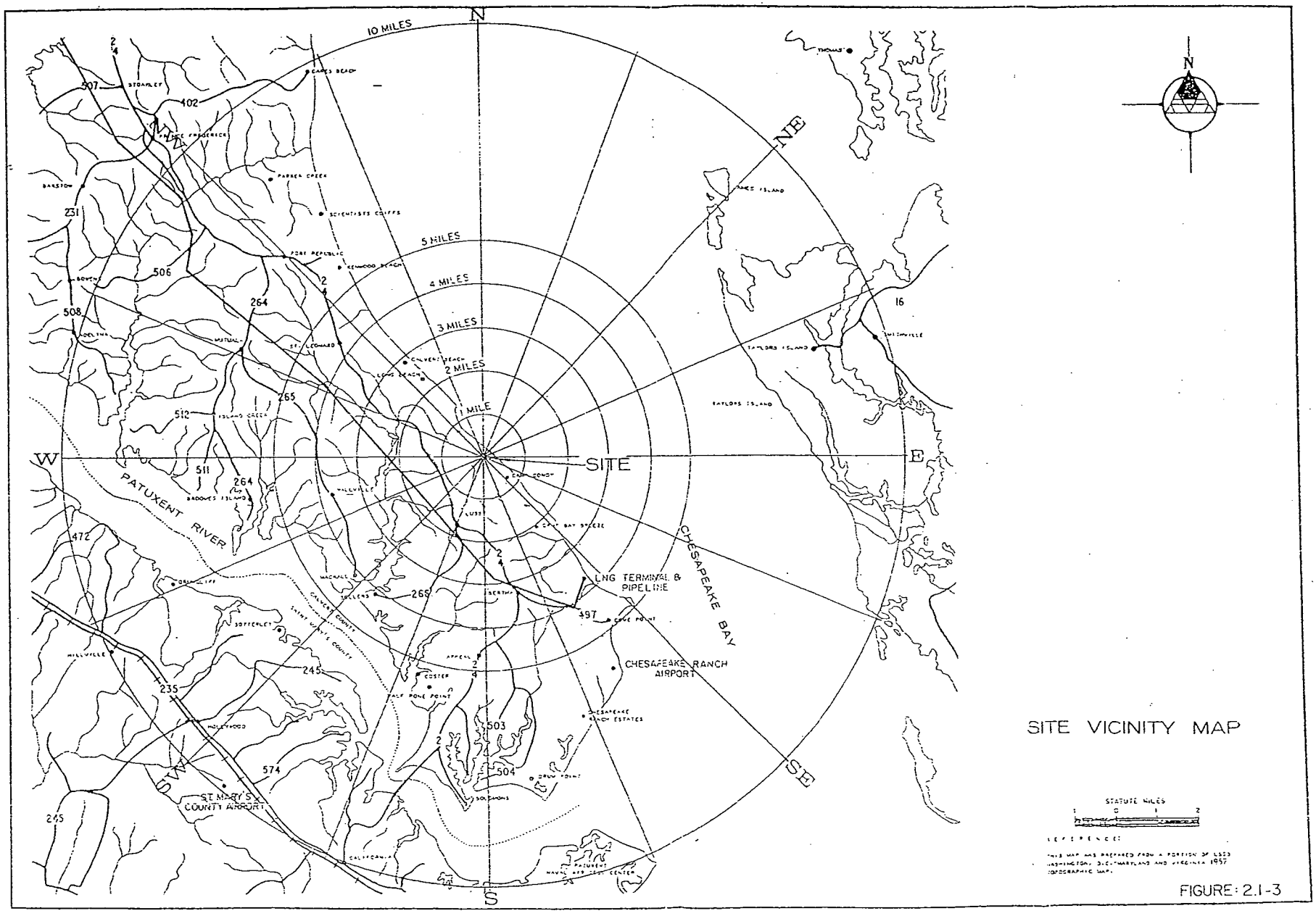
George H. Gellrich, Vice President
Calvert Cliffs Nuclear Power Plant, LLC

ENCLOSURE

Location of the Calvert Cliffs Nuclear Power Plant,

Independent Spent Fuel Storage Installation and Surrounding Area

Calvert Cliffs Nuclear Power Plant, LLC
June 15, 2010



SITE VICINITY MAP

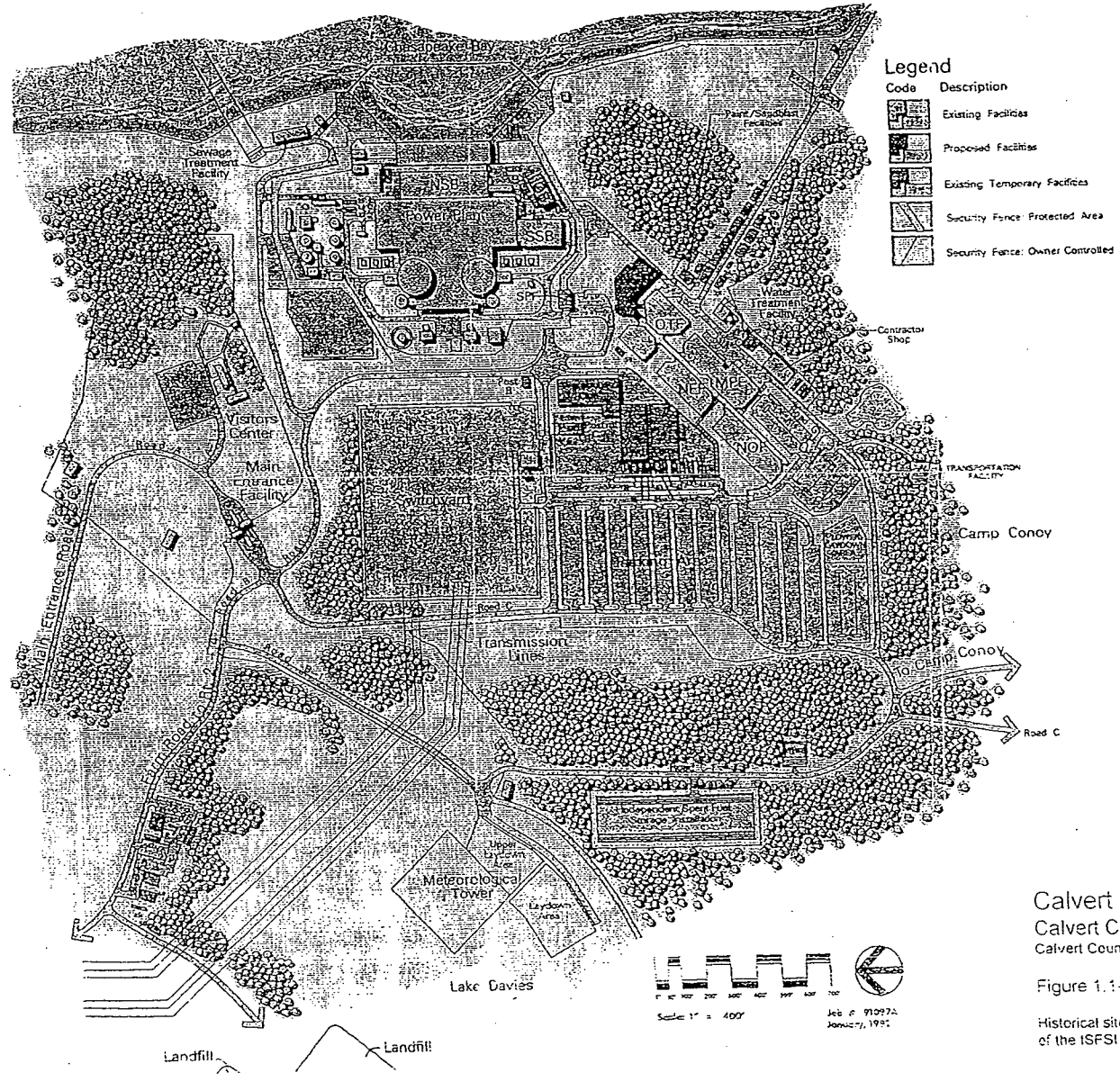
STATUTE MILES

0 1 2

REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS WASHINGTON, D.C.-MARYLAND AND VIRGINIA 1957 TOPOGRAPHIC MAP.

FIGURE: 2.1-3



Calvert Cliffs Site Plan

Calvert Cliffs Nuclear Power Plant

Calvert County, Maryland

Figure 1.1-1

Historical site plan showing the relative location of the ISFSI with respect to the power plant.

Revision 3

Appendix F

Additional Information

F1.0 ADDITIONAL INFORMATION

F1.1 TRAINING AND QUALIFICATIONS (10 CFR 72.28)

F1.1.1 Technical Qualifications

Calvert Cliffs Nuclear Power Plant, LLC (Calvert Cliffs) has been involved in the generation of electricity through the use of nuclear fuels since the 1970s. Reactor operation and the resulting generation of spent fuel, has provided Calvert Cliffs with extensive experience in the receipt, handling and storage of nuclear fuel. Operation of Independent Spent Fuel Storage Installation (ISFSI) is conducted by the operating staff of the Calvert Cliffs operating units. Calvert Cliffs has safely operated the ISFSI since 1992.

A discussion of the technical qualifications was submitted with the initial ISFSI license application for the ISFSI (Reference F1.4.1), in Chapter 2. Additionally, discussion of personnel qualifications can also be found in Section 9.1.3 of the ISFSI Updated Safety Analysis Report (Reference F1.4.2).

In Chapter 2 of Calvert Cliffs' initial ISFSI license application (Reference F1.4.1), a commitment to staff and operate the ISFSI with an adequate number of personnel who possessed the necessary skills to safely run the ISFSI facility was made. Calvert Cliffs remains committed to maintain skilled personnel in order to provide continued assurance for the safe operation of the ISFSI throughout the ISFSI license renewal period.

F1.1.2 Personnel Training

A discussion of training for personnel involved with the ISFSI was submitted in Reference F1.4-1, Section 7. Additionally, discussion of the training programs is provided in Section 9.3 of Reference F1.4.2 and in Section 12.2 of Calvert Cliffs Updated Final Safety Analysis Report (Reference F1.4.3). Both of these documents are updated in accordance with existing regulatory requirements. Calvert Cliffs remains committed to ensure adequate training will be conducted in order to provide continued assurance for the safe operation of the ISFSI throughout the ISFSI license renewal period.

F1.1.3 Operating Organization

Operation of the ISFSI is integrated with operation of the Calvert Cliffs operating units. A description of the operating organization at Calvert Cliffs is contained in Section 12.1 of Reference F1.4.3. This document is updated when changes are made in accordance with existing regulatory requirements.

F1.2 FINANCIAL ASSURANCE FOR DECOMMISSIONING (10 CFR 72.30)

In accordance with the requirements of 10 CFR 72.30, a decommissioning funding plan for the ISFSI was submitted to the Nuclear Regulatory Commission in Reference F1.4.4. A revised plan was subsequently submitted (Reference F1.4.5). These documents contained a Decommissioning Trust Agreement which is a summary of expected decommissioning costs for the ISFSI and a schedule of planned contribution rates to the external sinking fund for ISFSI decommissioning. This ISFSI license renewal application does not alter the basic elements of the decommissioning plan. The actual activities at the time of decommissioning of the ISFSI will be dependent upon the regulations and practices in place at that time.

As part of the ISFSI decommissioning funding plan Calvert Cliffs committed to updating both the estimated decommissioning costs and the annual deposits made to the decommissioning fund on a periodic basis. This practice will continue to be performed throughout the ISFSI license renewal period. These steps will help provide reasonable assurance of adequate decommissioning funding at the permanent termination of ISFSI operations.

F1.3 EMERGENCY PLANNING (10 CFR 72.32)

A description of emergency planning was submitted in Reference F.1.4.1, Section 9.5. The site's Emergency Response Plan was determined to be adequate for events which might occur involving the ISFSI. The site's Emergency Response Plan has been prepared in accordance with 10 CFR 50.47 and therefore satisfies the requirements of 10 CFR 72.32. There are no specific changes to the site's Emergency Response Plan as a result of this ISFSI license renewal application.

Emergency planning for the ISFSI is integrated into the emergency plan for the Calvert Cliffs operating units. As such the emergency plan and its implementing procedures are updated as required in accordance with 10 CFR Part 50, Appendix E; 10 CFR 50.54(q); and 10 CFR 72.44(f).

F1.4 REFERENCES

F1.4.1 Letter from G. C. Creel (BGE) to Director, Division of Industrial and Medical Nuclear Safety (NRC), dated December 21, 1989, Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) Application

F1.4.2 Calvert Cliffs Independent Spent Fuel Storage Installation Updated Safety Analysis Report, Revision 18, September 15, 2009

F1.4.3 Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report, Revision 41, September 15, 2009

F1.4.4 Letter from G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated July 26, 1990, Decommissioning Plan for Calvert Cliffs ISFSI

F1.4.5 Letter from G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated August 18, 1992, Revision to the ISFSI Decommissioning Plan

ATTACHMENT (2)

CALVERT CLIFFS INDEPENDENT SPENT FUEL STORAGE INSTALLATION REQUEST FOR EXEMPTION FROM 10 CFR 72.42(a)

ATTACHMENT (2)

CALVERT CLIFFS INDEPENDENT SPENT FUEL STORAGE INSTALLATION REQUEST FOR EXEMPTION FROM 10 CFR 72.42(a)

In accordance with the provisions of 10 CFR 72.7, Calvert Cliffs Nuclear Power Plant, LLC (Calvert Cliffs) requests an exemption from 10 CFR 72.42(a) in regards to the license renewal period for our Independent Spent Fuel Storage Installation (ISFSI). Specifically, 10 CFR 72.42(a) states that the license term for an ISFSI must not exceed 20 years. Calvert Cliffs is requesting a license renewal period of 40 years.

Irradiated fuel storage for both Units at Calvert Cliffs is maintained at the Calvert Cliffs spent fuel pool, which is located in the site's Auxiliary Building, and the ISFSI. Given the limited space for storage of irradiated fuel in the spent fuel pool, renewal of the ISFSI license is necessary to support the continued operation of both Units during their renewed operating license periods.

In addition, storage of irradiated fuel at the ISFSI will continue to be necessary since the Department of Energy (DOE) has not begun, and currently has not identified any viable solution for being able to take irradiated fuel as it is required to do under the provisions of the Nuclear Waste Policy Act. This failure of the DOE to fulfill its obligations, combined with the renewed license period for Calvert Cliffs operating Units, makes it necessary that the ISFSI license renewal period factor in these considerations.

Submittal of a license renewal period of 20 years as allowed by 10 CFR 72.42(a) would mean Calvert Cliffs must submit a second license renewal application for the ISFSI before the expiration of the renewed license periods for Calvert Cliffs operating Units. This effort would create an unnecessary expenditure of Calvert Cliffs and Nuclear Regulatory Commission (NRC) resources.

It should further be noted that NRC staff has proposed and has received the Commission's approval to change 10 CFR 72.42(a) to allow an ISFSI license term to be extended up to a 40 year period. However, since this change has not been implemented as at this time, a request for an exemption from 10 CFR 72.42(a) is necessary.

This request for an exemption of the license renewal period is based on the need for a longer license renewal period. It is supported by the technical justification presented in Attachment (1) of this submittal that demonstrates the ability of ISFSI to safely perform its intended functions for a 40 year license renewal period.

Need for a 40 Year License Renewal Period

The current Renewed Facility Operating License for Calvert Cliffs Unit Nos. 1 and 2 expire on July 31, 2034 and August 13, 2036, respectively.

The current Calvert Cliffs ISFSI Materials license (SNM-2505) expires on November 30, 2012. Submittal of a license renewal for a period of 20 years would result in an expiration date of November 30, 2032 which is several years before the expiration of the operating Units renewed license period. This situation would require Calvert Cliffs to submit a second ISFSI license renewal to cover the remainder of the operating Units renewed license periods.

It is uncertain as to when the DOE will begin to accept irradiated fuel. With the current uncertainty concerning the Yucca mountain repository license application and the fact that no other potential site or method has been identified for the removal of irradiated fuel from the site, it is likely there will be a need for Calvert Cliffs to continue to operate the ISFSI for an indeterminate period of time following the expiration of the operating Units renewed license period. Even after the DOE is ready to accept irradiated

fuel there is a length of time that will be needed to carry out the process of shipping the irradiated fuel off-site. Until all irradiated fuel is removed from the site it is necessary to maintain the ISFSI license.

As previously discussed, this requested ISFSI 40 year renewal period goes beyond the renewed operating license period for both Unit 1 and Unit 2. Should Calvert Cliffs decide not to pursue a further operating extension for both Unit 1 and Unit 2, it is recognized we would still be required to maintain the operation of the spent fuel pool and its associated support components for the remaining duration of the ISFSI renewal period. Title 10 CFR 50.54(bb) requires a licensee to "submit written notification to the Commission for its review and preliminary approval of the program by which the licensee intends to manage and provide funding for the management of all irradiated fuel at the reactor following permanent cessation of operation of the reactor until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository." This requirement will ensure components covered under 10 CFR Part 50 which support the site's ISFSI operation will continue to be appropriately maintained for the remaining duration of the ISFSI renewal period.

Based on the above reasons, Calvert Cliffs requests an ISFSI license renewal period exemption. This license renewal request for 40 years would extend the ISFSI license renewal period through November 30, 2052.

Technical Justification

The technical justification that the ISFSI will be able to fulfill its safety functions over a license renewal period of 40 years is provided in the application for renewed ISFSI license, which is contained in Attachment (1) of this submittal. The ISFSI license renewal application addresses the applicable provisions of 10 CFR Part 72, Subpart B, as required by 10 CFR 72.42(b). The systems, structures, and components (SSCs) that are within the scope of ISFSI license renewal, and the required evaluations are identified. Aging management reviews identify the SSC materials and environments to which these SSCs are exposed, as well as any aging effects requiring management. Time-limited aging analyses included in the current licensing basis are identified and revised for SSCs within the ISFSI license renewal scope. Aging management activities are identified that provide reasonable assurance that SSCs within the scope of ISFSI license renewal will continue to perform their intended functions consistent with the current licensing basis for the renewal period.

Requirements of 10 CFR 72.7

The specific requirements for granting an exemption from 10 CFR Part 72 regulations are set forth in 10 CFR 72.7. Under 10 CFR 72.7, the NRC is authorized to grant an exemption upon demonstration that the exemption: (i) is authorized by law, (ii) will not endanger life or property or the common defense and security, and (iii) is in the public interest. These requirements are met as indicated below:

A. The Exemption Request is Authorized by Law

The NRCs authority to grant an exemption from 10 CFR Part 72 is established by law as discussed in 10 CFR 72.7. During the license renewal period Calvert Cliffs will maintain and operate the ISFSI such that it is able to safely perform all required safety functions. Therefore, granting this exemption is explicitly authorized by NRC regulations.

B. The Exemption Request Will Not Endanger Life or Property or the Common Defense and Security

A 40 year license renewal period has been technically evaluated in the ISFSI license renewal application which is contained in Attachment (1) of this submittal. This application determines that

there is reasonable assurance that SSCs within the scope of license renewal will continue to perform their intended safety functions for the duration of the license renewal period. Continued operation of the ISFSI does not endanger life or property as discussed in the Environmental Report Supplement, which is included as Appendix E to the ISFSI license renewal application.

The common defense and security of the United States is also not endangered by the 40 year license renewal of the ISFSI as Calvert Cliffs will continue to provide all necessary security measures. A 40 year ISFSI license renewal period will allow for the continued operation of the Calvert Cliffs operating Units for the duration of their renewed operating license period. The continued safe operation of the operating Units enhances the common defense and security of the United States by helping to provide reliable, low cost electricity to meet energy demands. This 40 year ISFSI license renewal period will also provide additional time for continued safe storage of irradiated fuel here at Calvert Cliffs while the DOE develops the capability to accept irradiated fuel.

C. The Exemption is in the Public Interest

The exemption to allow a 40 year ISFSI license renewal period would provide Calvert Cliffs the ability for continued safe storage of irradiated fuel at the ISFSI throughout the renewed operating license period without the need to repeat the ISFSI license renewal process. The granting of this exemption would therefore conserve both Calvert Cliffs and NRC resources and allow these resources to be focused where needed to maintain high standards of nuclear, industrial and personnel safety.

Conclusion

The requested exemption for a 40 year ISFSI license renewal period vice the 20 year period contained in 10 CFR 72.42(a) has been evaluated to have no adverse impact on safety. As demonstrated in the ISFSI license renewal application [Attachment (1) of this submittal], monitoring activities will provide reasonable assurance that SSCs within the scope of license renewal will continue to perform their intended functions throughout the 40 year ISFSI license renewal period.

Therefore because the requested exemption for the ISFSI license renewal period is authorized by law, will not endanger life or property or the common defense and security, is in the public interest, and is requested for good cause, Calvert Cliffs requests that, in accordance with the provisions of 10 CFR 72.7, the NRC grant this requested exemption.