Omaha Public Power District

10 CFR 50.54(q)
10 CFR 72.212(b)(10)
10 CFR 50.90
10 CFR 72.32
10 CFR 50.47(b)

LIC-19-0001
February 28, 2019

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

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

Fort Calhoun Station
Independent Spent Fuel Storage Installation
NRC Docket No. 72-054

Subject: **License Amendment Request (LAR) 19-01: Independent Spent Fuel Storage Installation (ISFSI) Emergency Plan and Emergency Action Level Scheme**

References:

1. OPPD Letter (S. Marik) to USNRC (Document Control Desk) –"License Amendment Request 16-07; Revise the Fort Calhoun Station Emergency Plan to the Permanently Defueled Emergency Plan and Permanently Defueled Emergency Action Level Scheme," dated December 16, 2016 (LIC-16-0108) (ML16351A464)

2. OPPD Letter (T. Burke) to USNRC (Document Control Desk) – "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel," dated November 13, 2016 (LIC-16-0074) (ML16319A254)

3. Letter USNRC (J. Kim) to OPPD (M. Fisher) – "Fort Calhoun Station, Unit No. 1, Post-Shutdown Decommissioning Activities Report", dated March 23, 2017 (LIC-17-0033) (CAC No. 9536) (ML18011A687)

4. Letter USNRC (J. Kim) to OPPD (M. Fisher) – "Fort Calhoun Station, Unit No. 1, Exemptions from Certain Emergency Planning Requirements and Related Safety Evaluation", dated December 11, 2017 (LIC-16-0109) (CAC No. MF9067) (ML17263B198; ML17263B191; ML17278A178)

Pursuant to 10 CFR 50.90, 10 CFR 50.54(q), 10 CFR 50.47(b), and 10 CFR 50, Appendix E, Omaha Public Power District (OPPD) hereby requests an amendment to Renewed Facility License Number DPR-40 for Fort Calhoun Station (FCS). The proposed amendment would replace the FCS Permanently Defueled Emergency Plan (PDEP) (Reference 1) and associated

Emergency Action Level (EAL) technical bases document with the Independent Spent Fuel Storage Installation Only Emergency Plan (IOEP) and its associated Independent Spent Fuel Storage Installation (ISFSI) EAL Technical Bases Document. The IOEP will be used at FCS during the period when all spent fuel is stored in the FCS ISFSI. The proposed changes are being submitted to the NRC for approval prior to implementation, as required under 10 CFR 50.54(q)(4) and 10 CFR Part 50, Appendix E, Section IV.B.2, and 10 CFR 72.44(f).

By letter dated November 13, 2016 (Reference 2), FCS submitted a certification of permanent cessation of power operations and permanent removal of fuel from the reactor vessel. Consequently, as specified in 10 CFR 50.82(a)(2), the station's 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel.

This proposed change reflects the complete removal of all fuel from the spent fuel pool (SFP) and permits specific reductions in the size and makeup of the Emergency Response Organization (ERO) due to the elimination of the remaining design basis accident related to spent fuel handling. The Post-Shutdown Decommissioning Activities Report (PSDAR) (Reference 3) documented OPPD's expectation that all spent fuel would be completely transferred to the ISFSI by the end of 2022. OPPD awarded contracts in the first quarter of 2018 which expedited transferring all spent fuel to the ISFSI by the middle of 2020. To comport to the reduced scope of potential radiological accidents with spent fuel in dry cask storage within the ISFSI, OPPD determined that replacement of the FCS PDEP and EAL Technical Bases Document with the IOEP and the ISFSI EAL Technical Bases Document were warranted.

The proposed IOEP continues to rely on previously requested exemptions (Reference 4) from certain emergency planning requirements as the basis for these exemptions has not changed and remains in effect. The proposed IOEP changes have been determined to represent changes in both the EAL scheme and the staffing level previously requested to implement the PDEP in accordance with the requirements of 10 CFR 50.54(q) and therefore require NRC approval prior to implementation.

Enclosure 1 to this letter contains a description, technical analysis, significant hazards determination, and environmental considerations evaluation for the proposed amendment. Enclosure 1, Attachment 1, contains the supporting evaluations and calculations. Enclosure 1, Attachment 2, contains a comparison matrix of the Proposed FCS Emergency Classification System and ISFSI EALs. Enclosure 1, Attachment 3, contains the ISFSI Only Emergency Plan. Enclosure 1, Attachment 4, contains the ISFSI Emergency Action Level Technical Bases Document.

OPPD requests review and approval of the proposed license amendment by January 31, 2020. Once approved, the Amendment will be implemented within ninety (90) days of OPPD's submittal of a written certification to the NRC that the final spent nuclear fuel assembly has been transferred out of the SFP and placed in storage within the ISFSI.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and OPPD has determined that these changes involve no significant hazards. OPPD has also determined that the proposed changes satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22(c)(9) and do not require an environmental review. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required.

Pursuant to 10 CFR 50.91, "*Notice for public comment; State consultation*," paragraph (b), OPPD is notifying the State of Nebraska of this application for license amendment by transmitting a copy of this letter and supporting attachments to the designated state official.

If you have any questions regarding this transmittal, please contact Mr. Bradley H. Blome – Director – Licensing & Regulatory Assurance at (402) 533-6041.

The proposed changes have been reviewed and approved by the Fort Calhoun Station Plant Operations Review Committee (PORC). This letter contains no new regulatory commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 28, 2019.

Respectfully,

Mary J. Fisher
Vice President Energy Production and Nuclear Decommissioning

MJF/jef/cac

Enclosure 1: Description of Proposed Changes, Technical and Regulatory Evaluation, Significant Hazards Determination, and Environmental Considerations

c: S. A. Morris, NRC Regional Administrator, Region IV
M. C. Layton, NRC Director, Division of Spent Fuel Management
J. D. Parrott, NRC Senior Project Manager
C. D. Steely, NRC Health Physicist, Region IV
Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska Health and Human Services, State of Nebraska

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION

DOCKET NUMBER 50-285 / LICENSE NUMBER DPR-40

ENCLOSURE 1

DESCRIPTION OF PROPOSED CHANGES, TECHNICAL AND REGULATORY EVALUATION, SIGNIFICANT HAZARDS DETERMINATION, AND ENVIRONMENTAL CONSIDERATIONS

DESCRIPTION OF PROPOSED CHANGES, TECHNICAL AND REGULATORY EVALUATION, SIGNIFICANT HAZARDS DETERMINATION, AND ENVIRONMENTAL CONSIDERATIONS

Subject: Independent Spent Fuel Storage Installation Only Emergency Plan (IOEP) and Emergency Action Level (EAL) Scheme

1.0 INTRODUCTION

This evaluation supports a request to amend the Renewed Facility Operating License (OL) DPR-40 for Fort Calhoun Station (FCS).

By letter dated August 25, 2016, OPPD informed the NRC that FCS will permanently cease power operations in accordance with 10 CFR 50.82(a)(1)(i), specifying a shutdown date of October 24, 2016 (Reference 7.1). By letter dated November 13, 2016, FCS submitted a certification of permanent cessation of power operations and permanent removal of fuel from the reactor vessel (Reference 7.2). Consequently, as specified in 10 CFR 50.82(a)(2), the station's 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel.

The proposed IOEP continues to rely on previously requested exemptions from certain emergency planning requirements (Reference 7.3), as the basis for these exemptions has not changed and remains in effect. The proposed IOEP has been determined to represent changes in both the EAL scheme and the staffing level previously requested to implement the Permanently Defueled Emergency Plan (PDEP) (Reference 7.4) in accordance with the requirements of 10 CFR 50.54(q) and therefore, require NRC approval prior to implementation. Additional changes to the FCS PDEP and EAL Technical Bases Document are warranted to reflect the storage of all fuel in the Independent Spent Fuel Storage Installation (ISFSI) facility.

The Post-Shutdown Decommissioning Activities Report (PSDAR) (Reference 7.5) documented OPPD's expectation that all spent fuel would be completely transferred to the ISFSI by the end of 2022. OPPD awarded contracts in the first quarter of 2018 which expedited transferring all spent fuel to the ISFSI by the middle of 2020. To comport to the reduced scope of potential radiological accidents with spent fuel in dry cask storage within the ISFSI, OPPD determined that implementation of the IOEP and the ISFSI EAL Technical Bases Document will be warranted.

The proposed emergency plan is related to the operation of the ISFSI and would be implemented after all spent fuel has been removed from the spent fuel pool (SFP) and placed in dry storage within the ISFSI. Implementation of the IOEP would involve the establishment of administrative controls for radiological source term accumulation limits and methods to control the accidental dispersal of the radiological source.

The NRC approved AREVA TN Americas' Amendment 14, to the standardized NUHOMS Certificate of Compliance (CoC) No. 1004 for Spent Fuel Storage Casks, on April 25, 2017 (Reference 7.6). This revision deleted the License Condition requiring a return to the SFP for inspection. With the approval of the CoC, there is no longer a requirement to return spent fuel to the SFP.

Consistent with the condition that the proposed emergency plan may be implemented ninety (90) days after all spent fuel has been certified to have been removed from the SFP, FCS has submitted a LAR to revise the FCS Facility Operating License to comport to the ISFSI-Only condition that there is no longer a requirement to return spent fuel to the SFP.

2.0 DESCRIPTION

The proposed amendment would modify the FCS license by replacing the existing FCS PDEP and the associated EAL scheme with the IOEP and the ISFSI EAL scheme to reflect the storage of all fuel in the ISFSI. The proposed changes reduce the scope of onsite emergency planning requirements to reflect the reduced scope of potential radiological accidents with all spent fuel in dry cask storage within the ISFSI. After all spent fuel is in dry cask storage within the ISFSI, the number and severity of potential radiological accidents possible at FCS are substantially lower. There continues to be no need for offsite emergency response plans at FCS because no postulated design basis accident or reasonably conceivable beyond design basis accident can result in a radioactive release that exceeds Environmental Protection Agency (EPA) Protective Action Guides (PAGs) beyond the “site boundary”, as described in EPA’s PAG Manual "Protective Action Guides and Planning Guidance for Radiological Incidents" dated January 2017 (EPA PAG Manual) (Reference 7.7).

The robust nature and high integrity of the spent fuel storage system selected for use at the ISFSI is designed to prevent the release of radioactivity in the event of an accident, including environmental phenomena (e.g., earthquake and flooding). As a result of the high integrity dry shielded canister’s design and the substantial protection afforded the canisters, leakage of fission products from a canister is not considered to be a credible event.

The radioactive source term for an accidental release at the defueled reactor site is reduced by radioactive decay and transfer of spent fuel from the SFP to the ISFSI. Potential offsite doses were calculated at FCS to verify that the necessary administrative radiological source term accumulation limits would be adequate during decontamination and dismantling of radioactive systems, structures, and components contained in the non-operational nuclear unit. These administrative radiological source term accumulation limits ensure that if a radiological release were to occur, it would not exceed two times the Offsite Dose Calculation Manual (ODCM) limits (two (2) times 1500 millirem/year) at the site boundary for sixty (60) minutes (and therefore not result in doses to the public above EPA PAGs beyond the controlled area boundary). In addition to administrative limits on radioactive source term accumulation, administrative controls will be in place to limit the dispersal of radioactive material. These administrative limits and dispersal controls are in addition to the requirements already specified in the ODCM for control of effluent releases.

The PDEP EAL scheme used at FCS in is based on NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors,” Revision 6 (Reference 7.8). The proposed IOEP EAL scheme format is based on NEI 99-01, Revision 6, as appropriate after the transfer of the spent fuel from the SFP to the ISFSI. The proposed revisions constitute a change in the emergency planning function commensurate with the ongoing and anticipated reduction in radiological source term at FCS.

3.0 PROPOSED CHANGES

Replacement of the FCS PDEP and associated EAL Technical Bases Document with the IOEP and the ISFSI EAL Technical Bases Document involves the following major changes to the FCS PDEP:

1) Removal of the various emergency actions related to the SFP,
2) Removal of non-ISFSI-related emergency event types,
3) Removal of the judgment EAL's
4) Clarifying definitions for security EALs
5) Revision of the Emergency Response Organization (ERO), and
6) Identification of the "ISFSI Shift Supervisor (ISS) title as the position that assumes the Emergency Director (ED) responsibilities following an emergency declaration
7) Removal of requirement to perform accountability after declaration of an emergency.

The off-normal events and accidents addressed in the IOEP are related to the dry storage of spent nuclear fuel within the ISFSI and include only the off-normal, accident, natural phenomena, and hypothetical events and consequences presented in the Updated Final Safety Analysis Report (UFSAR), NUH-003, "Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel", for the AREVA TN Americas. After all fuel is removed from the FCS SFP, there will no longer be any potential for the accidents previously described in the FCS emergency plan that would increase risk to the health and safety of the public. These accidents included events specifically related to the storage of the spent fuel in the SFP. After the transfer of the spent fuel from the SFP to the ISFSI, the spent fuel storage and handling systems will be removed from operation.

The proposed revisions to the FCS emergency plan and associated EAL scheme are commensurate with the reduction in radiological hazards associated with the transfer of the spent fuel from the SFP to the ISFSI and will allow the facility to transition to an emergency plan and EAL scheme specifically related to the storage of the spent fuel in the ISFSI. The proposed changes are necessary to properly reflect the conditions of the facility and to maintain the effectiveness of the emergency plan.

3.1 Elimination of SFP Initiating Conditions and EALs and Alert Classification

The Initiating Conditions (ICs) and EALs associated with emergency classification in the PDEP are based on NEI 99-01, Revision 6. Specifically, Appendix C of NEI 99-01 contains a set of ICs and EALs for permanently defueled nuclear power plants that had previously operated under a 10 CFR Part 50 license and have permanently ceased power operations.

After all spent fuel has been transferred from the SFP to dry storage within the ISFSI, the NEI 99-01, Appendix C ICs and EALs that are specifically associated with the SFP are no longer required to be in the emergency plan. Additionally, certain ICs and EALs, the primary function of which is not associated with the SFP, are also no longer required to be in the emergency plan when administrative controls are established to limit source term accumulation and the offsite consequences of uncontrolled effluent releases.

Therefore, the ICs listed in Table 1, below, are proposed for elimination and are not included in the IOEP and EAL scheme.

With respect to the aircraft-related EALs; Interim Compensatory Measures (ICM) Order EA-02-026, Section B.5.b mitigation strategies (dated February 25, 2002) (Reference 7.9) was

issued and subsequent security-based ICs and EALs were provided to licensees in NRC Bulletin (BL) 2005-02, "Emergency Preparedness and Response Actions for Security Based Events," dated July 18, 2005 (Reference 7.10). BL 2005-02 was addressed to all holders of operating licenses for nuclear power reactors, except those who had permanently ceased operation and had certified that fuel has been removed from the reactor vessel.

In 2009, the NRC amended its security regulations adding new security requirements pertaining to nuclear power reactors. This rulemaking established and updated generically applicable security requirements similar to those previously imposed by Commission orders issued after the terrorist attacks of September 11, 2001. In the Statements of Consideration (SOC) for the Final Rule for Power Reactor Security Requirements (74 *Federal Register* (FR) 13926; March 27, 2009), the Commission stated, in part:

> *"Current reactor licensees comply with these requirements through the use of the following 14 strategies that have been required through an operating license condition. These strategies fall into the three general areas identified by §§ 50.54(hh)(2)(i), (ii), and (iii). The firefighting response strategy reflected in § 50.54(hh)(2)(i) encompasses the following elements:….*
>
> *7. Spent fuel pool mitigation measures"*

As such, the staff maintained EALs for potential or actual aircraft threats for facilities transitioning into decommissioning with spent fuel stored in a SFP, in addition to maintaining the mitigative strategies license conditions required by NRC Order, EA-02-026, "Interim Compensatory Measures (ICM) Order," issued February 25, 2002 (67 FR 9792; March 4, 2002).

The SOC further stated, in part:

> *"The NRC believes that it is inappropriate that § 50.54(hh) should apply to a permanently shutdown defueled reactor where the fuel was removed from the site or moved to an ISFSI. The Commission notes that the § 50.54(hh) do not apply to any current decommissioning facilities that have already satisfied the § 50.82(a) requirements."*

On November 28, 2011, the NRC issued a letter that rescinded Item B.5.b of the ICM Order EA-02-26 (Reference 7.18). The rulemaking codified generically applicable security requirements previously issued by orders and updated the existing power reactor security requirements.

Neither the ICM Order nor 10 CFR 50.54(hh) continue to apply to FCS. Therefore, the ICs deleted also include those associated with the mitigative strategies and response procedures for potential or actual aircraft attack procedures as the spent fuel has been removed from the SFP and is stored in the ISFSI.

10 CFR Part 50, Appendix E (IV)(A)(7) defines "hostile action" as an act directed toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end, as it applies to the capability of implementing the emergency plan during such events. However, in the Statement of Considerations for the 2011 Emergency Plan Final Rule, the NRC excluded non-power reactors from the definition of "hostile action" because a non-power reactor as

defined in 10 CFR 50.2, “Definitions,” is not a nuclear power plant, and presently a regulatory basis had not been developed to support the inclusion of non-power reactors in the definition of “hostile action.”

Even though FCS will continue to maintain a facility license under the auspices of 10 CFR 50, the FCS ISFSI is licensed in accordance with the requirements of 10 CFR 72.212, “Conditions of General License Issued Under 10 CFR 72.210”. As such, the radiological consequences to the public from the FCS ISFSI have been developed in accordance with the requirements of 10 CFR 72.104, “Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS,” and 10 CFR 72.106, “Controlled Area of an ISFSI or MRS.” The use of these regulations to develop the FCS ISFSI Technical Specifications provides corollary alignment for development of an ISFSI EAL scheme that meets the historical purpose of an Emergency Plan, protecting the public from radiological exposure in the event of a design basis accident, using the regulatory technical bases for ISFSI facilities. This technical basis also provides the foundation for development of a radiological EAL that is more in line with the standardized risk from an ISFSI.

FCS recognizes that the practice of using Emergency Planning requirements set forth under 10 CFR 50.47 for Independent Spent Storage Facilities located at operating Nuclear Power Reactors is prudent, and that prudency extends through the period that used fuel is stored in the Spent Fuel Pool for a facility that has submitted the certifications required under 10 CFR 50.82. During these periods, having an Emergency Plan and EAL scheme that is familiar to the Certified Fuel Handlers and operating staff allows for a manageable transition from power operations to removal of all fuel from the Spent Fuel Pool. Once all fuel is placed in dry storage in the ISFSI, the makeup of the facility staff can shift dramatically. This shift, concurrent with the significant reduction in risk to the public, predicates the use of an emergency plan that more closely aligns to that of an emergency plan developed under 10 CFR 72.32. The most significant difference between the proposed FCS EAL scheme and that of a decommissioning power reactor using a 10 CFR 72.32 Emergency Plan is the required use of the ALERT classification for 10 CFR 72.32 emergency plans. All other terminology is essentially the same.

This rationale justifies the exclusion of facilities with permanent removal of fuel from the reactor vessel from the definition for a “hostile action” and its related requirements (including conducting hostile action exercises) as they apply to the Emergency Plan. Elements for security-based events should be maintained for facilities, including ISFSI-only facilities with a 10 CFR Part 50 license to help ensure assistance can be made available during these events. As such, the Alert security classification based on a “hostile action” is being redefined for the FCS IOEP.

Even though a Hostile Action-Based program is not necessary for an ISFSI-only site, precedence from other utilities and regulatory guidance provides that consideration of actions by an adversary for EAL purposes is still applicable. Therefore, the use of the term “ADVERSARIAL ACTION” and the revised definition is included, to reflect those aspects associated with an ISFSI-only site and is utilized in the EALs.

Judgements EALs are being eliminated as part of this submittal to align with Draft Regulatory Guide 1346. The draft does not include the Judgement EALs as part of the IOEP scheme.

Draft Regulatory Guide 1346 also proposes an alternate EAL for determining the occurrence of damage to a loaded storage cask following an event that may cause damage to the loaded casks. This proposed methodology identifies any change in radiation levels above normal background as the initiating condition for the EAL. FCS is proposing to base this EAL on a change in radiation levels significant enough to warrant concern for exceeding the limit to dose to the general public as defined in 10 CFR 20.1301(a)(2) of 0.002 Rem (2 mRem) in any one hour. Establishing an EAL threshold of >2 mRem/hr within the ISFSI protected area or on a Horizontal Storage Module (HSM) concrete surface provides a level of margin to maintain protection of the public, while providing an easily identifiable set point for ISFSI personnel. This level of radiation is high enough to minimize instrument error and operational differences while still providing positive indication of an emergency condition.

The ICs listed in Table 1 are not included in the proposed ISFSI EAL scheme for FCS. The ICs in Table 1 are either associated only with SFP operation or are ICs for which administrative controls to limit possible effluent releases have been established.

Table 1 – Emergency Plan Initiating Conditions Being Deleted

ALERT	UNUSUAL EVENT
PD-RA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mRem TEDE or 50 mRem thyroid CDE.(1)	**PD-RU1** Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.(1)
PD-RA2 UNPLANNED rise in facility radiation levels that impedes facility access required to maintain spent fuel integrity.(1)	**PD-RU2** UNPLANNED rise in facility radiation levels.(1)
~~**PD-HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~ [**EA1** ADVERSARIAL ACTION is occurring or has occurred.] (2) 1. [An ADVERSARIAL ACTION is occurring or has occurred as reported by the security force.] ~~A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by security supervision.~~ ~~2. A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.~~	~~**PD-HU1**~~**[EU1]** Confirmed SECURITY CONDITION or threat.(2) 1. A SECURITY CONDITION ~~that does not involve a HOSTILE ACTION~~ as reported by [the security ~~supervision~~ force and impacting the ISFSI]. 2. Notification of a credible security threat directed at the ~~site~~ [ISFSI]. ~~3. A validated notification from the NRC providing information of an aircraft threat.~~
	PD-HU2 Hazardous event affecting SAFETY SYSTEM equipment necessary for spent fuel cooling.(1)
PD-HA3 Other conditions exist which in the judgment of the Emergency Director warrant declaration of Alert. (1)	**PD-HU3 Other** considerations exist which in the judgment of the Emergency Director warrant declaration of an Unusual Event. (1)
	PD-SU1 UNPLANNED spent fuel pool temperature rise.(1)
	~~**E-HU1**~~ **[EU2]:** Damage to a loaded cask CONFINEMENT BOUNDARY. 1. Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an ~~on-contact~~ [abnormal] radiation reading [of >2 mRem/hr (gamma) within the ISFSI Protected Area or on a Horizontal Storage Module (HSM) concrete surface.]

	~~• ≥ 1600 mRem/hr (gamma + neutron) on the Horizontal Storage Module (HSM) front surface~~ ~~**OR**~~ ~~• > 400 mRem/hr (gamma + neutron) on the HSM door centerline~~ ~~**OR**~~ ~~• > 16 mRem/hr (gamma + neutron) on the end shield wall exterior~~

(1) Indicates the IC and the associated EALs are being deleted in their entirety.

(2) Indicates only the portion of the IC or EAL shown in strikethrough text is being deleted. Text included with brackets [ ] will be added in the proposed ISFSI EAL Scheme.

The ICs being deleted include all ICs associated with the categories of abnormal radioactive release and system malfunction associated with the SFP as well as security conditions associated with aircraft. These categories apply only to SFP operation and are not appropriate given the minimized risk of having all spent fuel stored within the ISFSI.

The ICs listed in Table 2, below, are being retained. The ICs being retained in the ISFSI Only Emergency Plan are appropriate to address the condition of a facility in which all spent fuel is stored in the ISFSI.

Table 2 – ISFSI Emergency Plan Initiating Conditions

UNUSUAL EVENT
SECURITY
EU1 (formally PD-HU1) Confirmed SECURITY CONDITION, or threat, at the independent spent storage installation (ISFSI).
Independent Spent Fuel Storage Installation (ISFSI)
EU2 (formally E-HU1)**:** Damage to a loaded cask CONFINEMENT BOUNDARY.
ALERT
SECURITY
EA1 (formally PD-HA1) ADVERSARIAL ACTION is occurring or has occurred.

3.2 Emergency Response Organization Revision

The FCS PDEP provides for two (2) ERO augmented positions – a Technical Coordinator and a Radiation Protection Coordinator. The proposed FCS IOEP replaces these positions with a Resource Manager and an individual trained in radiological monitoring and assessment.

A Resource Manager is provided to assist in assessing the event and obtaining needed resources. The Resource Manager is required to be in contact with the Emergency Director (ED) within two (2) hours of declaration of an Unusual Event or an Alert. Entry into the IOEP would result from an extreme natural phenomenon (beyond design basis) or a security condition, either of which would negatively impact or restrict access to the site.

The Resource Manager augments the ED by assisting in assessing the emergency condition and coordinating the required resources, including serving as the public information interface. Services provided to the ED by the Resource Manager can be provided remotely and do not necessitate an onsite response by the Resource Manager. By responding remotely, the actual response time is decreased with no negative impact to services and functional responsibilities provided by the Resource Manager. The Resource Manager's functional responsibilities could be performed in a timely manner either by reporting to the site or performing the function remotely in the specified timeframe.

In addition, FCS proposes that, for a classified event involving radiological consequences, a minimum of one person trained in radiological monitoring and assessment will report to the ISFSI within four hours of the emergency declaration.

The proposed FCS IOEP also provides that additional personnel resources may be directed to report to FCS to provide additional support as needed to assess radiological conditions, support maintenance and repair activities, develop and implement corrective action plans, and assist with recovery actions. The augmentation personnel are available from FCS staff, OPPD, and from various contractors.

3.3 Replacement of the "Shift Manager" with the "ISFSI Shift Supervisor"

The FCS PDEP assigns the authority and responsibility for control and mitigation of emergencies to the Shift Manager (SM). If an emergency condition develops, the SM would assume the role of ED. The proposed FCS IOEP proposes replacing the SM position with an ISS within the IOEP.

The ISS will be at FCS on a continuous, 24 hour per day basis, and is the senior management position during off-hours. This position is responsible for monitoring ISFSI conditions and managing the activities at the FCS ISFSI. This position assumes overall command and control of the response as the ED and is responsible for monitoring conditions and approving all onsite activities. The IOEP identifies non-delegable responsibilities, along with other designated tasks. OPPD considers this an administrative change which will not impact the timing or performance of existing emergency response duties.

3.4 Removal of requirement to conduct accountability following declaration of an emergency.

The specification for accountability from section J.5 of revision 1 of NUREG-0654 (Reference 7.13) reads as follows.

> "*Each licensee shall provide for a capability to account for all individuals onsite at the time of the emergency and ascertain the names of missing individuals within 30 minutes of the start of an emergency and account for all onsite individuals continuously thereafter.*"

The previously approved exemptions and PDEP for FCS removed the requirements for Site Area Emergencies and General Emergencies. Accountability of personnel is a process required for the Protected Area at most nuclear plants when a Site Area Emergency or General Emergency has been declared. Accountability is necessitated at these classifications due to the potential for significant radiological exposure or other health hazards to site personnel. As the facility transitions to an ISFSI only site, the need for accountability diminishes as a result of the following:

- significantly smaller staff (less than 5% of an operational facility)
-the facility only has one building
-entry into the Protected Area is intermittent, with no permanent occupation other than that required for the security plan
-the staff at the ISFSI will be in continuous communication with each other
-the entire facility is under video surveillance, and monitored 24 hours a day

The proposed IOEP for FCS does not contain an emergency classification higher than ALERT, and considering the factors specified previously, the requirement to conduct accountability following an emergency declaration is no longer warranted.

3.5 Removal of emergency notification to the State of Iowa.

The State of Iowa Department of Homeland Security formally requested to be removed from any emergency notifications associated with FCS.

4.0 TECHNICAL EVALUATION

4.1 Radiological Consequences of Design Basis Events

FCS is located midway between Fort Calhoun and Blair, Nebraska, on the west bank of the Missouri River. The site is located approximately 19 miles North of Omaha, Nebraska and four (4) miles South of Blair, Nebraska. The ISFSI is located within a Protected Area on the site. Except for the city of Blair and the villages of Fort Calhoun and Kennard, the area within a ten mile radius is predominantly rural and land use is primarily devoted to general farming. There are no private businesses or public recreational facilities on the plant property.

Chapter 14 of the FCS Final Safety Analysis Report, as Updated described the Abnormal Operational Transients and Design Basis Accident (DBA) scenarios applicable to FCS during power operations. However, after permanent cessation of power operations and transfer of all irradiated fuel from the SFP to dry storage within the ISFSI, the remaining accident scenarios postulated in the Defueled Safety Analysis Report (DSAR) are no longer possible. The ISFSI is a passive storage system that does not rely on electric power for heat transfer. After removal of the spent fuel from the SFP, there are no credible fuel-related accidents for which actions of a Certified Fuel Handler, SM, or Non-Certified Operator are required to prevent occurrence or to mitigate the consequences. There is no credible accident resulting in radioactive releases requiring offsite protective measures.

The robust design and construction of the spent fuel storage system selected for use at the ISFSI prevents the release of radioactivity in the event of an off-normal or accident event as described in the NUHOMS UFSAR. Leakage of fission products from a canister confinement boundary breach is not considered to be a credible event, given the high integrity nature of the canister's design and the additional protection afforded by the storage casks.

FCS PSDAR documents the decommissioning strategy selected for FCS. Systems that are not required to support the spent fuel, HVAC, Emergency Plan, or site security will be drained, de-energized, and secured and the plant will remain in a stable condition until final decontamination and dismantlement activities begin. The PSDAR documents the time period that OPPD expects to have all spent fuel transferred to the ISFSI. After the fuel transfer is completed, the SFP and associated systems will be drained and de-energized.

After all the spent fuel has been removed from the SFP, the estimated radiological inventory (non-fuel) that remains at the reactor facility is primarily attributable to activated reactor components and structural materials. There are no credible accident scenarios that can mobilize a significant portion of this inventory for release. As a result, the potential accidents that could occur during decommissioning of the reactor facility have negligible offsite and onsite radiological consequences.

With all spent nuclear fuel in dry storage within the ISFSI, the radiological status of the facility required for implementing this proposed IOEP is summarized as follows:

- The remaining radiological source term at FCS will not create an unplanned/unanticipated increase in radiation or in liquid or airborne radioactivity levels that would result in doses to the public above EPA PAG limits at the site boundary.
- Source term accumulation from activities during decontamination and dismantlement of radioactive systems, structures, and components are administratively controlled at a level that would preclude declaring an Unusual Event.
- Necessary radiological support personnel will be administratively required to be onsite during active decontamination and dismantlement of radioactive systems, structures, and components.
- The IOEP, and certain ICs and EALs for which administrative controls to limit possible effluent releases will be established, do not apply to the decontamination and dismantlement of radioactive systems, structures, and components.

NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (NUREG-0586) (Reference 7.11) supports this conclusion in the following statement:

> *"The staff has reviewed activities associated with decommissioning and determined that many decommissioning activities not involving spent fuel that are likely to result in radiological accidents are similar to activities conducted during the period of reactor operations. The radiological releases from potential accidents associated with these activities may be detectible. However, work procedures are designed to minimize the likelihood of an accident and the consequences of an accident, should one occur, and procedures will remain in place to protect health and safety*

while the possibility of significant radiological accident exists."

NUREG-0586 also includes the following statement:

> *"The staff has considered available information, including comments received on the draft of Supplement 1 of NUREG-0586, concerning the potential impacts of non-spent fuel related radiological accidents resulting from decommissioning. This information indicates, that with the mitigation procedures in place, the impacts of radiological accidents are neither detectible nor destabilizing. Therefore, the staff makes the generic conclusion that impacts of non-spent fuel related radiological accidents are SMALL. The staff has considered mitigation and concludes that no additional measures are likely to be sufficiently beneficial to be warranted."*

Accordingly, administrative controls that are designed to minimize the likelihood and consequence of off-normal or accident events would be implemented when decontamination or dismantling activities involving radioactive systems, structures, or components are being performed.

Implementation of the IOEP would involve FCS establishing administrative controls for radiological source term accumulation limits and methods to control the accidental dispersal of the radiological source. Examples of radiological source term accumulation limits are based on:

- Radioactive materials collected on filter media and resins (dose rate limit)
- Contaminated materials collected in shipping containers (dose rate limit)
- Surface or fixed contamination on work areas that may create airborne radioactive material (activity limits)
- Radioactive liquid storage tank(s) (activity concentration limits)

An example of a method to control accidental dispersal of the radiological source term is limitation on dispersal mechanisms that may cause a fire (e.g., limits on combustible material loading, use of fire watch to preclude fire, etc.), or placement of a berm around a radioactive liquid storage tank. If the dispersal control fails, the limits on source term would preclude exceeding the site boundary source term limit.

As discussed in the previously requested exemptions from various emergency planning requirements contained in 10 CFR 50.47 and 10 CFR 50, Appendix E, an analysis of the potential radiological impact of a design basis accident at FCS in a permanently defueled condition indicates that any releases beyond the site boundary are below EPA PAG exposure levels. The basis for these exemptions has not changed and remains in effect for the proposed IOEP.

4.2 Radiological Consequences of Postulated Events

Although the limited scope of postulated accidents that remain applicable to the FCS facility justifies a reduction in the necessary scope of emergency response capabilities, FCS also assessed beyond design basis events using past industry precedence, including information contained in Appendix I, "Radiological Accidents," of NUREG-0586.

With spent fuel stored within the SFP, the most severe postulated beyond design basis event involved a highly unlikely sequence of events that causes heatup of the spent fuel, postulated to occur without any heat transfer, such that the zircaloy fuel cladding reaches ignition temperature (adiabatic heat up). The resultant zircaloy fire could lead to the release of large quantities of fission products to the atmosphere. However, after removal of the spent fuel from the SFP, the configuration of the spent fuel stored in dry storage precludes the possibility of such a scenario.

With this previously limiting beyond design basis scenario no longer possible, FCS assessed the following beyond design basis events associated with performance of decommissioning activities with all irradiated fuel stored in the ISFSI. A summary of the assessments is provided below:

1. Cask Drop Event (Fuel-Related Event)

FCS is the holder of a general license for the storage of spent fuel in an ISFSI at power sites in accordance with the provisions of 10 CFR 72.210 and 10 CFR 72.212. The generally licensed ISFSI at FCS is used for interim onsite dry storage of spent nuclear fuel assemblies in the Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel, (Certificate of Compliance (CoC) 1004).

As documented in the NUHOMS UFSAR, NUH-003, analysis of the normal events, including drop events, determined that canister drops can be sustained without breaching the confinement boundary, preventing removal of spent fuel assemblies, or creating a criticality accident. There are no evaluated normal conditions or off-normal or accident events that result in damage to the canister producing a breach in the confinement boundary. Neither normal conditions of operation or off-normal events preclude retrieval of the fuel for transport and ultimate disposal.

The dry spent fuel storage casks used at FCS are approved for storage of spent fuel per 10 CFR 72.214; and, as such, are in compliance with the requirements of 10 CFR 72.24 and 10 CFR 72.122 for off-normal and accident events to ensure that they will provide safe storage of spent fuel during all analyzed off-normal and accident events. Therefore, no radiological release beyond the site boundary would be expected to occur.

2. Radioactive Material Handling Accident (Non-Fuel-Related Event)

The limiting non-fuel related event involves the release of radioactive material from a concentrated source, such as filters, resins, and shipping containers (as discussed in NUREG-0586, Appendix 1). The initiator to these events could be a fire, explosion, or a handling event (cask drop). After all spent fuel has been moved to the ISFSI, there would be no concentrated source of radioactive material available to be released to the environment in an amount that could exceed two (2) times the ODCM limit at the site boundary (2 times 1500 millirem/year). During decontamination and dismantlement activities, administrative controls would limit the total amount of activity that could accumulate in a concentrated source. FCS Calculation FC08566 (Attachment 1) details an activity accumulation limit methodology for decontamination and dismantlement of irradiated stainless steel (e.g., reactor vessel internals) and irradiated concrete (e.g., reactor coolant loop bio-shield walls) based on isotopic mixtures from NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials," (Reference 7.12) such that a release to the environment from concentrated sources of these radioactive materials would not

exceed two times the ODCM at the site boundary.

It is expected that representative material samples will be taken and analyzed prior to actual decontamination/dismantlement work. Using the methodology consistent with this calculation, container/filter maximum radioactivity limits will be derived.

The results of the above assessment indicate that the projected radiological doses at the controlled area boundary are less than the EPA PAGs.

3. Accidents Initiated by External Events

The effects of external events, such as fires, floods, wind (including tornados), earthquakes, lightning, and physical security breaches on the ISFSI remain unchanged from the effects that were considered under the proposed PDEP. Externally initiated events are addressed by the proposed ISFSI EALs.

In summary, there continues to be a low likelihood of any postulated event resulting in radiological releases requiring offsite protective measures, and there is no credible radioactive material event (non-fuel related) resulting in radiological releases requiring declaration of an emergency.

4.3 ISFSI ONLY EMERGENCY PLAN

The FCS IOEP is provided in Enclosure 1, Attachment 3 to this submittal for NRC review and approval. This proposed emergency plan is associated with EALs for events related to the ISFSI. The IOEP addresses the applicable regulations stipulated in 10 CFR 50.47, "Emergency Plans;" 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities" (considering the exemptions requested in Reference 7.3); and 10 CFR 72.32, "Emergency Plan," and is consistent with the applicable guidelines established in NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (Reference 7.13).

The IOEP describes FCS's plan for responding to emergencies while all spent fuel is in dry storage within an ISFSI. After all spent fuel at FCS is in dry storage within the ISFSI, the number and severity of potential radiological accidents is significantly less than when fuel is stored in the SFP.

The FCS IOEP conservatively provides that the emergency planning zone for the ISFSI is the area within the site boundary. At FCS, the site boundary completely encompasses the controlled area. The controlled area, as defined in 10 CFR 72.3, "Definitions," means the area immediately surrounding an ISFSI for which FCS exercises authority over its use and within which ISFSI operations are performed.

The controlled area is established to limit dose to the public during normal operations and design basis accidents in accordance with the requirements of 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS," and 10 CFR 72.106, "Controlled Area of an ISFSI or MRS." FCS's analysis of the radiological impact of potential accidents at the ISFSI conclude that any releases beyond the ISFSI controlled area are expected to be less than the EPA PAGs. The controlled area is

completely enclosed within the site boundary. Thus, any radiological releases beyond the site boundary will also be less than the EPA PAGs.

Based on the reduced number and consequences of potential radiological events with all spent fuel in dry storage within the ISFSI, there will continue to be no need for offsite emergency response plans for the protection of the public beyond the site boundary. Additionally, the scope of the onsite emergency preparedness organization and corresponding requirements in the emergency plan may be reduced without an undue risk to the public health and safety.

The analysis of the potential radiological impact of an accident in a condition with all irradiated fuel stored in the ISFSI indicates that any releases beyond the site boundary are below the EPA PAG exposure levels. Exposure levels, which warrant pre-planned response measures, are limited to onsite areas. For this reason, radiological emergency planning is focused onsite.

4.4 ISFSI Emergency Action Levels

Enclosure 1, Attachment 4 of this submittal provides the FCS ISFSI EAL Technical Bases Document, which contains the proposed FCS ISFSI EAL scheme for NRC review and approval. . The proposed ISFSI EAL scheme is to be implemented by the FCS ISFSI Emergency Plan (provided in Enclosure 1).

Deletions from the proposed Permanently Defueled EAL scheme are identified in Table 1, "Emergency Plan Initiating Conditions Being Deleted," in Section 3.1, "Elimination of SFPs Initiating Conditions and EALs," above.

Related Documents

Supporting evaluations/calculations for establishing appropriate radioactive material administrative control limits are provided in Attachment 1 to this submittal.

Operating Modes and Applicability

The proposed ISFSI EALs are only applicable after the final spent nuclear fuel assembly has been transferred out of the SFP and placed in dry storage within the ISFSI.

State and Local Government Review of Proposed Changes

State and local emergency management officials are advised of EAL changes that are implemented. Prior to implementation of the EAL scheme proposed in this License Amendment Request (LAR), FCS will provide an overview of the new classification scheme to State and local emergency management officials in accordance with 10 CFR 50, Appendix E, Section IV.B.1.

5.0 REGULATORY EVALUATION

The proposed emergency plan does not meet all standards of 10 CFR 50.47(b) and requirements of 10 CFR Part 50, Appendix E. However, FCS previously received exemptions from portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR 50, Appendix E, Section IV, by letter dated December 11, 2017 (Reference 7.3). The basis for these exemptions has not changed and

remains in effect for the emergency plan changes requested in this document. Considering the previously approved exemptions, the emergency plan, as revised, will continue to meet the remaining applicable requirements in 10 CFR Part 50, Appendix E and the remaining applicable planning standards of 10 CFR 50.47(b).

5.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," OPPD requests NRC approval of a reduction in effectiveness of the site Emergency Plan by the removal of several EALs and corresponding changes to the emergency plan, to be implemented after all spent fuel has been removed from the SFP and placed in dry storage within the ISFSI. The proposed IOEP and ISFSI EAL Technical Bases Document are commensurate with the reduction in radiological source term at FCS. The PSDAR documents the time period that FCS expects to have all spent fuel transferred to the ISFSI. To comport to the reduced scope of potential radiological accidents with all spent fuel in dry cask storage within the ISFSI, FCS proposes a new emergency plan and corresponding EAL scheme.

Pursuant to 10 CFR 50.92, OPPD has reviewed the proposed changes and concludes that the changes do not involve a significant hazards consideration because the proposed changes satisfy the criteria in 10 CFR 50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed changes would revise the FCS emergency plan and EAL scheme commensurate with the hazards associated with a permanently shut down and defueled facility that has transferred all spent fuel from the SFP to dry cask storage within the ISFSI.

The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

1. <u>Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   Response: No.

   The proposed amendment would modify the FCS facility operating license by revising the emergency plan and EAL scheme. FCS has permanently ceased power operations and is permanently defueled. The proposed amendment is conditioned on all spent nuclear fuel being removed from wet storage in the SFP and placed in dry storage within the ISFSI. Occurrence of postulated accidents associated with spent fuel stored in a SFP is no longer credible in a SFP devoid of such fuel. The proposed amendment has no effect on plant systems, structures, or components (SSC) and no effect on the capability of any plant SSC to perform its design function. The proposed amendment would not increase the likelihood of the malfunction of any plant SSC. The proposed amendment would have no effect on any of the previously evaluated accidents in the FCS DSAR.

Because FCS has permanently ceased power operations, the generation of fission products has ceased and the remaining source term continues to decay. This continues to significantly reduce the consequences of previously evaluated postulated accidents.

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

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

   Response: No.

   The proposed amendment constitutes a revision of the emergency planning function commensurate with the ongoing and anticipated reduction in radiological source term at FCS.

   The proposed amendment does not involve a physical alteration of the plant. No new or different types of equipment will be installed and there are no physical modifications to existing equipment as a result of the proposed amendment. Similarly, the proposed amendment would not physically change any SSC involved in the mitigation of any postulated accidents. Thus, no new initiators or precursors of a new or different kind of accident are created. Furthermore, the proposed amendment does not create the possibility of a new failure mode associated with any equipment or personnel failures. The credible events for the ISFSI remain unchanged.

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

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

   Response: No.

   Because the 10 CFR Part 50 license for FCS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. With all spent nuclear fuel transferred out of wet storage from the SFP and placed in dry storage within the ISFSI, a fuel handling accident is no longer credible. There are no credible events that would result in radiological releases beyond the site boundary exceeding the EPA PAG exposure levels, as detailed in the EPA's PAG Manual "Protective Action Guides and Planning Guidance for Radiological Incidents" dated January 2017 (EPA PAG Manual).

   The proposed amendment does not involve a change in the plant's design, configuration, or operation. The proposed amendment does not affect either the way in which the plant SSCs perform their safety function or their design margins. Because there is no change to the physical design of the plant, there is no change to these margins.

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

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

5.2 Applicable Regulatory Requirements/Criteria

The regulatory requirements, considering the previously requested exemptions are discussed below.

Title 10 of the Code of Federal Regulations (10 CFR), Section 50.47, "Emergency Plans," set forth emergency plan requirements for nuclear power plant facilities. The regulations in 10 CFR 50.47(a)(1)(i) state, in part:

> *“no initial operating license for a nuclear power reactor will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.”*

Section 50.47(b) establishes the standards that emergency response plans must meet for NRC staff to make a positive finding that there is reasonable assurance that the licensee can and will take adequate protective measures in the event of a radiological emergency.

- Planning Standard (1) of Section 50.47(b) states, in part: “*[E]ach principal response organization has staff to respond and to augment its initial response on a continuous basis.”*
- Planning Standard (2) of Section 50.47(b) states, in part: “*On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available…”*
- Planning Standard (4) of Section 50.47(b) requires that a licensee’s emergency response plan contain the following: “*A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee.”*
- Planning Standard (8) of Section 50.47(b) states, in part: “*Adequate emergency facilities and equipment to support the emergency response are provided and maintained.”*

10 CFR 50.54(q)(4) specifies the process for revising emergency plans where the change reduces the effectiveness of the plan. This regulation states the following:

> *“The changes to a licensee’s emergency plan that reduce the effectiveness of the plan as defined in paragraph (q)(1)(iv) of this section may not be implemented without prior approval by the NRC.”*

Section IV.A of Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” to 10 CFR Part 50, states, in part: “*The organization for coping with radiological emergencies shall be described, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee’s emergency organization…”*

Section IV.C.1 of Appendix E requires that each emergency plan define the emergency classification levels that determine the extent of participation of the emergency response organization.

Section IV.E of Appendix E states, in part: "*Adequate provisions shall be made and described for emergency facilities and equipment…*". As identified in 10 CFR 72.13, "Applicability," the applicable emergency plan requirements for an ISFSI associated with a general license are specified in 10 CFR 72.32(c) and (d).

The proposed emergency plan continues to rely on previously requested exemptions from certain emergency planning requirements as the basis for these exemptions has not changed and remains in effect.

The proposed changes are conservatively being considered as a change to the EAL scheme development methodology. Pursuant to 10 CFR Part 50, Appendix E, Section IV.B.2, a revision to an entire EAL scheme must be approved by the NRC before implementation.

5.3 Precedent

Similar changes to emergency plans and associated EAL schemes approved by the NRC for plants that have transitioned to ISFSI-only status include: 1) the La Crosse Boiling Water Reactor (LACBWR) facility on September 8, 2014 (Reference 7.15); 2) the Zion Facility on May 14, 2015 (Reference 7.16); and 3) Duke Energy Florida, Inc. for the Crystal River Unit 3 Nuclear Generating Station on August 12, 2016 (Reference 7.17).

5.4 Conclusion

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

6.0 ENVIRONMENTAL CONSIDERATIONS

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section 5.1 of this evaluation, the proposed changes involve no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed changes do not involve any physical alterations to the plant configuration or any changes to the operation of the facility that could lead to a change in the type or amount of effluent release offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not involve any physical alterations to the plant configuration or any changes to the operation of the facility that could lead to a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, OPPD concludes that the proposed change meets the eligibility criteria for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

7.0 REFERENCES

7.1 OPPD Letter (T. Burke) to USNRC (Document Control Desk) – "Certification of Permanent Cessation of Power Operations," dated August 25, 2016 (LIC-16-0067) (ADAMS Accession No. ML16242A127)

7.2 OPPD Letter (T. Burke) to USNRC (Document Control Desk) – "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel," dated November 13, 2016 (LIC-16-0074) (ADAMS Accession No. ML16319A254)

7.3 Letter USNRC (J. Kim) to OPPD (M. Fisher) – "Fort Calhoun Station, Unit No. 1, Exemptions From Certain Emergency Planning Requirements and Related Safety Evaluation", dated December 11, 2017 (LIC-16-0109) (CAC No. MF9067) (ML17263B198; ML17263B191; ML17278A178)

7.4 OPPD Letter (T. Burke) to USNRC (Document Control Desk) – "License Amendment Request 16-05 to Revise the Fort Calhoun Station Emergency Plan to the Permanently Defueled Emergency Plan and Permanently Defueled Emergency Action Level Scheme," dated December 16, 2016 (LIC-16-0108) (ADAMS Accession No. ML16351A464)

7.5 Letter USNRC (J. Kim) to OPPD (M. Fisher) – "Fort Calhoun Station, Unit No. 1, Post-Shutdown Decommissioning Activities Report", dated March 23, 2017 (LIC-17-0033) (CAC No. 9536) (ML18011A687)

7.6 Nuclear Regulatory Commission to AREVA TN Americas' CoC 1004, Amendment 14, CoC, dated March 31, 2017, effective April 25, 2017. (ADAMS Accession No. ML 17191A236)

7.7 U.S. Environmental Protection Agency, "Protective Action Guide and Planning Guidance for Radiological Incidents," dated January 2017 (PAG Manual)

7.8 Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012. (ADAMS Accession No. ML12326A805)

7.9 NRC Interim Compensatory Measures (ICM) Order EA-02-026, Section B.5.b mitigation strategies (dated February 25, 2002) (ADAMS Accession No. ML020510635)

7.10 NRC Bulletin (BL) 2005-02, "Emergency Preparedness and Response Actions for Security Based Events," dated July 18, 2005 (ADAMS Accession No. ML051740058)

7.11 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," Supplement 1, Volume 1, dated November 2002

7.12 NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials," dated August 2000

7.13 NUREG-0654, FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, published November 1980

7.14 Letter, Mark Thaggard (USNRC) to Susan Perkins-Grew (NEI), "U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 99-01, Revision 6, Dated November, 2012 (TAC No. D92368)," dated March 28, 2013 (ADAMS Accession No. ML12346A463)

7.15 Letter from U.S. Nuclear Regulatory Commission to Dairyland Power Cooperative (La Crosse Boiling Water Reactor) "Issuance of Amendment Relating to the Dairyland Power Cooperative La Crosse Boiling Water Reactor Request for Changes to the Emergency Planning Requirements," dated September 8, 2014 (ADAMS Accession No. ML14155A112)

7.16 Letter from U.S. Nuclear Regulatory Commission to Zion Solutions LLC (Zion Nuclear Power Station), "Issuance of Amendments Relating to the Emergency Planning Requirements for Zion Nuclear Power Station, Units 1 and 2," dated May 14, 2015 (ADAMS Accession No. ML15092A423)

7.17 Memo, Office of Nuclear Security and Incident Response, Reactor Licensing Branch, Division of Preparedness and Response to Office of Nuclear Materials Safety and Safeguards, Division of Decommissioning, Uranium Recovery and Waste

Programs, Reactor Decommissioning Branch, "Safety Evaluation Input for the Crystal River Unit 3 Independent Spent Fuel Storage Installation Only Emergency Plan (CAC No L53129)," dated August 12, 2016 (ADAMS Accession No. ML16201A135)

7.18 Letter from U.S. Nuclear Regulatory Commission for Holders of Licenses for Operating Power Reactors "Rescission or Partial Rescission of Certain Power Reactor Security Orders Applicable to Nuclear Power Plants," dated November 28, 2011 (ADAMS Accession No. ML111220447)

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION

DOCKET NUMBER 50-285 / LICENSE NUMBER DPR-40

ATTACHMENT 1

SUPPORTING EVALUATIONS AND CALCULATIONS

ATTACHMENT 1
Design Analysis Cover Sheet
Page 1 of 1

Design Analysis		**Last Page No.** [6] 32	
Analysis No.: [1]	FC08566	**Revision:** [2] 0 Major ☒	Minor ☐
Title: [3]	Dose Consequences of a High Integrity Container (HIC) Drop Event		
EC No.: [4]	70115	**Revision:** [5] 0	

		Component(s): [14]	
Station(s): [7]	FCS		
Unit No.: [8]	1		
Discipline: [9]	Nuc		
Descrip. Code/Keyword: [10]	RW		
Safety/QA Class: [11]	Safety Related		
System Code: [12]	NA		
Structure: [13]	NA		

CONTROLLED DOCUMENT REFERENCES [15]

Document No.:	From/To	Document No.:	From/To
RP Calc FC-17-001	From		
CH-ODCM-0001	From		
FC08790	From		

Is this Design Analysis Safeguards Information? [16] Yes ☐ No ☒ If yes, see SY-FC-101-106

Does this Design Analysis contain Unverified Assumptions? [17] Yes ☐ No ☒ If yes, ATI/AR#:

This Design Analysis SUPERCEDES: [18] NA in its entirety.

Description of Revision (list changed pages when all pages of original analysis were not changed): [19]

	Print Name	Sign Name	Date
Preparer: [20]	Carol Waszak	Carol Waszak	10/29/18

Method of Review: [21] Detailed Review ☒ Alternate Calculations (attached) ☐ Testing ☐

	Print Name	Sign Name	Date
Reviewer: [22]	Jim Carlson	Jim Carlson	10/29/18

Review Notes: [23] Independent review ☒ Peer review ☐

(For External Analyses Only)

	Print Name	Sign Name	Date
External Approver: [24]	NA		
FCS Reviewer: [25]	NA		

Independent 3rd Party Review Reqd? [26] Yes ☐ No ☒

	Print Name	Sign Name	Date
FCS Approver: [27]	Jerry Shuck		10-30-18

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1.0 PURPOSE

The purpose of this calculation is to determine the radiation dose to the public due to a postulated High Integrity Container (HIC) drop event. In addition, this calculation will assess if the dose is below the acceptance criteria listed below.

Acceptance Criteria:

1) Less than 1 rem TEDE over 4 days at the Control Area Boundary based on the EPA PAG (ref 5.10) for the Early Phase release.

2) Less than 10 mrem TEDE at the Control Area Boundary based on NEI 99-01 (Ref 5.11), Appendix C, Table PD-1.

2.0 INPUTS

- Respirable Airborne Release Fraction is 1E-3 (See att. 1 and ref 5.8)
- BR = 3.50E-04 m^3/sec - Breathing rate of reference man is in accordance with Reg Guide 1.183 (ref 5.5)
- Nuclides in a Resin Mix (Attachment 3) 10 CFR Part 61 analyses for previous resin shipments were used to create a bounding resin-HIC which represents the maximum values for observed ratios of hard to detect nuclides.
- Nuclides in Plant Mix - The non-resin HIC uses ratios previously determined in FC-17-001 (ref 5.3).

3.0 ASSUMPTIONS

- Damage Fraction is 1.0 (100% damage to container)(ref 5.8)
- Leakage Fraction is 1.0 (all contents of container potentially at risk)(ref 5.8)
- Maximum gross weight of HIC was used as weight of contents inside the HIC, ignoring density.
- A bounding value of 1000 curies is used as the content of the HIC. This is conservative based on the weight of the HICs used at Fort Calhoun
- No ventilation or filtration is credited for reducing release
- Release of airborne material during the event occurred as a 'puff' release in which all of the material is released at once.
- 100% of the activity is available for release. Although the bulk of the activity in inert metals that may be loaded into the container are internal in the metal itself, it is tremendously conservative to assume 100% of the activity is available to be released. In reality, only the loose surface contamination would be released on a dropped container. There is no way to predict the size, shape, surface area, thickness or portion of the activity that is due to surface contamination vs fixed or internal contamination, thus the assumption is that the contents are of a granular or powder like substance where all of the activity of the material could potentially become airborne. This is likewise for resins, the radioactive particles are ionically bonded to the resin bead media.

- The shortest distance from the drop location to the control area boundary is 400 meters. This is reasonable due to the Aux Building Stack distance to the control area boundary is 464.82 meters (Ref 5.12).

- A 20% factor of conservatism is applied to the summary values. For dose values, the reported mrem values are 20% greater than calculated values.

- Both HICs and other plant hardware will be analyzed as ratios can vary dramatically in resins.

4.0 IDENTIFICATION OF COMPUTER PROGRAMS

XOQDOQ, Version 2 was used to calculate the X/Q. This computer program is used by the NRC in its independent meteorological evaluation of continuous and anticipated intermittent release from commercial nuclear power reactors. The program implements the assumptions outlined in Section C of NRC Regulatory Guide 1.111. (Ref. 5.13).

XOQDOQ, Version 2 is maintained approved in OPPD SWIMS per SCRC0000132018.

5.0 REFERENCES

5.1 CH-ODCM-0001, Rev 28, *Offsite Dose Calculation Manual (ODCM)*

5.2 10 CFR Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste.*

5.3 FC-17-001 *Evaluation of Instrument Response to Measured Plant Radionuclide Mix" Fort Calhoun Station Calculation, January 2017.*

5.4 RG 1.195 *Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors.*

5.5 RG 1.183 *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.*

5.6 EPA Federal Guidance Report 11 *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation Submersion and Ingestion.* (September 1988)

5.7 EPA Federal Guidance Report 12 *External Exposure to Radionuclides in Air, Water, and Soil.* (September 1993)

5.8 ANL EAD TM-53 *Supplemental Analysis of Accident Sequences and Source Terms for Waste Treatment and Storage Operations and Related Facilities for the U.S. Department of Energy Waste Management Programmatic Environmental Impact Statement.*

5.9 40CFR190 EPA *Environmental Radiation Protection Standards for Nuclear Power Operations.*

5.10 EPA-400/R-17/001, PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents

5.11 NEI 99-01, Rev 6, *Development of Emergency Action Levels for Non-Passive Reactors*

5.12 FC08790, Rev 0, *Atmospheric Dispersion Factors (X/Qs) at the Decommissioning Exclusion Area Boundary (EAB) for Radiological Releases from the Fort Calhoun Station*

5.13 NUREG/CR-2919, *XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations* (September 1982)

5.14 RG 1.145 *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*.

6.0 METHOD OF ANALYSIS

6.1 X/Q Analysis

The X/Q is calculated using the same methodology as the ODCM (Ref 5.1). The long term atmospheric dispersion factor, X/Q, for normal effluent releases was used because the Unusual Event initiating condition from Ref. 5.11 was defined in terms of ODCM limits and calculations used to assess compliance with those limits use a non-accident dispersion factor.

The NRC approved computer code XOQDOQ, Version 2 (Ref. 5.13) was used to calculate the X/Q, as discussed in Section 6.4 of the ODCM (Ref 5.1).

6.2 Source Term

All Calculations performed follow guidance set forth by RG 1.183 (ref 5.5) and RG 1.195 (ref 5.4).

The Integrated Activity of Release (IAR) is equivalent to the 'source term' from the DOE guide on the Supplemental Analysis of Accident Sequences and Source Terms (ref 5.8). The equation for Source term is as follows:

$$\text{Source Term (IAR)} = MAR*RARF*LPF*DF$$

LPF = Leak Path Factor, which is a term where a reduction in the source during the event is credited. This value is not credited in the evaluation and thus the value is 1.0 (100% of the material at risk is available to be released).

DF = Damage Factor, which is a term for the percentage of damage to the cask. This value is not credited in the evaluation thus the value is 1.0 (100% damage i.e., the container is completely demolished releasing all of its contents instead of the more likely scenario of a crack forming and releasing only a fraction of its contents).

RARF = Respirable Airborne Release Fraction is a combination of the respirable fraction (RF) which involves estimating the Aerodynamic Mean Aerosol Diameter(AMAD) compared with the particle size that can be inhaled and remain in the human body. The RF is then multiplied by the Airborne Release Fraction (ARF) which is the fraction of the material that is released into the air. The ARF is a function of both the material composition inside the container and the type of damage that has occurred to the container. Reference 5.8 conveniently combines these values into 1 term, the RARF. The RARF for the types of materials that may be found in a HIC at FCS each have an RARF of 1E-03. Attachment 1 lists the RARFs for various materials.

MAR = Material At Risk which is the curie contents inside the container which are at risk of being released. It is conservatively assumes that all of the curie content inside the container is at risk of being released. In reality, only the loose surface contamination would be released on a dropped container. There is no way to predict the size, shape, surface area, thickness or

portion of the activity that is due to surface contamination vs fixed or internal contamination, thus the assumption is that the contents are of a granular or powder like substance where all of the activity of the material could potentially become airborne. This is likewise for resins, the radioactive particles are ionically bonded to the resin bead media.

Note on selection of MAR: If dose calculations were performed individually on each radionuclide, the MAR of each individual nuclide would have to be calculated. As this document uses the DCF_{eff} value to calculate dose, only one MAR is calculated. The MAR calculated is in units of µCi of Total Activity as these units are required by our first set of DCF_{eff}-TA with units of mrem/ µCi of Total Activity.

Source Term (IAR) = The amount of radioactive material that becomes airborne and is of a respirable AMAD that can be inhaled by reference man. The units of Source Term (IAR) is identical to the units of MAR as the other terms are unitless ratios. Source Term as defined by Ref 5.8 is identical to the Integrated Activity of Release (IAR) of Ref. 5.4.

6.3 Dose equations for EAB

Inhalation doses at unrestricted area boundary (CEDE and Organ):

$$\text{Inhalation Dose (mrem)} = DCF_{eff}\left(\frac{\text{mrem}}{\mu\text{Ci}}\right) * \text{IAR } (\mu\text{Ci}) * \frac{\chi}{Q}\left(\frac{\text{sec}}{\text{m}^3}\right) * \text{BR}\left(\frac{\text{m}^3}{\text{sec}}\right)$$

Submersion doses from standing in semi-infinite cloud at unrestricted area boundary (DDE and Skin):

$$\text{Submersion Dose (mrem)} = DCF_{eff}\left(\frac{\text{mrem}\cdot\text{m}^3}{\mu\text{Ci-sec}}\right) * \text{IAR } (\mu\text{Ci}) * \frac{\chi}{Q}\left(\frac{\text{sec}}{\text{m}^3}\right)$$

6.4 Description of DCF_{eff}:

The effective dose conversion factor (DCF_{eff}) described is the sum of the isotope ratio-weighted dose conversion factors. By using this value, the dose equations can be simplified by performing the calculation one time instead of performing the dose calculation ~21 times. Using these effective values also permit great simplification in reverse calculating curie content based on mrem value, as this also would have to be performed ~21 times and then the results summed. Using DCF_{eff} is an automatic sum of dose consequence of all nuclides including hard to detect nuclides. Additionally Am-241 in-growth is included in the decay correction of the values. The mathematical representation for DCF_{eff} is as follows:

DCF_{eff} Based on Total Activity (DCF_{eff}-TA)

$$DCF_{eff}\left(\frac{\text{mrem}}{\mu\text{Ci of Total Activity}}\right) = \sum DCF_i\left(\frac{\text{mrem}}{\mu\text{Ci of nuclide i}}\right) * \text{ratio}_i\left(\frac{\mu\text{Ci of nuclide i}}{\mu\text{Ci of Total Activity}}\right)$$

This equation also applies to submersion dose, simply by adding m^3/sec to the original DCF.

6.5 DCF values for each nuclide come directly from EPA FGR 11 (Ref 5.6) for inhalation doses and EPA FGR 12 for submersion dose. The units for inhalation DCFs in EPA FGR 11 are in Sv/Bq which can be converted to mrem/µCi by multiplying by 3.7E+09. Similarly submersion DCF values in EPA FGR 12 (Ref 5.7) are in Sv-m^3/Bq-sec which can be converted to mrem-m^3/µCi-sec by multiplying by the same factor. The converted units are listed in Att. 4

6.6 Ratios that are an input to DCF_{eff}

The method of determining the ratios of nuclide i to Total Activity is the same. Multiple lab reports are referenced. For resins, the actual lab reports are included in Att. 3. For plant mix ratios, the values can be found in FCS RP document FC-17-001 (ref 5.3). To determine the worst case ratio, each set of 10 CFR part 61 data ratios were first decay corrected (including ingrowth of Am-241 from Pu-241) to the same date. Ratios were then performed on each set of data then compared. Whichever set of data had the higher ratio was the ratio used in the 'bounding' ratio. Thus the ratio of the nuclides in the resins is not an average, but a conservative maximum. These ratios of each nuclide to the Total Activity was performed such that the values could be used in practical manner as well as theoretical.

7.0 NUMERIC ANALYSIS

7.1 X/Q:
XOQDOQ, Version 2 was run using a distance of 400 meters from the release point. The meteorological data from 2009 was used as this is the highest value that corresponds to the value in the ODCM (Ref 5.1).

The computer run results are contained in Attachment 5. The worst case X/Q from the computer run is 8.10E-05 sec/m^3. The computer output file is contained in Attachment 6.

7.2 MAR:
The MAR is the total activity in the HIC. For the resin container the total activity (or MAR) was calculated on the attached excel sheet for the analysis. The value for the MAR for the resin container was determined to be 2.62E+08μ**Ci.**

The value for the MAR for the plant mix container was assumed to be 1 μ**Ci**. This was used to find the contributions from each nuclide.

7.3 IAR: The IAR input to the dose equations were calculated in this manner.

$$IAR(\mu Ci)TA = HIC\ Total\ Activity\ (\mu Ci) * RARF * LPF * DF$$

Table 7.0

Contents	Ratio method	MAR(μCi)	RARF	LPF	DF	Source Term (IAR) (μCi)	Notes:
Resin	TA	2.62E+08	1.00E-03	1	1	2.62E+05	These values are from the actual resin HICs recently shipped. See Attachment 3.
Plant Mix	TA	1.00E+0	1.00E-03	1	1	1.00E-03	As no HIC with Plant Mix materials has been shipped, 1 μCi was used to determine IAR per 1μCi of container contents.

7.4 DCFeff equation:

DCFeff is dependent on the surrogate nuclide to which ratios are created. Additionally, since ratios differ between resin and plant mix, there needs be 2 sets of DCFeff. Thus the following equation was performed 16 times. The calculations are performed in Excel. The original DCFs along with their conversion to mrem/µCi can be found in Att. 4.

$$DCF_{eff} = \sum DCF_{ij} * ratio_i \text{ to surrogate}$$

7.5 Equation used to calculate ratios: ratios can be found in the attached excel sheet.

$$ratio_i = \frac{\mu Ci \text{ of nuclide } i}{\mu Ci \text{ of surrogate}}$$

7.6 Decay correction equation and Am-241 ingrowth equation. Half-lives and decay corrected ratios can be found in Att. 4.

$$A(t) = A(0)\, e^{-\lambda t}$$

$$\lambda(days^{-1}) = \frac{\ln(2)}{\text{half-life}(days)}$$

Since Pu-241 decays to Am-241, Am-241 activity will slowly build up to a maximum and then decay. No equilibrium is achieved as Am-241 has a longer half-life than Pu-241. The Activity for combined ingrowth and decay is as follows:

$$A_{Am241} = A_{Pu241}\left(\frac{\lambda_{Am241}}{\lambda_{Am241} - \lambda_{Pu241}}\right)\left(e^{-\lambda_{Pu241}*t} - e^{-\lambda_{Am241}*t}\right) + A_{Am241}\left(e^{-\lambda_{Am241}*t}\right)$$

7.7 <u>Dose equations</u>: Table 7.6.1 was calculated using these equations.

$$CEDE\ (mrem) = DCF_{eff\text{-}CEDE}\left(\frac{mrem}{\mu Ci}\right) * IAR\ (\mu Ci) * \frac{X}{Q}\left(\frac{sec}{m^3}\right) * BR\left(\frac{m^3}{sec}\right)$$

$$Bone\ (mrem) = DCF_{eff\text{-}bone}\left(\frac{mrem}{\mu Ci}\right) * IAR\ (\mu Ci) * \frac{X}{Q}\left(\frac{sec}{m^3}\right) * BR\left(\frac{m^3}{sec}\right)$$

$$DDE\ (mrem) = DCF_{eff\text{-}DDE}\left(\frac{mrem\text{-}m^3}{\mu Ci\text{-}sec}\right) * IAR\ (\mu Ci) * \frac{X}{Q}\left(\frac{sec}{m^3}\right)$$

$$Skin\ (mrem) = DCF_{eff\text{-}Skin}\left(\frac{mrem\text{-}m^3}{\mu Ci\text{-}sec}\right) * IAR\ (\mu Ci) * \frac{X}{Q}\left(\frac{sec}{m^3}\right)$$

$$TEDE\ (mrem) = CEDE\ (mrem) + DDE\ (mrem)$$

Table 7.6.1 Dose per Curies of Total Activity

	Resin			Plant Mix		
Curies of Total Activity (Ci)	Organ Dose -Bone (mrem)	Skin Dose (mrem)	TEDE (mrem)	Organ Dose -Bone (mrem)	Skin Dose (mrem)	TEDE (mrem)
1000	1.43E+01	9.16E-03	2.62E+00	8.62E+01	1.38E-02	7.10E+00

8.0 RESULTS

The total effective dose equivalent (TEDE) at the control area boundary after a drop of a High Integrity Container (HIC) containing resin with a 1000 curies of total activity with 20% conservatism applied is 2.62 mrem.

The total effective dose equivalent (TEDE) at the control area boundary after a drop of a High Integrity Container (HIC) containing plant components with a 1000 curies of total activity with 20% conservatism applied is 7.10 mrem.

9.0 CONCLUSION

The conclusion and interpretation of the results show that the expected resultant dose from a radioactive waste handling event (dropped HIC) of 1,000 Curies of total activity for both the isotopic mix contained in resins or the isotopic mix contained in other plant components are less than the 1 Rem Criterion for the EPA PAGs and less than the 10 mRem criterion for the NEI 99-01 guidance at the control area boundary.

10.0 ATTACHMENTS

10.1 Attachment 1 - RARF Factors (1 page)

10.2 Attachment 2 - HIC Specifications (13 pages)

10.3 Attachment 3 - 10 CFR Part 61 Analyses (4 pages)

Two separate copies of results from resin analyses are included:

- LIMS #: 7148167 reflects the results from Resin shipment number 17-20. The analysis was performed on 10/17/2011.
- LIMS #: L48167 reflects the results from Resin shipment number 17-16. The analysis was performed on 01/20/2017.

10.4 Attachment 4 - DCF from FGR and mrem/μCi conversion of them (2 pages)

10.5 Attachment 5 - XOQDOQ run Results (1 page)

10.6 Attachment 6 – XOQDOQ output file (imbedded txt file)

TABLE D.2 WM PEIS Waste RARFs for LLW, LLMW, and TRUW Storage and Handling

	Stresses[a]							
	Mechanically Driven Releases			Fire		Explosively Driven Releases		
Categories/Subcategories[b]	Free-Fall Spill	Crush Impact	Over-pressurization	Small	Large	Blast	Shock	High Pressure
1. Org. combustible liq.								
a. Solutions	1E-4	1E-4	1E-4	1E-2	1E-1	1E-1	Mass TNT Eq.	6E-4
b. Slurries	4E-5	4E-5	1E-4	6E-5	6E-5	6E-5	Mass TNT Eq.	6E-4
2. Aqueous liquids								
a. Solutions	1E-4	1E-4	1E-4	2E-3	2E-3	1E-4	Mass TNT Eq.	2E-3
b. Slurries	4E-5	4E-5	1E-4	2E-3	2E-3	1E-4	Mass TNT Eq.	6E-4
3. Powders, noncombust.	6E-4[c]	6E-4[c]	2E-3[c]	6E-5[c]	6E-5[c]	7E-2[c]	0.2 [mass TNT Eq.][c]	7E-2[c]
4. Combustible solids								
a. DAW	1E-3	1E-3	1E-3	5E-4	2E-2[d]	5E-4	Mass TNT Eq.	1E-3
b. Plastics (incl. elast.)	1E-3	1E-3	1E-3	1E-2	2E-2[d]	1E-2	Mass TNT Eq.	1E-3
c. Cellulosics	1E-3	1E-3	1E-3	1E-2	2E-2[d]	1E-2	Mass TNT Eq.	1E-3
d. Polystyrene	1E-3	1E-3	1E-3	1E-2	2E-2[d]	1E-2	Mass TNT Eq.	1E-3
5. Metals								
a. Inert	1E-3	1E-3	2E-3	6E-5	6E-5	7E-2	Mass TNT Eq.	7E-2
b. Reactive	1E-3	1E-3	2E-3	6E-5[c]	1E-2[c]	7E-2	Mass TNT Eq.[c]	7E-2
6. Noncomb. aggregate[e]	2E-11 ρgh	2E-11 ρgh	1E-3	D.2.1.1.9	D.2.1.1.9	7E-2	Mass TNT Eq.	7E-2
7. HEPA filters	5E-4	5E-4	1E-2	1E-4	1E-4	1E-2	2E-6	1E-2

[a] Stresses as defined in Mueller et al. (1996).

[b] Physical forms as defined by Mueller (1994).

[c] Of material; all others must be multiplied by the concentration of the material-of-concern in the matrix.

[d] See Section D.2.2.1.

[e] ρ = waste density, g = gravitational constant (981 cm/s^2), h = height.

Attachment 2- HIC Specs

Liner type	Total volume (CF)	Max internal volume (CF)	Usable volume (CF) dewater/solid	Gross weight (lbs)	Empty Weight (lbs)
PL 8-120 FR	120.3	107.6	101	10000	650
PL 10-160 FR	145.7	129.8	121.8	9500	750

Shipment	Waste Volume		Waste Weight	
	(cf)	(cc)	(lbs)	(g)
17-16	100	2.83E+06	3350	1.52E+06
17-20	100	2.83E+06	3100	1.41E+06

Liner Codes

Suffix		Suffix	
C	Conical	EXM	Expanded Metal Bottom
BT	Barrel Top	F	Flat
CMT	Cement	IF	Internal Foam
E	Epoxy	MT	Empty
EDX	Ecodex	P	Powdex
WM	Wide Mouth	R	Bead Resin

Polyethylene Liners Volumes/Weights

Polyethylene Liners	Burial Vol (CF)	Max Internal Vol (CF)	Usable Vol (CF) Dewater/Solid	Gross Weight (lbs)	Empty Weight (lbs)
PL 6-80 MT	83.4	73.3	NA	7,500	500
PL 6-80 MTIF	83.4	64.1	NA	7,500	525
PL 6-80 FR	83.4	73.3	64/NA	7,500	550
PL 6-80 FP/FEDX	83.4	73.3	62/NA	7,500	625
PL 8-120 MT	120.3	107.6	NA	10,000	600
PL 8-120 MTIF	120.3	95.8	NA	10,000	625
PL 8-120 FR	120.3	107.6	101/NA	10,000	650
PL 8-120 FP/FEDX	120.3	107.6	99/NA	10,000	725
PL 8-120 CMT	120.3	107.6	101/NA	14,000	720
PL 10-160 MT	145.7	129.8	NA	9,500	700
PL 10-160 MTIF	145.7	115.6	NA	9,500	735
PL 10-160 FR	145.7	129.8	121.8	9,500	750
PL 10-160 FP/FEDX	145.7	129.8	119.4	9,500	825
PL 10-160 CMT	145.7	129.8	121.8	16,000	820
PL 14-170 MT	170.8	150.3	NA	10,800	800
PL 14-170 MTIF	170.8	134.9	NA	10,800	850
PL 14-170 FR	170.8	150.3	141/NA	10,800	850

Polyethylene Liners	Burial Vol (CF)	Max Internal Vol (CF)	Usable Vol (CF) Dewater/Solid	Gross Weight (lbs)	Empty Weight (lbs)
PL 14-170 FP/FEDX	170.8	150.3	138/NA	10,800	1000
PL 14-170 CMT	170.8	150.3	141/NA	18,000	1000
PL 14-195 MT	194.1	171.4	NA	12,200	850
PL 14-195 MTIF	194.1	154.6	NA	12,200	900
PL 14-195 FR	194.1	171.4	162/NA	12,200	900
PL 14-195 FP/FEDX	194.1	171.4	159/NA	12,200	1050
PL 14-195 CMT	194.0	171.4	N/A	19,000	1,000
PL 14-215 MT	205.8	189.2	NA	13,000	1200
PL 14-215 MTIF	205.8	171.7	NA	13,000	1250
PL 14-215 FR	205.8	189.2	177/NA	13,000	1250
PL 14-215 FP/FEDX	205.8	189.2	174/NA	13,000	1400
PL 21-300 MT	314.2	285.1	NA	18,750	1100
PL 21-300 MTIF	314.2	262.1	NA	18,750	1175
PL 21-300 FR	314.2	285.1	269/NA	18,750	1150
PL 21-300 FP/FEDX	314.2	285.1	264/NA	18,750	1350
60 GAL OVERPACK	10.2	8.4	NA	1,200	125
SMALL OVERPACK	28.0	23.8	NA	2,500	250
246 GAL O.P.	36.5	32.7	NA	2,500	285
MEDIUM O.P.	38.3	33.5	NA	2,500	305

Suffix	
HUH	HUHIC
EL	Envirolene

NOTE: All liners below are empty – without internals.

	Burial Vol (CF)	Max Internal Vol (CF)	Usable Vol (CF) Dewater/Solid	Gross Wt (lbs)	Empty Wt (lbs)	Empty Wt (lbs) (w/Std Lift Assembly)	Empty Wt (lbs) (w/Grapple Lift Assembly)
Radlok Series							
Radlok 500	135.8	111.0	110.7	9,500	680		
Radlok 179	178.9	156.8	156.4	18,500	1,090		
Radlok 195	195.2	172.8	172.4	18,500	1,150		
Envirolene Series							
EL-50	51.2	41.0		4,200		600	909
EL-142	132.4	113.6		8,250		882	1,456
EL-190	174.3	150.6		14,800		1,025	1,843
EL-210	202.1	176.7		17,300		1,138	2,072

Polyethylene Liner Dimensions

Polyethylene Liners	Max Height (inches)	Max Diameter Standard Lift Bail with Centering Tabs (inches)	Max Diameter Standard Lift Bail without Centering Tabs (inches)	Max Diameter Grapple Lift Bail with Centering Tabs (inches)	Max Diameter Grapple Lift Bail without Centering Tabs (inches)	Max Diameter Grapple Stack Lift Bail with Centering Tabs (inches)	Max Diameter Grapple Stack Lift Bail without Centering Tabs (inches)
PL 6-80 MT	57.5	58.75	58	59.5	58	60	58.5
PL 6-80 MTIF	57.5	58.75	58	59.5	58	60	58.5
PL 6-80 FR	57.5	58.75	58	59.5	58	60	58.5
PL 6-80 FP/FEDX	57.5	58.75	58	59.5	58	60	58.5
PL 8-120 MT	74.5	61.75	61	N/A	61	N/A	61.5
PL 8-120 MTIF	74.5	61.75	61	N/A	61	N/A	61.5
PL 8-120 FR	74.5	61.75	61	N/A	61	N/A	61.5
PL 8-120 FP/FEDX	74.5	61.75	61	N/A	61	N/A	61.5
PL 8-120 CMT	74.5	61.75	61	N/A	61	N/A	61.5
PL 10-160 MT	76.25	67.25	65.5	67.5	65.5	N/A	N/A
PL 10-160 MTIF	76.25	67.25	65.5	67.5	65.5	N/A	N/A
PL 10-160 FR	76.25	67.25	65.5	67.5	65.5	N/A	N/A
PL 10-160 FP/FEDX	76.25	67.25	65.5	67.5	65.5	N/A	N/A
PL 10-160 CMT	76.25	67.25	65.5	67.5	65.5	N/A	N/A
PL 14-170 MT	72.75	75.25	73.5	75.5	73.5	75.5	74
PL 14-170 MTIF	72.75	75.25	73.5	75.5	73.5	75.5	74
PL 14-170 FR	72.75	75.25	73.5	75.5	73.5	75.5	74
PL 14-170 FP/FEDX	72.75	75.25	73.5	75.5	73.5	75.5	74
PL 14-170 CMT	72.75	75.25	73.5	75.5	73.5	75.5	74
PL 14-195 MT	79.5	76.75	75	77	75	77	75
PL 14-195 MTIF	79.5	76.75	75	77	75	77	75
PL 14-195 FR	79.5	76.75	75	77	75	77	75
PL 14-195 FP/FEDX	79.5	76.75	75	77	75	77	75
PL 14-195 CMT	79.5	76.75	75	77	75	77	75

Polyethylene Liners	Max Height (inches)	Max Diameter Standard Lift Bail with Centering Tabs (inches)	Max Diameter Standard Lift Bail without Centering Tabs (inches)	Max Diameter Grapple Lift Bail with Centering Tabs (inches)	Max Diameter Grapple Lift Bail without Centering Tabs (inches)	Max Diameter Grapple Stack Lift Bail with Centering Tabs (inches)	Max Diameter Grapple Stack Lift Bail without Centering Tabs (inches)
PL 14-215 MT	79.5	N/A	76.125	N/A	76.375	N/A	N/A
PL 14-215 MTIF	79.5	N/A	76.125	N/A	76.375	N/A	N/A
PL 14-215 FR	79.5	N/A	76.125	N/A	76.375	N/A	N/A
PL 14-215 FP/FEDX	79.5	N/A	76.125	N/A	76.375	N/A	N/A
PL 21-300 MT	108.5	82.75	81	82.75	81.5	82.5	82
PL 21-300 MTIF	108.5	82.75	81	82.75	81.5	82.5	82
PL 21-300 FR	108.5	82.75	81	82.75	81.5	82.5	82
PL 21-300 FP/FEDX	108.5	82.75	81	82.75	81.5	82.5	82
60 GAL OVERPACK	35	N/A	25.5	N/A	N/A	N/A	N/A
SMALL OVERPACK	57	N/A	34	N/A	N/A	N/A	N/A
246 GAL O.P.	74.25	N/A	34	N/A	N/A	N/A	N/A
MEDIUM O.P.	78	N/A	34	N/A	N/A	N/A	N/A

	Max Height Include slings (inches)	Outside Diameter (inches)	Outside Height (inches)	Manway Opening Diameter (inches)	Fill Port Opening Diameter (inches)	Lift Assembly Outside Diameter w/Std Lift Assembly (inches)	Lift Assembly Outside Diameter w/Grapple Lift Assembly (inches)
Radlok Series							
Radlok 500	115.875	64.5	71.875	16.0	8.375		
Radlok 179	119.50	73.50	72.875	19.250	10.125		
Radlok 195	126.625	73.50	79.50	19.250	10.125		
Envirolene Series							
EL-50	74.0	46.5	51.0	19.8		64.5	65.0
EL-142	101.0	64.0	70.0	19.8		73.5	74.5
EL-190	107.0	73.0	71.0	19.8		75.5	76.5
EL-210	114.0	75.0	78.0	19.8		N/A	N/A

Note: The exact liner height and diameter can be verified by contacting Energy*Solutions* Liner Operations. To verify the Dimensions the liner serial number will be required.

Liner Codes

Prefix		Suffix	
L	Carbon Steel	C	Conical
R	Resin (Bead)	BT	Barrel Top
P	Powdex	CMT	Cement
EDX	Ecodex	E	Epoxy
F	Flat	EXM	Expanded Metal Bottom
MT	Empty		

Steel Liners

Steel Liners	Height (inches)	Diameter (inches)	Burial Vol (CF)	Max Internal Vol (CF)	Usable Vol (CF)	Gross Wt (lbs)	Empty Wt (lbs)
L 6-80 MT	57.0	58.0	87.2	82.9	NA	9,900	1,000
L 6-80 CMT	57.0	58.0	87.2	82.9	NA/80	9,900	1,150
L 6-80 IN-SITU	57.0	58.0	87.2	49.8	64/NA	9,900	3,500
L 6-80 FP	57.0	58.0	87.2	82.9	62/NA	9,900	1,050
L 6-80 FP/FEDX	57.0	58.0	87.2	82.9	NA	9,900	1,225
L 8-120 MT	74.0	61.0	125.2	120.2	NA	14,500	1,200
L 8-120 CMT	74.0	61.0	125.2	120.2	NA/117	14,500	1,350
L 8-120 IN-SITU	74.5	61.0	126.0	80.3	NA	14,500	4,200
L 8-120 FR	74.0	61.0	125.2	120.2	114/NA	14,500	1,250
L 8-120 FP/FEDX	74.0	61.0	125.2	120.2	112/NA	14,500	1,325
L 14-170 TVA	73.25	69.0	158.5	151.3	NA/147	20,750	1,450
L 14-170 MT	71.375	74.5	180.1	172.7	NA	20,750	1,550
L 14-170 CMT	71.375	74.5	180.1	172.7	NA/168	20,750	1,750
L 14-170 IN-SITU	74.0	74.5	186.7	66.1	NA	20,750	TBD
L 14-170 FR	71.375	74.5	180.1	172.7	163/NA	20,750	1,600
L 14-170 FP/FEDX	71.375	74.5	180.1	172.7	160/NA	20750	1750

Steel Liners	Height (inches)	Diameter (inches)	Burial Vol (CF)	Max Internal Vol (CF)	Usable Vol (CF)	Gross Wt (lbs)	Empty Wt (lbs)
L 14-195 MT	79.0	76.0	207.4	199.6	NA	23700	1650
L 14-195 CMT	79.0	76.0	207.4	199.6	NA/195	23700	1850
L 14-195 IN-SITU	78.5	76.0	206.1	138.5	NA	23700	6300
L 14-195 FR	79.0	76.0	207.4	199.6	190/NA	23700	1700
L 14-195 FP/FEDX	79.0	76.0	207.4	199.6	187/NA	23700	1850
L 21-300 MT	108.0	82.0	330.1	320.6	NA	27250	2200
L 21-300 FP/FEDX	108.0	82.0	330.1	320.6	303/NA	27250	2450
1-13G INSERT	43.75	19.25	7.4	6.2	6.2/NA	5000	300
1-13 LINER	51.375	25.0	14.6	14.6	12/NA	5000	500
3-55 LINER	109.25	34.0	57.4	57.4	52/NA	7800	945
PV-24-79	78.875	24.0	20.7	18.3	16/NA	1900	570
PV-24-72	72.0	24.0	18.8	16.6	14/NA	1750	530
PV-24-51	51.0	24.0	13.4	11.3	9/NA	1150	415

Prefix		Suffix	
ES	Enviro Steel	O.T.	Open Top

Carbon & Stainless Steel	Height (inches)	Diameter (inches)	Burial Vol (CF)	Max Internal Vol (CF)	Gross Wt (lbs)	Empty Wt (lbs)
ES-50	51.00	47.25	52.00	49.30	4,200	250
ES-142	69.75	63.50	128.3	122.20	10,000	1,100
ES-190	71.00	72.50	170.2	162.40	16,800	1,285
ES-210	78.25	74.75	199.4	191.00	20,000	1,475
ES-210 O.T.	79.00	75.00	202.00	189.50	20,000	1,710

Steel Wide-Mouth Liners

Non-Stackable, Wide Mouth Steel Liner, Standard and In-Situ, Slings or Grapple Large Diameter Round Lid, Bolt On

Steel Wide Mouth	Height (inches)	Diameter (inches)	Opening Diameter (inches)	Empty Weight (lbs)	Gross Weight (lbs)	Max Internal (CF)
L6-80	57	58	47 7/8	*	*	82.9
L8-120	74	61	50 7/8	*	*	120.2
L14-170	72 1/4	74 1/2	64 3/8	*	*	172.7
L14-195	78 1/2	76	65 7/8	*	*	199.6

Non-Stackable, Steel, Square Lid (low profile) Liner, Slings Only, Bolt on lid

Steel Wide Mouth	Height (inches)	Diameter (inches)	Opening Square (in^2)	Empty Weight (lbs)	Gross Weight (lbs)	Max Internal (CF)
L6-80	54	58	32 x 32	*	*	82.9
L8-120	71 1/2	61	36 x 36	*	*	120.2
L14-170	67 7/8	74 1/2	48 x 48	*	*	172.7
L14-195	75 1/2	76	48 x 48	*	*	199.6
L21-300	104 1/2	82	56 x 56	*	*	320.6

*Gross and empty weights are very close to the same size MT container configuration.

Non-Stackable, Steel Wide Mouth Liner, Round Raised Lid, Bolt On, Slings Only

Steel Wide Mouth	Height (inches)	Diameter (inches)	Opening Diameter (inches)	Empty Weight (lbs)	Gross Weight (lbs)	Max Internal (CF)
L6-80	52	58	43	1102	10,052	82.5
L8-120	69 1/2	61	46	1275	14,675	118
L14-170	67 3/8	74 1/2	59 1/2	1830	21,030	175
L14-195	73 1/2	76	61	1945	22,050	198
L21-300	102 1/2	82	67	2550	27,600	317

Liner & Cask Compatibility Table

Liners	Max Height (inches)	Max Diameter (inches)	Burial Volume (CF)	Cask Compatibility	Cask Shield Equiv & Max R/hr
PL 6-80	57.5	60	83.4	6-80	5.0"/1.860R/hr
EL-50	51.0	47.0	51.2	6-80	5.0"/1.860R/hr
PL 8-120	74.5	61.75	120.3	8-120A or B	4.5"/880R/hr.
EL-142	70.0	64.5	132.4	EL-142 – will not fit in 8-120 cask because of ht. 8-120 liner will not fit in OH-142 cask because of diameter. Difference bet. 10-160B interior hgt and liner is 7".	3.13"/78 R/hr. (10-160B)
PL 14-170	72.75	75.5	170.8	14-170	2.13"/15R/hr.
NUHIC-136	71	65	136.3	14-170 w/shoring	2.13"/15R/hr.
EL-190	71	73.5	174.3	14-170 series II or III	2.13"/15R/hr.
Radlok P-500	71.0	71.1	163.3	14-170 14-195 14-215 w/shoring	2.13"/15R/hr. 2.75"/21R/hr. 2.73"/20R/hr.
PL 14-195	79.5	76.75	194.1	14-195	2.75"/21R/hr.
EL-210	78.0	75.5	176.7	14-195 14-215	2.75"/21R/hr. 2.73"/20R/hr.
Radlok P-195	79.5	73.5	195.2	14-195 14-215	2.75"/21R/hr. 2.73"/20R/hr.
PL 14-215	79.5	76.75	205.8	14-215	2.73"/20R/hr.
EL-210	78.0	75.5	202.1	14-195 14-215	2.75"/21R/hr. 2.73"/20R/hr.
Radlok P-179	72.9	73.5	178.9	14-195 14-215	2.75"/21R/hr. 2.73"/20R/hr.
Radlok P-195	79.5	73.5	195.2	14-215	2.73"/20R/hr.
PL 21-300	108.5	82.75	314.2	21-300	1.5"/1.9R/hr.

*Note: If using grapple bails, or steel inserts, listed liner may not fit. Consult Transportation and Liner Operations prior to ordering a shipping cask.

Liners

Liner (OP-Overpack)	Max Height (inches)	Max Diameter (inches)	Opening Diameter (inches)	Max Internal Volume (CF)	Burial Volume (CF)	Empty Weight (lbs)
60 gal (OP)	35	25.5	NA	8.4	10.2	125
Small OP	57	34	NA	23.8	28	250
Medium OP	78	34	NA	33.5	38.3	305
246 gal OP	74.25	34	NA	32.7	36.5	285
PL 6-80	57.5	60	22.5	73.3	83.4	500
1-13G	43.75	19.25	NA	6.2	7.4	300
1-13	51.375	25	NA	12.5	14.6	500
3-55	72	24	NA	16.6	18.8	945
NUHIC-55 (OP)	41.62	31	27	14.8	18.8	65
EL-50	51.0	47.0	25	41	51.2	500
PL 8-120	74.5	61.75	22.5	107.6	120.3	600
EL-142	70.0	64.5	25	113.6	132.4	650
PL10-160	76.25	67.5	22.5	129.8	145.8	825
PL 14-170	72.75	75.25	22.5	150.3	170.8	800
NUHIC-136	71	65	22	127	136.3	600
EL-190	71	73.5	25	150.6	174.3	800
Radlok P-500	71.0	71.1			163.3	695
PL 14-195	79.5	76.75	22.5	171.4	194.1	850
PL 14-215	79.5	76.75	22.5	189.2	205.8	1200
EL-210	78.0	75.5	25		202.1	900
Radlok P-179	72.9	73.5			179.4	1090
Radlok P-195	79.5	73.5			195.7	1150
PL 21-300	108.5	82.75	22.5	285.1	314.2	1100

Cask and Liner Compatibility Table

Cask	Shielding Equiv/Max Rad Level Limit (R/hr)	Type	Internal Dimension (Height x Diameter) (inches)	Payload (lbs)	Compatible Liners
6-80-2	5.0/1,860	DOT-7A	58 x 59	7,500	6-80
6/100L	6.00/	DOT-7A	62 x 61	12,000	6-80
6/100H	6.00/	DOT-7A	62 x 61	12,000	6-80
8-120	4.5/880	DOT-7A	75 x 62	20,000	8-120, 6-80*
8-120	4.5/880	B	75 x 62	14,680	8-120, 6-80*
10/140	3.6/	DOT 7A	73 x 66	15,000	6-80*
10-140 MB	3.25/	B	73 x 66	15,000	6-80*
10-142	4.25/350	CoC Expired	72 x 66	10,000	6-80*
10-160	3.13/78	B	75 x 67	14,500	8-120, 6-80*
14-170-II 14-170 III	2.13/15 2.13/15	DOT-7A DOT-7A	73.25 x 75.5 73.25 x 75.5	14,000 17,500	14-170, 14-150*, 7-100*, 6-80*
14/190L	2.00/7	DOT-7A	73.38 x 75.5	20,000	14-170, 14-150*, 7-100*, 6-80*
14/190M	2.25/10	DOT-7A	73.38 x 75.5	20,000	14-170, 14-150*, 7-100*, 6-80*
14/190H	3.50/150	DOT-7A	73.38 x 75.5	20,000	14-170, 14-150*, 7-100*, 6-80*
14-190H	3.50/60	DOT-7A	73.38 x 75.5	20,000	14-170, 14-150*, 7-100*, 6-80*
14-195H	2.75/21	DOT-7A	80 x 77	17,700	14-215, 14-195, 14-170*, 14-150*, 8-120*, 7-100* 6-80*
14/210L	2.00/7	DOT-7A	80.25 x 77.25	20,000	14-215, 14-195, 14-170*, 14-150*, 8-120*, 7-100* 6-80*
14/210H	2.73/20	DOT-7A	80.25 x 77.25	20,000	14-215, 14-195, 14-170*, 14-150*, 8-120*, 7-100* 6-80*
14-215H	2.73/20	DOT-7A	80.25 x 77.25	20,000	14-215, 14-195, 14-170*, 14-150*, 8-120*, 7-100* 6-80*
21-300	1.5/1.9	DOT-7A	109 x 83	27,250	21-300, 14-215*14-195*, 14-170*, 14-150*, 8-120*, 7-100*, 6-80*

* Proper shoring must be installed in cask.

Attachment 3 – 10 CFR part 61 Analyses
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Report of Analysis

12/01/11 13:52

L48167

Omaha Public Power District
OM715-3PCFR61

TELEDYNE BROWN ENGINEERING, INC.

Mr. Mark Breuer

Sample ID: PRIMARY (CVCS) RESIN
Station:
Description:
LIMS Number: L48167-2

Collect Start: 10/01/2011 00:00
Collect Stop:
Receive Date: 10/17/2011

Matrix: Resin (RS)
Volume: 1.480E-01 G
% Moisture:

Radionuclide	SOP#	Activity Conc	Uncertainty 2 Sigma	MDC	Units	Run #	Aliquot Volume	Aliquot Units	Reference Date	Count Date	Count Time	Count Units	Flag Values	
C-14	2002	**7.36E-03**	6.10E-04		uci/g		.0049	g wet		11/29/11	60	M	+	
FE-55	2006	**1.99E+00**	6.57E-02		uci/g		.0046	g wet	10/01/11 00:00	11/28/11	600	Sec	+	Yes
GR-A	2008	**9.96E-02**	4.75E-03		uci/g		.0009	g wet		11/29/11	50	M	+	
GR-B	2008	**1.78E+01**	3.55E-01		uci/g		.0009	g wet		11/29/11	49	M	+	
H-3	2010	<		**1.65E-03**	uci/g		.072	g wet		11/30/11	20	M	U	
I-129	2012	<		**1.09E-04**	uci/g		.0098	g wet	10/01/11 00:00	11/29/11	900	Sec	U	No
NI-59	2013	**3.26E-01**	1.33E-02		uci/g		.0081	g wet	10/01/11 00:00	11/28/11	1200	Sec	+	Yes
NI-63	2011	**1.65E+01**	2.44E-02		uci/g		.0091	g wet		11/30/11	10	M	+	
SR-89	2018	**4.01E-02**	3.22E-03		uci/g		.0091	g wet	10/01/11 00:00	11/30/11	15	M	+	
SR-90	2018	**1.27E-01**	2.38E-03		uci/g		.0091	g wet	10/01/11 00:00	11/30/11	13	M	+	
TC-99	2021	**1.32E-02**	1.14E-03		uci/g		.0009	g wet		11/28/11	99.03	M	+	
BE-7	2007	<		**1.56E-01**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
K-40	2007	<		**4.92E-02**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
CR-51	2007	<		**1.69E-01**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
MN-54	2007	**6.41E-01**	1.26E-02		uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes
CO-58	2007	**3.00E+01**	3.19E-02		uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes
FE-59	2007	<		**3.27E-02**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
CO-60	2007	**5.90E+00**	1.77E-02		uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes
ZN-65	2007	**8.34E-02**	2.17E-02		uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes
NB-94	2007	<		**1.01E-02**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
NB-95	2007	<		**1.35E-02**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
ZR-95	2007	<		**2.37E-02**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
MO-99	2007	<		**2.63E+00**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
RU-103	2007	<		**2.05E-02**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
RU-106	2007	<		**1.44E-01**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
AG-110M	2007	**7.05E-02**	7.29E-03		uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes
SB-124	2007	<		**1.13E-02**	uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U*	No
SB-125	2007	**1.55E-01**	2.13E-02		uci/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes

Flag Values
U = Compound/Analyte not detected (< MDC) or less than 3 sigma
+ = Activity concentration exceeds MDC and 3 sigma; peak identified(gamma only)
U* = Compound/Analyte not detected. Peak not identified, but forced activity concentration exceeds MDC and 3 sigma
High = Activity concentration exceeds customer reporting value
Spec = MDC exceeds customer technical specification
L = Low recovery
H = High recovery

No = Peak not identified in gamma spectrum
Yes = Peak identified in gamma spectrum

**** Unless otherwise noted, the analytical results reported are related only to the samples tested in the condition they are received by the laboratory

MDC - Minimum Detectable Concentration

Bolded text indicates reportable value.

Attachment 3 – 10 CFR part 61 Analyses
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Report of Analysis

12/01/11 13:52

L48167

Omaha Public Power District
OM755-1PCFR61

TELEDYNE BROWN ENGINEERING, INC.
A Teledyne Technologies Company

Mr. Mark Breuer

Sample ID: PRIMARY (CVCS) RESIN
Station:
Description:
LIMS Number: L48167-2

Collect Start: 10/01/2011 00:00
Collect Stop:
Receive Date: 10/17/2011

Matrix: Resins (RS)
Volume: 3.400E+01 G
% Moisture:

Radionuclide	SOP#	Activity Conc	Uncertainty 2 Sigma	MDC	Units	Run #	Aliquot Volume	Aliquot Units	Reference Date	Count Date	Count Time	Count Units	Flag Values	
I-131	2007	<		**9.44E-02**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
CS-134	2007	**3.39E-01**	8.92E-03		uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes
CS-137	2007	**7.78E+00**	2.42E-02		uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	+	Yes
BA-140	2007	<		**1.09E-01**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
LA-140	2007	<		**1.77E-02**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
CE-141	2007	<		**1.74E-02**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
CE-144	2007	<		**4.79E-02**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
RA-226	2007	<		**1.97E-01**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U*	No
TH-232	2007	<		**4.12E-02**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
NP-237	2007	<		**2.27E-02**	uCi/g		.14	g wet	10/01/11 00:00	10/25/11	3600	Sec	U	No
AM-241 (AS)	2001	**2.51E-04**	3.67E-05		uCi/g		.0045	g wet		11/21/11	60000	sec	+	
CM-242 (AS)	2001	**2.33E-05**	9.34E-06		uCi/g		.0046	g		11/21/11	60000	sec	+	
CM-243/244 (AS)	2001	**3.40E-04**	4.51E-05		uCi/g		.0046	g		11/21/11	60000	sec	+	
NP-237 (AS)	2001	<		**5.86E-06**	uCi/g		.0045	g		11/17/11	60004	sec	U	
PU-238 (AS)	2001	**2.28E-04**	2.84E-05		uCi/g		.0046	g wet		11/17/11	60004	sec	+	
PU-239/240 (AS)	2001	**6.69E-05**	1.43E-05		uCi/g		.0045	g wet		11/17/11	60004	sec	+	
PU-241	2001	**1.38E-03**	7.76E-04		uCi/g		.0045	g wet		11/18/11	60	M	+	
U-233/234 (AS)	2001	<		**7.54E-06**	uCi/g		.0046	g wet		11/30/11	60002	sec	U	
U-235 (AS)	2001	<		**2.81E-06**	uCi/g		.0036	g wet		11/30/11	60002	sec	U	
U-238 (AS)	2001	<		**9.89E-06**	uCi/g		.0046	g wet		11/30/11	60002	sec	U	

Flag Values
U = Compound/Analyte not detected (< MDC) or less than 3 sigma
\+ = Activity concentration exceeds MDC and 3 sigma, peak identified (gamma only)
U* = Compound/Analyte not detected. Peak not identified, but forced activity concentration exceeds MDC and 3 sigma
High = Activity concentration exceeds customer reporting value
Spec = MDC exceeds customer technical specification
L = Low recovery
H = High recovery

Bolded text indicates reportable value.

No = Peak not identified in gamma spectrum
Yes = Peak identified in gamma spectrum

**** Unless otherwise noted, the analytical results reported are related only to the samples tested in the condition they are received by the laboratory.

MDC - Minimum Detectable Concentration

Attachment 3 – 10 CFR part 61 Analyses
Page 3 of 4

Report of Analysis

02/03/17 11:36

L71393

Omaha Public Power District

OM755-3PCFR61

TELEDYNE BROWN ENGINEERING, INC.
A Teledyne Technologies Company

Sample ID: **PRIMARY RESIN SAMPLE**	Collect Start: 01/08/2016 00:00	Matrix: Resins	(RS)
Station:	Collect Stop:	Volume:	
Description:	Receive Date: 01/20/2017	% Moisture:	
LIMS Number: L71393-1			

Radionuclide	SOP#	Activity Conc	Uncertainty 2 Sigma	MDC	Units	Run #	Aliquot Volume	Aliquot Units	Reference Date	Count Date	Count Time	Count Units	Flag Values	
C-14	2002	**7.73E-03**	1.87E-03		uCi/g		.0008	g wet		01/31/17	30	M	+	
FE-55	2006	**4.43E+00**	2.92E-01		uCi/g		.0016	g wet	01/08/16 00:00	01/31/17	300	Sec	+	Yes
H-3	2010	<		**1.95E-02**	uCi/g		.0003	g wet		01/31/17	5	M	U	
I-129	2012	<		**5.19E-04**	uCi/g		.0016	g wet	01/08/16 00:00	01/31/17	3545	Sec	U	No
NI-59	2013	**2.12E-01**	2.72E-02		uCi/g		.0031	g wet	01/08/16 00:00	01/31/17	600	Sec	+	Yes
NI-63	2013	**1.46E+01**	5.26E-02		uCi/g		.0031	g wet		02/02/17	5	M	+	
SR-89	2018	<		**1.69E+00**	uCi/g		.0031	g wet	01/08/16 00:00	02/03/17	15	M	U	
SR-90	2018	**2.68E-01**	8.04E-03		uCi/g		.0031	g wet	01/08/16 00:00	02/03/17	15	M	+	
TC-99	2021	<		**5.33E-03**	uCi/g		.0023	g wet		01/31/17	30	M	U	
BE-7	2007	<		**3.06E+01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U*	No
K-40	2007	<		**4.37E-02**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
CR-51	2007	<		**1.97E+03**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
MN-54	2007	**6.21E-02**	3.09E-02		uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	+	Yes
CO-57	2007	<		**2.65E-02**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
CO-58	2007	<		**6.69E-01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
FE-59	2007	<		**6.96E+00**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
CO-60	2007	**7.96E+00**	4.04E-02		uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	+	Yes
ZN-65	2007	<		**1.20E-01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
NB-94	2007	<		**1.31E-02**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
NB-95	2007	<		**9.11E-01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
ZR-95	2007	<		**1.63E+00**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
RU-103	2007	<		**1.75E+01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
RU-106	2007	<		**3.19E-01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
AG-110M	2007	<		**7.48E-02**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
SB-124	2007	<		**6.24E-01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No
SB-125	2007	<		**8.59E-02**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U	No

Flag Values
U = Compound/Analyte not detected (< MDC) or less than 3 sigma
+ = Activity concentration exceeds MDC and 3 sigma; peak identified(gamma only)
U* = Compound/Analyte not detected. Peak not identified, but forced activity concentration exceeds MDC and 3 sigma
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MDC - Minimum Detectable Concentration

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TBE-ROA002

Attachment 3 – 10 CFR part 61 Analyses
Page 4 of 4

Report of Analysis
02/03/17 21:36

L71393

Omaha Public Power District
OM755-3PCFR61

TELEDYNE BROWN ENGINEERING, INC.
A Teledyne Technologies Company

Mr.David Conn

Sample ID: PRIMARY RESIN SAMPLE
Station:
Description:
LIMS Number: L71393-1

Collect Start: 01/08/2016 00:00
Collect Stop:
Receive Date: 01/20/2017

Matrix: Resin (RS)
Volume:
% Moisture:

Radionuclide	SOPs	Activity Conc	Uncertainty 2 Sigma	MDC	Units	Run #	Aliquot Volume	Aliquot Units	Reference Date	Count Date	Count Time	Count Units	Flag Values		
I-131	2007	<		**3.98E+12**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U		No
CS-134	2007	**3.22E-01**	1.74E-02		uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	+		Yes
CS-137	2007	**2.86E+01**	8.12E-02		uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	+		Yes
BA-140	2007	<		**6.42E+07**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U		No
LA-140	2007	<		**4.56E+06**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U		No
CE-141	2007	<		**6.19E+01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U		No
CE-144	2007	<		**1.95E-01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U		No
RA-226	2007	<		**3.79E-01**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U		No
TH-232	2007	<		**7.26E-02**	uCi/g		.0395	g wet	01/08/16 00:00	01/24/17	3600	Sec	U		No
AM-241 (AS)	2001	**7.16E-04**	8.25E-05		uCi/g		.0016	g wet		02/01/17	60001	sec	+		
CM-242 (AS)	2001	**4.14E-05**	1.62E-05		uCi/g		.0016	g		02/01/17	60001	sec	+		
CM-243/244 (AS)	2001	**9.43E-04**	9.88E-05		uCi/g		.0016	g		02/01/17	60001	sec	+		
PU-238 (AS)	2001	**8.69E-04**	1.44E-04		uCi/g		.0016	g wet		02/03/17	60001	sec	+		
PU-239/240 (AS)	2001	**1.59E-04**	4.60E-05		uCi/g		.0016	g wet		02/03/17	60002	sec	+		
PU-241	2001	**3.02E-02**	4.70E-03		uCi/g		.0016	g wet		02/03/17	10	M	+		

Flag Values
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MDC - Minimum Detectable Concentration

Attachment 4 - DCF from FGR and mrem/μCi Conversion
Page 1 of 2

EPA FGR 11, Table 2.1 - Inhalation Dose Factors						
	Conversion	3.70E+09	mrem-Bq/Sv-uCi			
Nuclide	CEDE Sv/Bq	Bone Sv/Bq	Lung Sv/Bq	CEDE mrem/uCi	Bone mrem/uCi	Lung mrem/uCi
C-14	5.64E-10	5.64E-10	5.64E-10	2.09E+00	2.09E+00	2.09E+00
Fe-55	7.26E-10	5.14E-10	1.06E-09	2.69E+00	1.90E+00	3.92E+00
Ni-59	3.58E-10	3.51E-10	1.20E-09	1.32E+00	1.30E+00	4.44E+00
Ni-63	8.39E-10	8.22E-10	3.07E-09	3.10E+00	3.04E+00	1.14E+01
Sr-89	1.12E-08	8.37E-09	8.35E-08	4.14E+01	3.10E+01	3.09E+02
Sr-90	3.51E-07	7.27E-07	2.86E-06	1.30E+03	2.69E+03	1.06E+04
Tc-99	2.25E-09	4.52E-11	1.67E-08	8.33E+00	1.67E-01	6.18E+01
Cr-51	9.03E-11	2.74E-11	5.34E-10	3.34E-01	1.01E-01	1.98E+00
Mn-54	1.81E-09	2.56E-09	6.66E-09	6.70E+00	9.47E+00	2.46E+01
Co-58	2.94E-09	6.93E-10	1.60E-08	1.09E+01	2.56E+00	5.92E+01
Fe-59	4.00E-09	2.91E-09	1.38E-08	1.48E+01	1.08E+01	5.11E+01
Co-60	5.91E-08	1.35E-08	3.45E-07	2.19E+02	5.00E+01	1.28E+03
Zn-65	5.51E-09	3.36E-09	2.10E-08	2.04E+01	1.24E+01	7.77E+01
Nb-95	1.57E-09	2.42E-09	8.32E-09	5.81E+00	8.95E+00	3.08E+01
Zr-95	6.39E-09	1.03E-07	4.07E-08	2.36E+01	3.81E+02	1.51E+02
Ag-110m	2.17E-08	5.19E-09	1.20E-07	8.03E+01	1.92E+01	4.44E+02
Sb-124	6.80E-09	3.41E-09	4.14E-08	2.52E+01	1.26E+01	1.53E+02
Sb-125	3.30E-09	2.73E-09	2.17E-08	1.22E+01	1.01E+01	8.03E+01
Cs-134	1.25E-08	1.10E-08	1.18E-08	4.63E+01	4.07E+01	4.37E+01
Cs-137	8.63E-09	7.94E-09	8.82E-09	3.19E+01	2.94E+01	3.26E+01
Am-241	1.20E-04	2.17E-03	1.84E-05	4.44E+05	8.03E+06	6.81E+04
Cm-242	4.67E-06	4.87E-05	1.55E-05	1.73E+04	1.80E+05	5.74E+04
Cm-243/4	8.30E-05	1.47E-03	1.94E-05	3.07E+05	5.44E+06	7.18E+04
Pu-238	1.06E-04	1.90E-03	3.20E-04	3.92E+05	7.03E+06	1.18E+06
Pu-239/40	1.16E-04	2.11E-03	3.23E-04	4.29E+05	7.81E+06	1.20E+06
Pu-241	2.23E-06	4.20E-05	3.18E-06	8.25E+03	1.55E+05	1.18E+04
Nb-94	1.12E-07	1.97E-08	7.48E-07	4.14E+02	7.29E+01	2.77E+03
I-129	4.69E-08	1.38E-10	3.14E-10	1.74E+02	5.11E-01	1.16E+00
Ra-226	2.32E-06	7.59E-06	1.61E-05	8.58E+03	2.81E+04	5.96E+04
Co-57	2.45E-09	4.52E-10	1.69E-08	9.07E+00	1.67E+00	6.25E+01
Ce-144	1.01E-07	4.54E-08	7.91E-07	3.74E+02	1.68E+02	2.93E+03
H-3	1.73E-11	1.73E-11	1.73E-11	6.40E-02	6.40E-02	6.40E-02

Attachment 4 - DCF from FGR and mrem/μCi Conversion

Page 2 of 2

EPA FGR 12, Table III.1 - Submersion Dose Factors				
	Conversion	3.70E+09	mrem-m3-Bq/Sv-uCi-sec	
Nuclide	DDE Sv-m3 /Bq-sec	Skin Sv-m3 /Bq-sec	DDE mrem-m3 /uCi-sec	Skin mrem-m3 /uCi-sec
C-14	2.24E-19	2.43E-16	8.29E-10	8.99E-07
Fe-55	0.00E+00	0	0.00E+00	0.00E+00
Ni-59	0	0	0.00E+00	0.00E+00
Ni-63	0	0	0.00E+00	0.00E+00
Sr-89	7.73E-17	3.69E-14	2.86E-07	1.37E-04
Sr-90	7.53E-18	9.2E-15	2.79E-08	3.40E-05
Tc-99	1.62E-18	2.74E-15	5.99E-09	1.01E-05
Cr-51	1.51E-15	1.75E-15	5.59E-06	6.48E-06
Mn-54	4.09E-14	4.67E-14	1.51E-04	1.73E-04
Co-58	4.76E-14	5.58E-14	1.76E-04	2.06E-04
Fe-59	5.97E-14	7.13E-14	2.21E-04	2.64E-04
Co-60	1.26E-13	1.45E-13	4.66E-04	5.37E-04
Zn-65	2.9E-14	3.29E-14	1.07E-04	1.22E-04
Nb-95	3.74E-14	4.3E-14	1.38E-04	1.59E-04
Zr-95	3.60E-14	4.5E-14	1.33E-04	1.67E-04
Ag-110m	1.36E-13	1.57E-13	5.03E-04	5.81E-04
Sb-124	9.15E-14	1.26E-13	3.39E-04	4.66E-04
Sb-125	2.02E-14	2.65E-14	7.47E-05	9.81E-05
Cs-134	7.57E-14	9.45E-14	2.80E-04	3.50E-04
Cs-137	7.74E-18	8.63E-15	2.86E-08	3.19E-05
Am-241	8.18E-16	1.28E-15	3.03E-06	4.74E-06
Cm-242	5.69E-18	4.29E-17	2.11E-08	1.59E-07
Cm-243/4	5.88E-15	9.79E-15	2.18E-05	3.62E-05
Pu-238	4.88E-18	4.09E-17	1.81E-08	1.51E-07
Pu-239/40	4.75E-18	3.92E-17	1.76E-08	1.45E-07
Pu-241	7.25E-20	1.17E-19	2.68E-10	4.33E-10
Nb-94	7.70E-14	9.52E-14	2.85E-04	3.52E-04
I-129	3.80E-16	1.10E-15	1.41E-06	4.07E-06
Ra-226	3.15E-16	4.79E-16	1.17E-06	1.77E-06
Co-57	5.61E-15	6.63E-15	2.08E-05	2.45E-05
Ce-144	8.53E-16	2.93E-15	3.16E-06	1.08E-05
H-3	3.31E-19	0	1.22E-09	0.00E+00

Attachment 5 – XOQDOQ Run Results

HEAT EMISSION RATE (CAL/SEC) 0.0

0ALL GROUND LEVEL RELEASES.
1USNRC COMPUTER CODE - XOQDOQ, VERSION 2.0 RUN DATE: 20180912

0OMAHA PUBLIC POWER

Auxiliary Building
CORRECTED USING SITE-SPECIFIC FACTORS
SPECIFIC POINTS OF INTEREST

0RELEASE ID	TYPE OF LOCATION	DIRECTION FROM SITE	DISTANCE (MILES)	DISTANCE (METERS)	X/Q (SEC/CUB.METER) NO DECAY UNDEPLETED	X/Q (SEC/CUB.METER) 2.000 DAY DECAY UNDEPLETED	X/Q (SEC/CUB.METER) 8.000 DAY DECAY DEPLETED	D/Q (PER SQ.METER)
A	SITE BOUNDARY	N	0.25	400.	5.8E-05	5.8E-05	5.5E-05	1.6E-07
A	SITE BOUNDARY	NNE	0.25	400.	5.3E-05	5.3E-05	5.0E-05	1.2E-07
A	SITE BOUNDARY	NE	0.25	400.	4.8E-05	4.7E-05	4.5E-05	7.4E-08
A	SITE BOUNDARY	ENE	0.25	400.	6.4E-05	6.3E-05	6.0E-05	6.2E-08
A	SITE BOUNDARY	E	0.25	400.	7.1E-05	7.0E-05	6.7E-05	8.4E-08
A	SITE BOUNDARY	ESE	0.25	400.	6.8E-05	6.7E-05	6.4E-05	1.2E-07
A	SITE BOUNDARY	SE	0.25	400.	5.6E-05	5.6E-05	5.3E-05	2.3E-07
A	SITE BOUNDARY	SSE	0.25	400.	7.4E-05	7.4E-05	7.0E-05	3.9E-07
A	SITE BOUNDARY	S	0.25	400.	3.5E-05	3.5E-05	3.3E-05	1.8E-07
A	SITE BOUNDARY	SSW	0.25	400.	3.5E-05	3.5E-05	3.3E-05	7.4E-08
A	SITE BOUNDARY	SW	0.25	400.	2.6E-05	2.6E-05	2.5E-05	6.3E-08
A	SITE BOUNDARY	WSW	0.25	400.	3.6E-05	3.6E-05	3.4E-05	6.5E-08
A	SITE BOUNDARY	W	0.25	400.	3.8E-05	3.8E-05	3.6E-05	7.3E-08
A	SITE BOUNDARY	WNW	0.25	400.	6.2E-05	6.1E-05	5.8E-05	1.4E-07
A	SITE BOUNDARY	NW	0.25	400.	8.1E-05	8.1E-05	7.7E-05	2.8E-07
A	SITE BOUNDARY	NNW	0.25	400.	7.2E-05	7.2E-05	6.8E-05	2.1E-07

Attachment 6 – XOQDOQ Output File (imbedded txt file)

FORT CALHOUN STATION

DOCKET NUMBER 50-285 / LICENSE NUMBER DPR-40

ATTACHMENT 2

COMPARISON MATRIX FOR ISFSI EALS BASED ON THE PROPOSED REGULATORY GUIDE DG-1346 “EMERGENCY PLANNING FOR DECOMMISSIONING NUCLEAR REACTORS” TO THE PROPOSED FCS EMERGENCY CLASSIFICATION SYSTEM AND ISFSI EALS

DG-1346, Appendix A ICs/EALs	Proposed EAL Matrix for FCS	Comparison
EU2 **ECL:** Unusual Event **Initiating Condition:** Damage to a loaded cask CONFINEMENT BOUNDARY. **Operating Mode Applicability:** Not Applicable	**EU2** **ECL:** Unusual Event **Initiating Condition:** Damage to a loaded cask CONFINEMENT BOUNDARY.	•
Example Emergency Action Levels: (1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by a radiation monitor reading greater than NORMAL background at or near the cask.	**Emergency Action Levels:** 1 1. Damage to a loaded cask confinement BOUNDARY as indicated by an abnormal radiation reading of >2 mRem/hr. (gamma) within the ISFSI Protected Area or on a Horizontal Storage Module (HSM) concrete surface.	• Provided FCS-specific radiation levels that conform to 10 CFR 20.1301 allowable levels based on calculations.
Basis: This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage. The existence of "damage" is determined by radiological survey. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the IC may be determined based on measurement of a dose rate at some distance from the cask.	**Basis:** This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage. The existence of "damage" is determined by radiological survey. The radiation limits listed in the EAL reflect calculations based on 10 CFR 20.1301(a)(2) radiation dose limits to the public. In addition to aligning with 10 CFR 20.1301 limits, the radiation levels chosen are a reasonable indication that actual cask confinement boundary has occurred due to the level being greater than calculated levels. Security-related events for ISFSIs are covered under ICs EU1 and EA1.	• Added FCS specific basis information.

DG-1346, Appendix A ICs/EALs	Proposed EAL Matrix for FCS	Comparison
Security-related events for ISFSIs are covered under ICs EU1 and EA1.		
EU1 **ECL:** Unusual Event **Initiating Condition:** Confirmed SECURITY CONDITION, or threat, at the independent spent storage installation (ISFSI). **Applicability:** IOEP	**EU1** **ECL:** Unusual event **Initiating Condition:** Confirmed SECURITY CONDITION, or threat, at the independent spent storage installation (ISFSI).	
Example Emergency Action Levels: (1 or 2 or 3) (1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision) and impacting the ISFSI. (2) Notification of a credible security threat directed at the ISFSI. (3) A validated notification from the NRC providing information of an aircraft threat.	**Emergency Action Levels:** 1 or 2 1. A SECURITY CONDITION as reported by the security force and impacting the ISFSI. 2. Notification of a credible security threat directed at the ISFSI.	• Removed the term "HOSTILE ACTION" as it does not apply to an ISFSI Only Facility • Deleted EAL 3 related to aircraft threat
Basis: This initiating condition (IC) addresses events that pose a threat to plant personnel and, thus, represents a potential degradation in the level of plant safety. Security events which do not meet one of these emergency action levels (EALs) are adequately addressed by the requirements of Section 73.71, "Reporting of safeguards events," of Title 10 of the Code of Federal Regulations (10 CFR) Part 73, "Physical Protection of Plants and Materials," or Section 50.72, "Immediate notification requirements for operating nuclear power reactors," of	**Basis:** This IC addresses events that pose a threat to facility personnel or spent fuel, and thus represent a potential degradation in the level of facility safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR 73.71 or 10 CFR 50.72. Security events assessed as ADVERSARIAL ACTION are classifiable under IC EA1. Timely and accurate communication between the security force and the ISFSI Shift Supervisor/Emergency Director is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-	• Deleted reference to communicating with the Control Room and referenced communicating with the ISFSI Shift Supervisor/Emergency Director • Deleted wording associated with aircraft threats • Deleted wording regarding security-sensitive information

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10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and offsite response organizations (OROs). Security plans and terminology are based on the guidance provided by NEI 03-12 "Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]". EAL #1 references (site-specific security shift supervision) because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of safeguards and Section 2.390, "Public inspections, exemptions, and requests for withholding," of 10 CFR Part 2, "Agency Rules of Practice and Procedure," information. EAL #2 addresses the receipt of a credible security threat directed at the ISFSI. The credibility of the threat is assessed in accordance with (site-specific procedure). EAL #3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by North American Aerospace	related notifications to site personnel and Offsite Response Organizations (OROs). Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*. EAL #1 references the security force because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR 2.390 information. EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with Security procedures. Escalation of the emergency classification level would be via IC EA1.	

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Defense Command (NORAD) through the NRC. Validation of the threat is performed in accordance with (site-specific procedure). Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan. Escalation of the emergency classification level would be via IC EA1.		
EA1 **ECL:** Alert **Initiating Condition:** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. **Applicability:** IOEP	**EA1** **ECL:** Alert **Initiating Condition:** ADVERSARIAL ACTION is occurring or has occurred.	• Changed Initiating Condition wording • Deleted reference to airborne threat
Example Emergency Action Levels: (1) A HOSTILE ACTION is occurring or has occurred within the ISFSI as reported by the (site-specific security shift supervision). (2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	**Emergency Action Levels:** 1 1. An ADVERSARIAL ACTION is occurring or has occurred as reported by the security force.	• Reworded to make EAL specific to FCS ISFSI facility • Deleted Example EAL 2 related to aircraft threat
Basis: This IC addresses the occurrence of a HOSTILE ACTION within the ISFSI or	**Basis:** This IC addresses the occurrence of an ADVERSARIAL ACTION.	• Changed wording to reflect FCS ISFSI EAL wording • Deleted wording associated with aircraft threats

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notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack compromising stored spent fuel or damaging the storage casks, or the need to prepare the plant and staff for a potential aircraft impact. Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Security plans and terminology are based on the guidance provided by NEI 03-12. As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of possible onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of OROs, allowing them to be better prepared should it be necessary to consider further actions. This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72. EAL #1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the ISFSI. EAL #2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes.	Timely and accurate communication between the security force and the ISFSI Shift Supervisor/Emergency Director is essential for proper classification of a security-related event. As time and conditions allow, these events require a heightened state of readiness by the facility staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.	• Deleted reference to communicating with the Control Room and referenced communicating with the ISFSI Shift Supervisor/Emergency Director • Deleted wording regarding security-sensitive information

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The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with (site-specific procedure). The NRC HOO will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC. In some cases, it may not be readily apparent if an aircraft impact within the ISFSI was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration should not be unduly delayed while awaiting notification by a Federal agency. Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.		