US SFR OWNER, LLC - PLAN FOR A GENERAL AUDIT OF THE KEMMERER UNIT 1 CONSTRUCTION PERMIT APPLICATION (EPID NO. L-2024-CPS-0000)

Applicant:	US SFR Owner, LLC
Applicant Address:	15800 Northup Way Bellevue, WA 98008
Plant Name and Units:	Kemmerer Power Station Unit 1
Docket No.:	50-613

Background

By letter dated March 28, 2024, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24088A060) TerraPower, LLC (TerraPower), on behalf of its wholly-owned subsidiary US SFR Owner, LLC (USO), submitted a construction permit (CP) application and corresponding preliminary safety analysis report (PSAR) to the U.S. Nuclear Regulatory Commission (NRC) staff for a reactor facility pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," and section 103 of the Atomic Energy Act of 1954, as amended. The facility, referred to as Kemmerer Power Station Unit 1 (Kemmerer 1), would be built in Lincoln County, Wy. On May 21, 2024 (ML24135A109), the NRC staff accepted USO's CP application for docketing. A notice of the acceptance for docketing was published in the *Federal Register* on June 4, 2024 (89 FR 47997). The NRC staff is conducting a detailed review of the Kemmerer 1 CP application. This regulatory audit is intended to assist the NRC staff in its review of the Kemmerer 1 CP application.

Purpose

The purpose of this audit is for the NRC staff to gain a better understanding of the Kemmerer 1 CP application by providing an opportunity to review and discuss the material used to support statements or conclusions in the PSAR. The audit is expected to facilitate a more effective and efficient review by allowing the NRC staff to review and discuss the supporting material with the objective of improving communication and eliminating unnecessary requests for additional information. The NRC staff plans to audit material which may be needed to make a reasonable assurance finding that the preliminary design protects public health and safety. If the NRC staff identifies information that is needed to support a finding, USO will need to submit that information on the application docket(s).

Regulatory Audit Basis

The basis for the audit are the regulations in 10CFR pertinent to the NRC staff's review of the CP application, including 10 CFR 50.34, "Contents of applications; technical information," 10 CFR 50.35, "Issuance of construction permits," 10 CFR 100.10, "Factors to be considered when evaluating sites," and 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

Regulatory Audit Scope

This audit will focus on information provided by USO on the electronic reading room (eRR) and during meetings and interactions with USO. This audit will provide the NRC staff with information related to the review of the Kemmerer 1 CP application. Specific objectives and topics to be discussed will be identified in subsequent correspondence, such as an email, and dates will be proposed for discussion of the information.

Information and Other Material Necessary for the Regulatory Audit

USO should be prepared to provide documents, reports, calculations, computer code verification, and other material, as applicable, supporting the analyses documented in the PSAR. The NRC staff may request that USO make these additional materials available on the eRR or during a site visit if one is conducted. Initial information requests and discussion items are identified in the enclosed table. Additional requests will be provided during the remainder of the audit.

Team Assignments

Mallecia Sutton	Senior (Sr.) Project Manager, Audit Lead
Reed Anzalone	Sr. Reactor Systems Engineer, Lead Reviewer
Stephanie Devlin-Gill	Sr. Project Manager, Audit Support
Roel Brusselmans	Project Manager, Audit Support
Deion Atkinson	Project Manager, Audit Support

Additional audit team members will be added, as needed.

Logistics

Entrance Meeting: July 23, 2024, at 2:00 PM ET

Exit Meeting: January 2025, precise date, and time are to be determined.

The audit will follow the guidance in the Office of Nuclear Reactor Regulation's (NRR) Office Instruction LIC-111, Revision 1, "Regulatory Audits," (ML19226A274). Audit meetings will take place in a virtual format, using Microsoft Teams or other similar platform. Audit meetings will be scheduled on an as needed basis and, if applicable, once the NRC staff has had the opportunity to review documents placed on the eRR. If a site visit is needed, review topics will be provided to USO, and mutually agreeable dates will be established. The audit will begin July 2024, and continue for approximately 6 months, with activities occurring intermittently during that period. Interim status report(s) on topics being addressed by the audit will be provided to USO.

The audit period may be reduced or extended, depending on the progress made by the NRC staff and USO in addressing audit questions. Additional audit activities may be planned as necessary to support the NRC staff's understanding of information needed to complete the review of the Kemmerer 1 CP application.

To improve the efficiency of the audit, USO and the NRC staff discussed the use of the eRR, established by TerraPower, that would allow the NRC staff read-only access to the technical information provided by USO. Use of the eRR is acceptable provided that USO establishes measures to limit access to specific NRC staff (e.g., based on NRC email addresses or the use of passwords which will only be assigned to the NRC staff directly involved in the audit on a need-to-know basis), and to make the documents view-only (i.e., prevent NRC staff from saving, copying, downloading, or printing any documents). The conditions associated with the eRR must be maintained throughout the audit process. The NRC audit project managers will provide USO with the names of the NRC staff audit team that will require access to the eRR.

Special Requests

None.

Deliverables

At the completion of the audit, the audit team will issue an audit summary within 90 days after the exit meeting but will strive for a shorter duration. The audit summary will be declared and entered as an official agency record in ADAMS and be made available for public viewing.

Please contact Mallecia Sutton at 301-415-0673 or by via email at Mallecia.Sutton@nrc.gov with any questions related to the conduct of the audit.

Date: July 15, 2024

/RA/

Joshua Borromeo, Chief
Advanced Reactor Licensing Branch 1
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No.: 99902100

Docket No.: 50-613

Enclosure:
Audit Questions

cc: TerraPower Natrium via GovDelivery

SUBJECT: US SFR OWNER, LLC - PLAN FOR A GENERAL AUDIT OF THE KEMMERER 1 CONSTRUCTION PERMIT APPLICATION (EPID NO. L-2024-CPS-0000) DATED: JULY 15, 2024

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OFFICE	NRR/DANU/UAL1:PM	NRR/DANU/UAL1:PM	NRR/DANU/UAL1:LA	NRR/DANU/UTB2:BC
NAME	DAtkinson	MSutton	DGreene	CdeMessieres
DATE	7/2/2024	7/2/2024	7/11/2024	7/3/2024
OFFICE	NRR/DANU/UAL1:BC			
NAME	JBorromeo			
DATE	7/15/2024			

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Audit Questions for the Kemmerer 1 Preliminary Safety Analysis Report

Chapter 1, "General Plant and Site Description"		
Question Number	**Description**	**Section**
1-1	Kemmerer 1 preliminary safety analysis report (PSAR) table 1.4-2, "Topical Reports," does not list NATD-LIC-RPRT-0001, "Regulatory Management of Natrium Nuclear Island Design Interfaces." The NRC staff requests that US SFR Owner, LLC (USO) clarify why this topical report appears to be incorporated by reference into Kemmerer 1 PSAR 1.1.4.4.9, "Energy Island Facilities," but isn't included in table 1.4-2.	1.4, "Conformance with Regulatory Criteria and Referenced Material"
1-2	PSAR table 1.4-2 states that topical report TP-LIC-RPT-0005, "Radiological Release Consequences Methodology," is referenced in PSAR section 3.1.1, "Overview of Probabilistic Risk Assessment (PRA)." However, the NRC staff identified that the report appears in PSAR section 3.3.1, " Licensing Methodology for Design Basis Accidents," and it was not found in PSAR section 3.1.1. Please clarify if this was a typographical error in table 1.4-2.	1.4
1-3	PSAR table 1.4-2 states that TP-LIC-RPT-0003, "Radiological Source Term Methodology," is referenced in PSAR section 3.1, "Probabilistic Risk Assessment." However, the NRC staff identified that the report appears in in PSAR section 3.2, "Licensing Methodology for Mechanistic Source Term," and it was not identified in PSAR section 3.1.1. Please clarify this discrepancy.	1.4
1-4	There appears to be an inconsistency in the title for TP-LIC-RPT-0007, "Design Bases Accident (DBA) Transient Methods for Events with Radiological Release," in Kemmerer PSAR table 1.4-2 and the title of this report in Kemmerer PSAR 3.3.1.2. The title in table 1.4-2 is "Design Basis Accident (DBA) Transient Methods for Events with Radiological Release," while the title for this report in PSAR section 3.3.1.2, "Ex-Vessel DBAs and DBAs with Potential Radiological Release," is "Design Basis Accident Methodology Report for Potential Fuel Failure and Release." The NRC staff requests that USO clarify this discrepancy between the titles of TP-LIC-RPT-0007.	1.4
1-5	PSAR table 1.4-3, "Technical Reports," references TP-LIC-RPT-0009, "Major Accident Methodology Report," which appears to be inconsistent with PSAR section 3.3, "Licensing Methods for Evaluation of Licensing Basis Events," where a report with the same title is listed as TP-LIC-RPT-0010. The NRC staff requests that USO clarify the discrepancy regarding the report numbers.	1.4
1-6	PSAR table 1.4-5, "Consensus Codes and Standards," lists DANU-ISG-2023-01, draft, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications – Roadmap." This does not appear to be completely accurate. The identifier for this document is DANU-ISG-2022-01 and it was issued as final in March of 2023. The NRC staff requests that USO clarify the discrepancy regarding the edition of this document.	1.4
1-7	PSAR table 1.4-1, "Conformance with Regulatory Guides" does not list Regulatory Guide (RG) 1.11, Revision 1, "Instrument Lines Penetrating the Primary Reactor Containment" (ML100250396). This report appears to be incorporated by reference into PSAR 7.2.3.2.3,	1.4

	"Regulatory Guidance," and, as such, the staff requests clarification if it should be included in table 1.4-1.	
1-8	PSAR table 1.4-1 does not list RG 1.221, Revision 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants." This report appears to be incorporated by reference into PSAR 36.4, and, as such, the staff requests clarification if it should be included in table 1.4-1.	1.4
Chapter 2, "Site Information"		
2-1	The NRC staff requests that USO provide clarification and potential updates to the PSAR regarding the following items: a) USO evaluates the site-specific probable maximum flood (PMF) regarding streams using a combination of Hydrologic Engineering Center (HEC), Hydrologic Modeling System (HMS), and River Analysis System (RAS) models. Explain how the HEC-HMS subbasin outflows are used as input to HEC-RAS channels. Especially, clarify how the HEC-RAS upstream boundary conditions are set up for the HEC-RAS channel sections which are located within each HEC-HMS subbasin. b) The HEC-HMS was set up with four subbasins - West, Laydown, East and Confluence. The East and West subbasins cover approximately 67 percent and 27 percent of the entire basin, respectively, and are highly channelized so that they could be routed better by more refined subbasins. Discuss any potential bias (underestimation) in PMF estimates due to not fully accounting for detailed channel networks on these subbasins. c) USO states that a unit hydrograph method in HEC-HMS is used to transform rainfall to runoff. However, the models have not been calibrated and verified with observed data. Discuss the applicability of increasing the peak and decreasing time to peak as recommended by NUREG/CR-7046, "Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America" (ML11321A195).	2.5.1.2, "Probable Maximum Flood on Streams and River"
2-2	The NRC staff requests that USO provide clarification and potential updates to the PSAR regarding the potential for sediment build-up (soil and debris) on the channels near the plant site due to slowing and merging effects of channel flows, especially for the combined events of big flood and dam failure as addressed in PSAR section 2.5.1.3, "Potential Dam Failures."	2.5.1.2.9, "Sedimentation and Erosion Impact"
2-3	PSAR section 2.5.3.1.9 mentions that the maximum groundwater levels were obtained from a three-dimensional groundwater model. However, the PSAR does not include a description of this groundwater modeling. Please provide the calculation document (if available), or provide a brief description of the groundwater model, including the assumptions, data, setup, and calibration/simulation results involved in the groundwater modeling.	2.5.3.1.9, "Subsurface Hydrostatic Loading"
2-4	Section 2.5.3.1.11 states that a three-dimensional groundwater model was developed for the Kemmerer Unit 1 site to estimate construction dewatering rates. However, the PSAR does not include a description of this groundwater modeling. The NRC staff requests that USO provide the calculation document (if available) or provide a brief description of the groundwater model. The document should include the	2.5.3.1.11, "Reliability of Dewatering Systems"

	assumptions, data, setup, and calibration/simulation results involved in the groundwater modeling.	
2-5	The NRC staff would like USO to clarify if identified geological and seismology conditions (referenced in PSAR section 2.6) have been considered for potential effects on developing security plans under the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 73, "Physical Protection of Plants and Materials."	2.6, "Geology, Seismology, and Geotechnical Engineering"
2-6	The NRC staff requests that USO clarify if the identified regional climatology and local site meteorology (referenced in PSAR section 2.4) have been analyzed and considered for potential effects on developing security plans under the requirements in 10 CFR Part 73.	2.4, "Regional Climatology, Local Meteorology, and Atmospheric Dispersion"
2-7	The NRC staff would like USO to clarify if identified hazardous material in the vicinity, onsite, and nearby industrial, military, and transportation facilities (referenced in PSAR section 2.3) have been considered for potential effects on developing security plans under the requirements in 10 CFR Part 73.	2.3, "Regional Climatology, Local Meteorology, and Atmospheric Dispersion"
2-8	The NRC staff requests that USO provide the following input and output files in native file format (e.g., text file) for the purpose of the NRC staff performing confirmatory analysis of the atmospheric dispersion estimates in PSAR sections 2.4.4.1 and 2.4.4.2: • ARCON input and output files used to determine atmospheric dispersion factor (X/Q) at exclusion area boundary (EAB) and low population zone (LPZ) as discussed in PSAR section 2.4.4.1.1, "ARCON," and presented in PSAR table 2.4-79, "ARCON 99.5th Percentile X/Q Values at the EAB." • ARCON input and output files used to determine Control Room X/Q's as discussed in section 2.4.4.1.1 and presented in PSAR table 2.4-78, "ARCON 95th Percentile X/Q Values for the On-Site Control Room Evaluation for the Average Male." • XOQDOQ input and output files used to determine X/Q's as discussed in section 2.4.4.1.1 and presented in tables 2.4-80, "XOQDOQ Distances from [Reactor Building (RXB)] to Nearest Receptors Per Directional Sector" through 2.4-89 "[Energy Island (EI)] 50-mile X/Q and D/Q Values, Deposited."	2.4.4.1, "Short-Term Diffusion Estimates," and 2.4.4.2, "Long-Term Diffusion Estimates"
2-9	The NRC staff requests that USO provide the following for the purpose of the NRC staff performing confirmatory analysis of the atmospheric dispersion estimates provided in section 2.4.4. The NRC staff requests USO provide the three years of onsite meteorological data collected from the Naughton Power Plant meteorological tower. This data should be provided in RG 1.23, appendix A format and in a native file format (e.g., text file).	2.4.4, "Atmospheric Dispersion"
2-10	The NRC staff requests that USO provide the following: The Senior Seismic Hazard Analysis Committee (SSHAC) report covering the Seismic Source Model, Ground Motion Model, and Probabilistic Seismic Hazard Analysis (PSHA) and site response calculation sections, including any appendices and electronic attachments.	2.6.2, "Vibratory Ground Motion"
2-11	The NRC staff requests that USO provide reports or calculation packages that describe the following: • ALOHA analysis of hazards from a potential pipeline rupture, which should include the assumptions made and their bases.	2.3

	• ALOHA analysis of hazards from transportation of hazardous chemicals on nearby roads and at nearby facilities, which should include the assumptions made and their bases. • Analysis of explosion hazards from transportation of explosives to the Kemmerer mine for blasting coal.	
2-12	The NRC staff requests that USO provide reports or calculation packages that describe the following: • Subsurface Investigation Data Report (preferably in downloadable form for staff independent analyses), which should include the boring logs and core run photographs. • Foundation bearing capacity and associated settlement assessment report, which should include estimation of rock mass strength and stiffness calculations. • Estimation of lateral pressure. • Analyses of stability of both soil and rock slopes. • Description of the model to represent the response of the contact between soil and rock with the outer surface of the safety-related structures, e.g., Reactor Building (RXB), Fuel Handling Building (FHB), Reactor Auxiliary Building (RAB), and estimation of the model parameters that would be used in subsequent analyses (e.g., soil-structure interaction analysis).	2.6.4, “Stability of Subsurface Materials and Foundations” and 2.6.5, “Stability of Slopes”
	Chapter 3, “Licensing Basis Events”	
3-1	The NRC staff requests access to documentation of the development of the source terms provided in the summary table (PSAR tables 3.2-1, “Protected Loss of Flow or Loss of Offsite Power” through 3.2-19, “Unprotected Loss of Heat Sink, Full Core and [In-Vessel Storage (IVS)] Fail).” This includes calculations, description of the implementation of the source term methodology, as well as justifications for inputs and assumptions, computer code output, or other relevant documentation. Please provide similar information for the implementation of the major accident technical report.	3.2, “Licensing Methodology for Mechanistic Source Term”
3-2	The NRC staff requests that USO clarify the assumptions and basis for source term parameters to model the release to the environment which are not described in the PSAR, such as radionuclide chemical forms, aerosol size distribution, plume sensible heat.	3.2
3-3	The NRC staff requests that USO describe the use of the source term summary tables. Specifically, confirm that the cumulative activity release in each column is given for all time up to the listed time at the heading of each column (not for just the interval between the headings). The NRC staff requests that USO confirm that table 3.2-19 is the source term for the major accident.	3.2
3-4	The NRC staff requests that USO provide access to documentation of the consequence analyses for licensing basis events (LBEs) (anticipated operational occurrences (AOOs), design-basis events (DBEs), beyond design basis events (BDBEs), and design-basis accidents (DBAs)), including uncertainty quantification for comparison to the frequency-consequence (F-C) curve as plotted in PSAR Figure 3.5-1, “F-C Chart for LBEs with Uncertainty Bands,” as well as the integrated risk evaluations. The NRC staff requests that the documentation include calculations, description of the implementation of the radiological release consequences methodology, as well as justifications for inputs	3.3.1, “Licensing Methodology for Design Basis Accidents,” 3.3.2, “Licensing Methodology for AOO, DBE, and BDBE,” 3.5, “LBE Summary,” through 3.9, “Design Basis

	and assumptions, computer code output, or other relevant documentation.	Accidents," and 4.1, "Overall Plant Risk Performance Summary"
3-5	The NRC staff requests that USO provide access to documentation of the consequence analysis calculation and supporting documents for the major accident.	3.3.3, "Licensing Methodology for Major Accident"
3-6	The NRC staff requests that USO clarify which source terms are related to which LBEs or other quantified events (OQEs), given the naming conventions in various PSAR sections, as well as the emergency planning zone (EPZ) and major accident technical reports and the related topical reports currently under review.	3
3-7	PSAR table 3.1-1, "Potentially More-Than-Minor PRA Assumptions," states that the local core faults have an initiating event frequency of 1x10-3 per year. a) The assumption listed for the row in the table is "Local core blockage," but the description refers to "local core faults" and mentions loss of hydraulic hold down being included in the value. The NRC staff requests that USO clarify what initiators are considered in the event frequency. b) The table states that the generic data for core faults show a range of seven orders of magnitude. The NRC staff requests that USO provide further discussion on the generic data used to derive the frequency. The NRC staff also requests that USO discuss how the frequency was determined and how the uncertainty is represented in the analysis.	3.1, "Probabilistic Risk Assessment"
3-8	PSAR table 3.1-2, "Hazard Screening Summary," provides a list of hazards screened out of analysis. The NRC staff requests that USO provide access to documentation supporting the results discussed in the table.	3.1
3-9	PSAR section 3.3.4.3 discusses the termination criteria for transient analyses. a) One termination criterion is peak cladding temperature below 625 degrees Celsius (°C) and "stable." The NRC staff requests that USO provide the criteria used to determine whether temperature is stable prior to terminating an analysis. b) The section states that transient analyses may be terminated when k-effective is equal to one for BDBEs and for transients that do not result in a reactor scram. The NRC staff requests that USO provide a justification for terminating transients when the reactor is still critical. c) The NRC staff requests that USO discuss whether transients are evaluated for a return to criticality after an initial shut down period. If not, the NRC staff requests that USO provide a justification. If yes, the NRC staff requests that USO discuss the criteria used to determine whether a transient should be subject to a re-criticality evaluation.	3.3.4.3, "LBE Analysis End States"
3-10	The legend on PSAR Figure 3.5-1, "F-C Chart for LBEs with Uncertainty Bands" is difficult to interpret because the colors are very similar. The NRC staff requests that USO provide an alternate version (or multiple alternate versions, as appropriate) of the figure with a clear legend.	3.5

	Alternatively, provide the underlying information so the NRC staff can recreate the figure.	
3-11	PSAR section 3.11 states that the "[f]inal results, including radionuclide retention requirements…will be provided at the operating license stage." The NRC staff presumes that the "requirements" referred to are the design requirements for the fuel system related to radionuclide retention and release from the fuel matrix. The NRC staff requests that USO describe how the radionuclide retention and release from the fuel were modeled for the preliminary analyses supporting the CPA.	3.11, "Fuel System Design"
3-12	The NRC staff requests that USO provide detailed documentation of the LBE analyses summarized in PSAR sections 3.3.4 and 3.5 and discussed in detail on PSAR sections 3.6, "Anticipated Operational Occurrences," through 3.9. The NRC staff requests the documentation include calculations, description of the implementation of methodologies described in section 3.3, "Licensing Methods for Evaluation of Licensing Basis Events," as well as justifications for inputs and assumptions, computer code output, or other relevant documentation. Refer also to audit questions 3-4 and 3-7 above.	3.3.4, "LBE Plant Response and Analysis Overview," and 3.5, through 3.9, "Design Basis Accidents"
3-13	PSAR section 3.3.2.1 discusses licensing methodologies used to evaluate non-DBA LBEs for Kemmerer Unit 1. The section states that the AOO, DBE, and BDBE analyses may use either the same conservative approach used for DBAs or a best-estimate plus uncertainty (BEPU) based approach. a) The NRC staff requests that USO clarify which LBEs in the PSAR are conservative and which are BEPU. b) The BEPU approach does not appear to be described in the PSAR or in documentation provided in the pertinent topical reports submitted to date. The NRC staff requests that USO provide an overview of how uncertainties are incorporated into any BEPU event analyses discussed in chapter 3, including uncertainties in fuel performance, core design and neutronics, thermal-hydraulics, system-level responses, phenomena considered in the source term analyses, and phenomena considered in the consequence analyses.	3.3.2.1, "AOOs, DBEs, and BDBEs"
3-14	Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1 (ML19241A472), section 3.2.2,"LBE Selection Process, "Task 7a," Evaluate LBEs Against F-C Target" states that"…when the uncertainty bands […] defined by the 5th percentile and 95th percentile of the frequency estimates straddles a frequency boundary, the LBE is evaluated in both LBE categories." Many LBEs in the PSAR span multiple categories and it is unclear whether the events are analyzed in all appropriate categories. In particular, the NRC staff is concerned with the following events which have a 95th percentile frequency indicating that they should be considered in categories corresponding to more frequent events and requests that USO justify why the events in the PSAR that were not analyzed in those accident categories: • RRS-SPLX-BL, RRS-RWG-1, and RFH-LMCA-BL, which are categorized as DBEs but have upper 95th percentiles in the AOO frequency range (> 1×10-02), and	3.6, 3.7, "Design Basis Events," and 3.8, "Beyond Design Basis Events"

	• RFH-FDSP-2, RFH-ESWR-2, RFH-FDBL-1, RFH-FDEM-1, RFH-FDPI-BL, and RFH-FDRC-1, which are categorized as BDBEs but have upper 95th percentiles in the DBE frequency range (> 1×10-04).	
3-15	PSAR section 3.3.4.1.1 states that for increase in heat removal events, "[i]f a reactor power runback successfully completes and prevents a scram setpoint from being reached, the event is not classified as an LBE." The NRC staff requests that USO clarify why it is not appropriate to consider these events as LBEs.	3.3.4.1.1, "Increase in Heat Removal"
3-16	PSAR section 3.3.4.1.5 describes the response to "[a] localized fault, such as a localized blockage of fuel subchannels…during full power operating conditions." a) Fuel misloads are typically considered local faults and are typically also considered at conditions leading up to full power conditions. The NRC staff requests that USO clarify if local faults include fuel misloads. If so, the NRC staff requests that USO clarify how are they evaluated. b) The section states that "[t]he assumption an entire fuel assembly failed due to the event is conservative…." The NRC staff requests that USO provide justification that a local fault would not propagate to other assemblies in the core.	3.3.4.1.5, "Local Fuel Fault"
3-17	PSAR section 3.13.2, “Design Basis,” includes a reference to principal design criteria (PDC) 15, “Primary Coolant System Design.” However, PDC 15 is not identified in PSAR section 5.3.2.6, “PDC 15 - Primary Coolant System Design,” as being applicable to PSAR section 3.13.2. The NRC staff requests that USO clarify this discrepancy regarding PDC 15.	3.13, 3. “Thermal and Hydraulic Design,”
3-18	There do not appear to be any LBEs that represent reactivity transients initiated from a low power or shutdown conditions. Please discuss any considerations for these types of LBEs.	3.5, 3.6, 3.7, 3.8, 3.9
3-19	There are transient overpower events caused by control rod withdrawal in both the AOO and BDBE frequency ranges. However, there are no reactivity events in the DBE range, which necessitated the addition of the seismic DBA to cover reactivity insertion events. Please discuss why there are no reactivity DBEs.	3.6, 3.7, 3.8, 3.9
3-20	The NEI 18-04 methodology requires that for each DBE identified, a deterministic DBA be defined. AOOs and BDBEs with uncertainty bands that span the DBE frequencies should also be evaluated as DBEs. With that in mind, table 3.5-4, “Summary of DBAs,” and section 3.9 appear to be missing event RFH-OEFH-BL from table 3.5-2, “Summary of DBEs,” and all AOOs and BDBEs with uncertainty bands that straddle the DBE frequency boundary (at least one AOO and six BDBEs, looking at just the ones that result in release for the nominal evaluation). The NRC staff requests that USO justify why these events were not analyzed as DBAs, as the methodology requires.	Table 3.5-4, 3.9
3-21	PSAR section 3.1.1.2 indicates that PRA assumptions with “more-than-minor impacts” are described in table 3.1-1. The NRC staff requests that USO provide more information on the acceptance criteria used for “more-than-minor impacts” and provide examples of PRA assumptions that were screened out using these criteria.	3.1.1.2, “PRA Technical Adequacy”

3-22	Section 3.1.1.2 references an evaluation comparing the differences from the CP stage design to the CP stage PRA. The NRC staff requests that USO provide this evaluation or provide a list of these differences.	3.1.1.2
3-23	PSAR section 3.1.1.6 provides a list of plant operating states, but the names provided in the listing are not sufficient to determine what state the plant systems are in. The NRC staff requests that USO provide a more detailed description of these plant states and how they relate to or differ from each other.	3.1.1.6, "Plant Operating States"
3-24	PSAR section 3.1.1.7 states that "IEs were initially identified by reviewing published IE lists…" and "individual event frequencies were developed using appropriate and available generic data sources…" The NRC staff requests that USO provide references for these published IE lists and generic data sources and, where possible, provide the references in the eRR for staff audit.	3.1.1.7, "Initiating Event Identification"
3-25	PSAR table 3.1-1, consequential fuel damage states that "inadvertent motion of any partially raised core assembly during the refueling process does not result in damage that could result in radionuclide release of an adjacent assembly" and notes that this is reasonable given what is known about the current design. The NRC staff requests that USO provide a description of the design features being referenced here that support this assumption.	Table 3.1-1, "Potentially More-Than-Minor PRA Assumptions"
3-26	Regarding PSAR table 3.1-2, a) The NRC staff requests that USO provide justification for qualitatively screening out aircraft impact (versus quantitatively screening). b) "Corrosion (e.g., from salt water)" is listed as being screened out. The NRC staff requests that USO confirm that corrosion was considered as an initiator for leaks within the systems or provide justification for why not. c) Dropped loads are listed as screened out. The NRC staff requests that USO confirm whether this is only for the NOG-1 qualified crane or for all cranes within the facility. If for all cranes, provide further justification. d) High and low air temperatures are screened out, but the design temperatures listed in table 2.1-2, "Kemmerer Unit 1 Site Characteristics," indicate that they are based on the 99.6th percentile. The NRC staff requests that USO provide justification for screening out low and high temperature as initiators to events when temperatures outside of the design temperatures falls within an AOO frequency. e) With respect to screening out internal missiles, The NRC staff requests that USO confirm that there are not compressed gas cylinders within the facility that could be a source of internal missiles.	Table 3.1-2, "Hazard Screening Summary"
3-27	PSAR section 3.3.4.1.3 states that the Intermediate Sodium Pumps (ISPs) will reduce speed if a scram occurs. The NRC staff requests that USO explain if the ISPs also reduce speed if a power runback occurs and if not, could this result in a reactivity insertion due to overcooling.	3.3.4.1.3, "Reactivity"
3-28	The NRC staff requests that USO provide a more detailed description of the sequence of events that occur during a (a) power runback and (b) scram.	3.3.4.1.3

3-29	PSAR section 3.5.1 states that "LBEs are classified as risk-significant if the LBE site boundary dose exceeds 2.5 mrem over 30 days and the frequency of the dose is within one percent of the F-C target." The NRC staff requests that USO confirm that 95th percentile frequency and dose consequences were used in this determination.	3.5.1, "Summary Evaluation of AOOs, DBEs, and BDBEs"
3-30	PSAR section 3.5.1 states that "BDBEs are classified as high consequence is the dose exceeds 10 CFR 50.34 criteria." The NRC staff requests that USO confirm whether dose consequences were calculated over a 2-hour or 30-day duration.	3.5.1
3-31	Event RFH-OEFH-BL, "Removal of Fuel Assembly Prior to Decay with Active Cooling," is listed in table 3.5-2, but does not appear to be discussed anywhere else within the application, including PSAR section 3.7 where it should be described in more detail and section 3.9 where it should be evaluated as a DBA. The NRC staff requests that, if this event was removed as a DBE, provide justification for its removal.	Table 3.5-2
3-32	Events SUD-IACA-BL, SUD-IACA-1, and SUD-IACA-2 are events with loss of one train of IAC while shutdown. There do not appear to be any events for the same failure while at power. The NRC staff requests that USO provide justification for why this event would not occur at power or why it was not included within the PRA.	Table 3.5-1, "Summary of AOOs," Table 3.5-2, and Table 3.5-3, "Summary of BDBEs"
3-33	For RFH-LSPC-BL, the Spent Fuel Pool cooling system is restored prior to boiling. The NRC staff requests that USO explain the timing needed for this action and if this action is captured as a credited control.	3.6.4.2, "Loss of Spent Fuel Pool Cooling with Cooling Restored (RFH-LSPC-BL)"
3-34	For some events, the source term was released into alternative locations where the leak rate was higher than the expected release location. It is also important however that the alternative location has a conservatively low volume for dispersal (to give higher concentration). With that in mind, the NRC staff requests that USO confirm that the Ex-Vessel Handling Machine (EVHM) has a smaller volume than the Pin Removal Cell (PRC), the Head Access Area (HAA) has a smaller volume than the vapor trap cell compartment, the EVHM has a smaller volume than the Bottom Loading Transfer Cask (BLTC), and the EVHM has a smaller volume than the Pool Immersion Cell (PIC) or provide further justification on why the use of these alternative release locations is conservative.	"Mechanistic Source Term" sections 3.6.4.1.2, 3.7.3.2.2, 3.8.5.8.2, 3.8.5.9.2, 3.8.5.14.2, 3.8.5.15.2, 3.8.5.16.2, and 3.9.5.3.2
3-35	As described in RG 1.247, trial revision, "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities," (ML21235A008) PRA documentation should be sufficient to allow the NRC staff to determine the acceptability of the PRA and the PRA results used to support the application. Thus, the NRC staff requests that USO to identify and make available for the NRC staff audit the archival PRA-related documents (including PRA notebooks, other supporting documents, assumptions made in lieu of design, as-built, and as-operated details that should be tracked as the PRA is modified to reflect the lifecycle of the plant), to gain a full understanding of the technical bases of the PRA and how the assessment and its results are used to support the application.	Chapter 3, "Licensing Basis Events"
3-36	The NRC staff requests that USO to make the written PRA self-assessment report, which documents both the details and the summary findings of the self-assessment, available for the NRC staff	Chapter 3

	audit. Specifically, the NRC staff is looking to understand and audit the following: • Definition of the scope of the self-assessment, • Summary of the results of the self-assessment for each PRA technical element, • Summary of any "Findings" or "Facts and Observations (F&Os)" generated from the self-assessment, • Summary of identification of important assumptions and sources of uncertainty, their impacts, and the treatment, • Identification of the assessed capability category (CC) for each supporting requirement (SR) within the scope of the self-assessment and the basis for the assignment, and • The conclusions of the self-assessment team.	
3-37	The NRC staff requests that USO provide an overview and a demonstration of the software and analytical tools (e.g., Electric Power Research Institute Phoenix Architect 2.0) used to construct the PRA, perform the event sequence modeling, and quantify the risk. The demonstration should also include the UNCERT module used for the parametric uncertainty analysis to calculate the 5th and 95th percentile values. In addition, the demonstration should include the software and analytical tools that are used to determine the mechanistic source terms and evaluate the radiological consequences.	Chapter 3
3-38	The NRC staff requests that USO make available for audit the configuration and design control processes, along with their documentation, to ensure that the PRA is maintained, updated, and upgraded to reflect the as-designed, as-to-be-constructed, and as-to-be-operated plant design and margin to Quantitative Health Objectives (QHOs). Specifically, the NRC staff plans to audit: • The process for monitoring changes to the plant design, operation PRA technology, and industry experience, • The process that maintains and upgrades the PRA to be consistent with the as-designed, as-to-be-constructed, and as intended-to-operate, • The cumulative impact of important pending changes, • The process that maintains configuration control of computer codes and associated files used to support the PRA, and • Documentation of the configuration control program and its implementation to provide traceability of the work.	Chapter 3
3-39	NEI 18-04 requires documenting the basis for support structure, system, and component (SSC) categorization by the Integrated Decision-Making Process (IDP) and evaluating the adequacy of special treatment to SSC categorization, etc. Thus, the NRC staff requests that USO make available for audit the documentation relevant to the implementation of Licensing Modernization Project that USO has established.	Chapter 3
3-40	NEI 18-04 states that the decisions of the IDP should be documented and retained as a quality record; this function is critical to future decision-making regarding plant changes which have the potential to affect defense-in-depth (DID). The NRC staff requests that USO make	Chapter 3

	available all documents resulting from the IDP discussions and decisions (i.e., meeting minutes, summaries).	
Chapter 4 "Integrated Evaluations"		
4-1	The NRC staff requests that USO provide additional details on the other quantified events (OQEs) discussed in PSAR section 4.1, "Overall Plant Risk Performance". The NRC staff requests that USO describe the events that were considered for inclusion as OQEs but ultimately screened out, the events that were evaluated as OQEs, and how the OQEs included in the evaluation of Quantitative Health Objectives were evaluated.	4.1, "Overall Plant Risk Performance Summary"
Chapter 5, "Safety Functions, Design Criteria, and SSC Safety Classification"		
5-1	"PDC 19, Control Room," describes that the habitability dose analyses are described as being evaluated for DBAs using DBE source terms which credit non-safety-related with special treatment (NSRST) SSCs. Main Control Room (MCR) dose results to show PDC 19 criteria is met are not provided in PSAR section 5.3.2.10, "PDC 19 - Control Room," or other noted relevant PSAR sections (7.5.1, "Nuclear Island (NI)Heating, Ventilation, and Air Conditioning (HVAC) System," 7.6.7, "Control Room and Indications," and 7.8.4, "Nuclear Island Control Building"). The NRC staff requests the USO provide the NRC staff access to the analysis and supporting documentation describing the estimation of MCR dose consequences, including modeling of the MCR and the resulting DBA doses.	5.3.2.10, "Principal Design Criteria Regulatory Evaluation"
5-2	Classification for the instrumentation and controls (I&C) systems is not located in chapter 5 (e.g., reactor protection system (RPS), nuclear instrumentation system (XIS), reactor instrumentation system (RIS)); their classification is not identified until table 7.6.1.1a, "I&C System Classification and [defense line (DL)] Functions." Consider the addition of pointers between the tables or additional discussion added to the PSAR.	5.2, "Safety-Significant PRA Safety Functions"
5-3	PSAR section 3.13, "Thermal and Hydraulic Design," includes a reference to PDC 15, "Primary Coolant System Design." However, PDC 15 is not identified in PSAR section 5.3.2.6 as being applicable to PSAR section 3.13. Please clarify if "section 3.13" should be included in list of relevant sections in sections 5.3.2.6.	3.13, "Thermal and Hydraulic Design", 5.3
5-4	PSAR section 3.13 includes a reference to PDC 34, "Residual Heat Removal." However, PDC 34 is not identified in PSAR section 5.3.4.5 as being applicable to PSAR section 3.13. Please clarify if "section 3.13" should be included in the list of relevant section in section 5.3.4.5.	3.13, 5.3
5-5	PSAR section 3.12, "Nuclear Design," includes a reference to PDC 26, "Reactivity Control Systems." However, PDC 26 is not identified in PSAR section 5.3.3.7 as being applicable to PSAR section 3.12. Please clarify if "section 3.12" should be included in the list of relevant section in section 5.3.4.7.	3.12, "Nuclear Design", 5.3
5-6	PSAR section 5.3.4.3 states: "The RIM Program and assessment of available testing and service data provide an appropriate material surveillance program for the reactor vessel (RV) without the use of material surveillance coupons as described in section 7.1.2." PSAR section 7.1.2.2 states: "Available service degradation data for RV materials under anticipated environmental conditions from published testing results from operating and test reactors, and specific testing performed to qualify the RV materials, is evaluated to support the basis	5.3.4.3, "Fluid and Heat Transport Systems," 7.1.2, "Reactor Enclosure System"

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<td>for RV material selection.” PSAR section 7.1.2.3 states: “Based on the expected results of the evaluations of the effects on the metallic materials of neutron fluence and the high-temperature sodium operating conditions, there is margin to material degradation limits throughout the design lifetime such that an in-vessel material surveillance program is not required, as further described in section 7.1.2.2 for compliance with PDC 32.”

What is the maximum neutron fluence expected for the Natrium RV base and weld metals?
Please provide for audit the expected results of the evaluations of the effects on the metallic materials of neutron fluence and the high-temperature sodium operating conditions showing there is margin to material degradation limits throughout the design lifetime.
Please provide additional justification for the statement in the PSAR that an in-vessel material surveillance program is not required given the limited information available at the preliminary design stage.
What testing is planned to be performed to qualify the RV materials?
PSAR section 7.1.1.1 indicates the Natrium design will include “materials surveillance assemblies.” What is the purpose of these assemblies if “an in-vessel material surveillance program is not required”?
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<td>5-7</td>
<td>PSAR section 5.3.6.8 indicates that PDC 77 requires that “Components that are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leak tight integrity commensurate with the system’s safety significance, and (2) an appropriate material surveillance program for the intermediate coolant boundary.” PSAR section 7.1.4.1 states: “…the intermediate coolant boundary is designed to permit periodic inspection and functional testing of important areas and features to assess structural and leak tight integrity. The intermediate coolant boundary is within the scope of pre-service and in-service monitoring and non-destructive examination performed under the Reliability and Integrity Management Program described in chapter 8.”

The implementation summary in PSAR section 5.3.6.8 only discusses designing to permit access for periodic inspection and functional testing in accordance with the RIM Program, while section 7.1.4.1 also discusses “pre-service and in-service monitoring” under RIM. How does the Natrium design intend to meet item (2) from PDC 77 regarding “an appropriate material surveillance program for the intermediate coolant boundary”? Please provide available information based on the preliminary design regarding what components, materials, environments/locations and potential degradation would be covered by the material surveillance program."</td>
<td>5.3.6, “Additional PDC”, 7.1.4.1, “Intermediate Heat Transport System”</td>
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<td colspan="3">Chapter 7, “Descriptions for Safety-Significant SSCs”</td>
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<td>7-1</td>
<td>PSAR section 7.2.1, “Reactor Air Cooling System,” section 7.2.1.2, “Design Basis,” states that means are provided for monitoring radioactivity releases from the Reactor Air Cooling System (RAC).</td>
<td>7.2.1, “Reactor Air Cooling System”</td>
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	Activation of air is stated to result in N-16 and Ar-41 as gaseous effluents, that are evaluated in PSAR section 9.1. The NRC staff requests that USO provide information about its plan to monitor and quantify the gaseous effluent from the RAC.	
7-2	PSAR section 7.4.1, “Gaseous Radwaste Processing System”, section 7.4.1.2, “Design Basis,” under the description of regulatory guidance states that the applicant is in full conformance with RG 1.110, and that a cost benefit analysis is performed as required by 10 CFR 50.34a. Please provide this analysis for audit and indicate if the analysis results are provided in the PSAR?	7.4.1, “Gaseous Radwaste Processing System”
7-3	Clarify whether plant specific action items discussed in section 7.0, “Plant Specific Action Items” of the “RadICS Topical Report” safety evaluation (ML19233A177) will be addressed in the PSAR or in TR NAT-4950, “Instrumentation & Control Architecture and Design Basis Topical Report” that is currently under review by the NRC staff.	7.6.3.2, “Reactor Protection System Design Bases and Associated Safety Functions”
7-4	Clarify if any digital sensors will be used. If digital sensors will be used, please discuss common-cause failure regarding the digital sensors and specify whether that discussion is in the CP application or in NAT-4950.	7.6.5, “Reactor Instrumentation System”
7-5	PSAR section 7.6.7.1.1 states that the MCR contains equipment that allows operators to initiate or take manual control of functions associated with the RPS and Nuclear Island and Control System. Please discuss with the NRC staff to provide additional clarification regarding this statement.	7.6.7.1.1, “Plant Control”
7-6	The first sentence of the first paragraph of PSAR section 7.6.8.1 indicates that the Anticipatory Seismic Trip System (AST) provides actuation signals to Reactor Trip Breaker (RTBs) through an interface device. However, the AST inputs to the RTBs are not shown in Figure 5.1 of NAT-4950. Please discuss this discrepancy with the NRC staff.	7.6.8.1, “System Description, Architecture, and Equipment Locations”
7-7	Clarify how the AST signals are isolated from the RPS and clarify if the 125 VDC relay shown at top of RPS boxes is the isolating device.	7.6.8, “Anticipatory Automatic Seismic Trip System”
Chapter 8, “Plant Programs”		
8-1	For the commitment to RG 1.54, Revision 3, the PSAR states that “Regulatory Guide 1.54, Revision 3, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” is not used for Kemmerer Unit 1. No Service Level I, II, or III protective coatings are used in the design. Conformance to regulatory positions within RG 1.54 will be identified if Service Level I, II, or III protective coatings are identified as the design progresses.” This statement seems contradictory as it starts by saying that protective coatings are not used in the design, but then it opens the possibility that protective coatings may be used in the design if identified. If protective coatings are identified as the design progresses, confirm that any conformance or exceptions to the regulatory positions in RG 1.54 will be addressed. Conformance to the RG should be clear, assuming protective coatings are used.	8.1, “Quality Assurance”
8-2	For the commitment to RG 1.164, Revision 0, the PSAR states that Kemmerer Unit 1 fully conforms to Regulatory Guide 1.164, Revision 0, “Dedication of Commercial-Grade Items for Use in Nuclear Power Plants,” as shown in section 1.4.1. Regulatory Guide 1.164 Revision 0 is applicable to Kemmerer Unit 1 as a programmatic issue. Clarify what is meant by “[RG] 1.164 Revision 0 is applicable to Kemmerer Unit 1 as a programmatic issue.”	8.1

8-3	Section 8.1.2 states that "US SFR Owner, LLC (USO), a wholly owned subsidiary of TerraPower, LLC (TerraPower), is responsible for the establishment and execution of quality assurance program requirements. USO accomplishes this through agreement with TerraPower for TerraPower to be responsible for all activities for Kemmerer Unit 1 using the TerraPower quality assurance program description (QAPD). TerraPower may delegate the work of establishing and executing the quality assurance program, or any parts thereof, but retains responsibility for the quality assurance program." a) Explain how USO provides oversight to TerraPower if USO is utilizing the TerraPower QAPD. How does the TerraPower QAPD describe the USO organization as required by Criterion I, "Organization," of appendix B to 10 CFR Part 50? b) Explain what this "agreement" between USO and TerraPower consists of for the establishment and execution of the quality assurance program requirements.	8.1.2, "Plant Programs Regulatory Evaluation"
Chapter 9, "Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste"		
9-1	How are the RG 1.143 radwaste system design classifications determined for the Gaseous Radwaste Processing System (RWG), Liquid Radwaste Processing System (RWL), and Solid Radwaste Processing System (RWS)? The NRC staff requests access to supporting calculations or discussions for how USO determined the radwaste system classifications for its RWG, RWL, and RWS systems using the guidance contained in RG 1.143.	9.1, "Liquid and Gaseous Effluents"
9-2	Regarding PSAR section 9.1.3 "RWL Summary Description," the NRC staff requests that USO describe the methods for determining that liquid effluent is acceptable for evaporation. Specifically, the NRC staff requests clarification on if there will be any controls in place to establish representative sampling or monitoring of tanks, and to understand where the evaporated liquids are directed. From observations of Figure 9.1-1, "RWL Simplified Diagram," it points to a box labeled 'RWL Room Environment.' The NRC staff requests clarification on if this implies that the evaporated waste will be directed through the building ventilation system.	9.1
9-3	For the stated effluent dose results included in PSAR table 9.1-3, please provide input and output files for this analysis to facilitate understanding of the calculations. In addition, please provide any supporting calculation files that describe inputs and the justification, if available.	9.1
9-4	Does USO plan on addressing compliance with 10 CFR 20.1302(b)(i) for showing the annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in table 2 of appendix B to Part 20.	9.1
9-5	Are there any other release points for gaseous effluent besides the plant stack and the RAC? PSAR table 9.1-1 includes estimated release of tritium (H-3) from the cold salt tank cover gas and the steam generator. Where are those release points? Will there be effluent monitoring or sampling performed for this release point?	9.1
9-6	Regarding PSAR section 9.3 the NRC staff would like to understand the amount of waste storage space available for the facility and where	9.3, "Solid Radwaste Processing"

	solid waste is expected to be stored. Specifically, the NRC staff requests USO provide information on the following: • The currently estimated solid waste generation rates. • Expected locations for solid waste storage in the fuel handling building and accompanying discussions on how long it will take to fill the storage area. • Estimates on frequency of offsite waste shipments.	
Chapter 10, "Control of Occupational Dose"		
10-1	The NRC staff requests a discussion of the expected radiation sources within the facility. The NRC staff is looking to understand the areas of the facility where radiation sources are, where contamination may appear, and USO's plans for radiological controls around the facility. Regarding PSAR table 10.3-1, "Major Radiation Sources," the NRC staff requests that USO provide supporting information to help the NRC staff understand the radiation environment of the facility. To facilitate the discussion of radiation sources this information may include but is not limited to: • Using building maps to explain personnel access to the areas major radiation sources are located in. • Description of the estimated area dose rates for the areas where major radiation sources are located in. • Description of airborne radiation concerns, as well as the design features that may be used to limit them. • Shielding considerations around major radiation sources, • The controls in place to limit access to areas that these major radiation sources are in as well as expected occupancy level for those areas. • Current plans for radiation monitoring around these major radiation sources for both direct radiation monitoring and airborne radiation monitoring.	10
10-2	Please discuss the reason for not referencing RG 8.4, "Personnel Monitoring Device - Direct-Reading Pocket Dosimeters." The NRC staff is looking to understand if it is because USO has determined that it is not applicable to the design. If that is the reason what types of personnel monitoring are being considered by the facility?	10
10-3	Provide additional details about how the information contained in PSAR table 10.5-1, "Routine Online Operations Dose Assessment for Expected Occupancy," was determined. Is this generated based on estimates of expected work activities?	10
10-4	PSAR table 1.4-4 states that 10 CFR 50.34(f)(2)(vii), 10 CFR 50.34(f)(2)(xxvii), and 10 CFR 50.34(f)(2)(xxviii) are all addressed in PSAR section 10.1. However, no references or discussions are made to these items in section 10. The NRC staff requests a discussion on how these Three Mile Island (TMI) items are being addressed by the application.	10
Chapter 11, "Conduct of Operations"		
11-1	The NRC staff requests access to documentation of the consequence analysis performed to support the preliminary determination of the plume exposure pathway EPZ. This should include calculations, a description of the implementation of the plume exposure planning	11.3.3, "Emergency Planning Technical Evaluation"

	zone sizing methodology, as well as justifications for inputs and assumptions, computer code output, or other relevant documentation.	
11-2	Please clarify if the preliminary EPZ sizing consequence analysis considers security events consistent with the topical report methodology (ML23321A036).	Preliminary EPZ Technical Report
11-3	How will training programs that meet the requirements of 10 CFR 50.120 be established for the necessary categories of non-licensed nuclear plant personnel by 18 months prior to fuel load?	11.2, "Human Factors Engineering"
11-4	Aside from 10 CFR 50.34(f)(2)(iii) and 10 CFR 50.34(f)(2)(iv), how will the design of human system interfaces (HSI) and the HSI inventory satisfy the other technologically relevant, HSI-related requirements of 10 CFR 50.34(f)?	11.1.3, "Construction Organization"
11-5	How will the construction organization obtain personnel with sufficient experience and in sufficient numbers to provide adequate management and technical support for the facility?	11.1
11-6	What measures will ensure the control and preservation of documentation within the construction organization, including the documentation of communications?	11.1
11-7	What corporate officer within the organization has ultimate responsibility for nuclear safety matters and how will it be ensured that ancillary responsibilities will not detract from this?	11.1
11-8	What is the target date for completing the development of administrative procedures of a scope that would meet the regulatory guidance of RG 1.33, "Quality Assurance Program Requirements (Operation)?"	11.1
11-9	Will USO implement cybersecurity controls included in its Cybersecurity Plan (CSP) for a secure development and operational environment? If yes, which cybersecurity controls will be credited in support of the Secure Development and Operational Environment claims? If not, how does USO intend to meet the applicable regulatory evaluation criteria of RG 1.152 Revision 4, "Criteria for Programmable Digital Devices in Safety-Related Systems of Nuclear Power Plants," (ML23054A463) of a Secure Development and Operational Environment for the RPS?	11.6, "Fitness-for-Duty and Security"
11-10	USO indicated that it intends to provide the names and demographics of what appears to be approximately 1600 workers for the NRC staff to run the names through the Federal Bureau of Investigation (FBI) Threat Screening Center (TSC). Office of Nuclear Security and Incident Response (NSIR) has had a memorandum of understanding with the TSC since 2006 to do that for our operating Nuclear Power Plants (NPPs). We currently receive and submit that information from licensees via electronic transfer to the TSC on a monthly periodicity. In what format will USO provide the approximately 1600 names, etc. to the NRC staff.	11, "Conduct of Operations"
11-11	10 CFR 50.160(b)(1)(iv)(B)(9), "Drills and Exercises," states that the emergency plan should describe the drill and exercise program, with references to the process for testing and implementing major portions of the planning, preparations, capabilities, and coordination with offsite organizations to maintain the key skills of emergency responders. PSAR section 11.3.10, "Performance Monitoring," states, in part, that drills and exercises will be used to demonstrate the capabilities to perform and maintain emergency response functions as	11.3, "Emergency Preparedness"

	listed in 10 CFR 50.160(b)(1)(iii)(A) through (H) and performance metric and objectives for each of these functions will be developed and used for evaluation. Confirm that supporting organizations identified in the Kemmerer Emergency Plan as having an emergency response role, responsibility, or authority, will be requested to participate in scheduled drills and exercises. Describe the periodicity of those drills and exercises to include support organizations so that to maintain those key skills necessary for emergency responders.	