



SAFETY EVALUATION

Related to the
U.S. SFR Owner, LLC
Construction Permit Application for
the Kemmerer Power Station Unit 1

Completed: November 2025



ABSTRACT

This safety evaluation documents the U.S Nuclear Regulatory Commission (NRC) staff's technical review of the construction permit application submitted by TerraPower, LLC, on behalf of U.S. SFR Owner, LLC, under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," for Kemmerer Power Station Unit 1. The application is for the construction of a Sodium reactor plant proposed to be built in Lincoln County, Wyoming.

The reactor is intended to demonstrate the Sodium technology and will provide electricity generation capacity in the PacifiCorp service area. The Sodium reactor plant is an 840 megawatt thermal pool-type sodium-cooled fast reactor and includes a molten salt energy storage system that enables the plant to vary its supply of energy to the grid, up to 500 megawatt electric net, while maintaining constant reactor power.

This application documents the first use of a fully risk-informed, performance-based (RIPB) approach to establish the licensing basis for a commercial power reactor. Specifically, the application followed the Licensing Modernization Project (LMP) methodology which outlines a technology-inclusive, RIPB process covering event selection, safety classification of structures, systems, and components, and defense-in-depth evaluation.

The NRC's Advisory Committee on Reactor Safeguards independently reviewed those aspects of the application that concern safety and provided the results of its review to the Commission in a report dated November 16, 2025. Appendix C to this safety evaluation includes a copy of the report.

This safety evaluation presents the staff's review of the KU1 construction permit application based on information submitted by USO through November 2025. On the basis of its review of the construction permit application, the staff has determined that the preliminary design and analysis of the KU1 reactor, including the principal design criteria; design bases; information relative to materials of construction and general arrangement; and preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility: (1) provides reasonable assurance that the final design will conform to the design basis; (2) includes an adequate margin of safety; (3) describes the structures, systems, and components which will provide for the prevention of accidents and the mitigation of consequences of accidents; and (4) meets applicable regulatory requirements and satisfies applicable NRC guidance. Therefore, the staff recommends that the Commission make the necessary findings with respect to the safety of the construction permit in accordance with 10 CFR 50.35, "Issuance of construction permits"; 50.40, "Common standards"; and 50.50, "Issuance of licenses and construction permits."

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ABBREVIATIONS AND ACRONYMS

AC	Alternating current
ACA	Average concentration duration chemical accidents
ACI	American Concrete Institute
ACR	Air change rate
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act
AHX	Sodium-air heat exchanger
AIA	Aircraft impact assessment
AISC	American Institute of Steel Construction
ALARA	As low as reasonably achievable
ALI	Annual Limits on Intake
ALOHA	Areal locations of hazardous atmospheres
AMC	Auxiliary monitoring and control system
ANFO	Ammonium nitrate fuel oil
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
AoA	Area of applicability
AOF	Allocation of function
AOO	Anticipated operational occurrence
API	American Petroleum Institute
ARCAP	Advanced reactor content of application project
ARDP	Advanced Reactor Demonstration Program
ARIS	Assembly receipt inspection stand
ARM	Area radiation monitoring
AR5	Fifth Assessment Report
ASCE	American Society of Civil Engineers
ASCE/SEI	American Society of Civil Engineers/Structural Engineering Institute
ASD	Adjustable speed drive
ASE	Air stack structures and equipment
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
AST	Anticipatory automatic seismic trip system
ASTM	American Society for Testing and Materials
ATWS	Anticipated transient without scram
AWOS	Automated weather observing system
AWS	American Welding Society
BDBE	Beyond design basis event
BEPU	Best-estimate plus uncertainty
BiMA	Bi-metallic assembly
BLEVE	Boiling liquid expanding vapor explosion
BLTC	Bottom loading transfer cask
BPC	Bechtel Power Corporation
BPVC	Boiler and Pressure Vessel Code
BR	Basin and Range
BTP	Branch technical position
BTS	Battery transfer switch

BU	Burnup
BWR	Boiling water reactor
CAFTA	Computer Aided Fault Tree Analysis System
CAID	Core assembly identification
CAP	Core assembly pot
CATT	Core assembly transfer tube
CBS	Core barrel structures
CC	Climate change
CCA	Collector cylinder assembly
CCF	Common cause failure
CCS	Chemistry control system
CCWS	Component cooling water system
CEUS	Central and Eastern United States
CFD	Computational fluid dynamics
CFMS	Creep-fatigue monitoring system
CFR	<i>Code of Federal Regulations</i>
CFW	Condensate and feedwater system
CIP	Core inlet plenum
CLP	Cask loading pit
CLSM	Controlled low strength material
CMAA	Crane Manufacturers Association of America
CMTR	Certified material test reports
CN	Runoff curve number
CNO	Chief Nuclear Officer
COL	Combined license
COLR	Core operating limits report
COS	Climate observing station
COTS	Commercial off-the-shelf
CP	Construction permit
CPU	Central processing unit
CP-WC	Colorado Plateau-Wyoming Craton
CQC	Complete quadratic combination
CRA	Control rod assembly
CRBRP	Clinch River Breeder Reactor Project
CRD	Control rod drive system
CRDM	Control rod drive mechanism
CREAT	Climate resilience evaluation and awareness tool
CRS	Core restraint system
CT	Critical technology
CTB	Centennial Tectonic Belt
CTC	Coolant temperature monitoring and control system
CTE	Critical technology element
CU	Consolidated-undrained
CV	Cleaning vessel
CVAP	Comprehensive vibration assessment program
CVIS	Core-vessel interface structure
CW	Completely weathered
CWIS	Cooling water intake structure
DA	Data analysis
DAC	Derived Air Concentrations

DBA	Design basis accident
DBE	Design basis earthquake
DBHL	Design basis hazard level
DC	Direct current
DCF	Dose conversion factor
DCS	Distributed control system
DF	Decontamination factor
DG	Draft regulatory guide
DH	Downhole
DHRS	Decay heat removal system
DHS	U.S. Department of Homeland Security
DID	Defense-in-depth
DIF	Difficulty, importance, and frequency
DL	Defense line
DMA	Degradation mechanism assessment
DNS	Direct numerical simulation
DOE	U.S. Department of Energy
DOE-EM	U.S. Department of Energy - Environmental Management
DOF	Degrees of freedom
DOT	U.S. Department of Transportation
D-RAP	Design reliability assurance program
DRG	Design review guide
DRMS	Digital radiation monitor system
DRS	Design response spectra
DSRS	Design specific review standard
D/Q	Atmospheric deposition factor(s)
EA	Environmental assessment
EAB	Exclusion area boundaries
EAL	Emergency action level
EBR-II	Experimental Breeder Reactor II
ECCS	Emergency core cooling system
ED	Emergency Director
EDC	Energy island DC power supply system
EEQ	Electrical equipment qualification
EFP	Energy island fire protection system
EHT	Energy island heat tracing system
EI	Energy island
EIC	Energy island control system
EIS	Environmental impact statement
ELV	Energy island AC electrical power low voltage system
EM	Evaluation model
EMDAP	Evaluation Model Development and Assessment Process
EMM	Kemmerer Municipal Airport
E[M]	Expected momentum magnitude
EMC	Electromagnetic compatibility
EMV	Energy island AC electrical power medium voltage system
EOP	Emergency operating procedure
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	Emergency planning zone

EQ	Equipment qualification
ER	Environmental report
ERF	Emergency response facility
ERFB	Emergency response facility backup
ERFP	Emergency response facility primary
ERM	Effluent radiation monitor
ERO	Emergency response organization
ES	Event sequence
ESCS	Equipment and structural cooling subsystem
ESF	Event sequence families
ESP	Energy island standby AC power system
ESQ	Event sequence quantification
ESRP	Eastern Snake River Plain
ESS	Salt heat transport system
ESWR	Excessive sodium-water reaction
ET	Event tree
ETR	Energy transfer ratio
ETTP	East Tennessee Technology Park
EUP	Energy Island uninterruptible AC power supply system
EVHM	Ex-vessel handling machine
EVST	Ex-vessel storage tank
E[M]	Expected value of moment magnitude
FAA	U.S. Federal Aviation Administration
FAB	Fuel auxiliary building
FACP	Fire alarm control panel
F-C	Frequency-Consequence
FCP	Financial capacity plan
FE	Finite element
FEMA	U.S. Federal Emergency Management Agency
FFC	Failed fuel canister
FFD	Fitness-for-duty
FFR	Failed fuel rack
FFTF	Fast flux test facility
FFV	Fueling floor valves
FGD	Flue gas desulfurization
FGR	Federal guidance report
FHA	Fire hazards analysis
FHB	Fuel handling building
FHC	Fuel handling supervisory control system
FHCR	Fuel handling control room
FHE	Ex-vessel fuel handling system
FHI	In-vessel fuel handling system
FHP	Water pool fuel handling system
FHS	Fuel transport and storage system
FI	Fracture index
FIPS	Federal information processing standard
FIRS	Foundation input response spectra
FIS	Flood insurance study
FIVS	Fixed in-vessel shielding
FMCSA	Federal Motor Carrier Safety Administration

FMEA	Failure modes and effects analysis
FO	Fuel oil
FOA	Funding opportunity announcement
FOAK	First-of-a-kind
FOCD	Foreign ownership, control, or domination
FOM	Figure of merit
FONSI	Finding of no significant impact
FPC	Fuel pool cooling
FPP	Fuel pool purification
FQ	Financial qualifications
FR	Federal Register
FRA	Functional requirements analysis
FRMAC	Federal Radiological Monitoring and Assessment Center
FSAR	Final safety analysis report
FSF	Fundamental safety function
FTA	Fuel transfer adapter
FTL	Fuel transfer lift
FTLSP	Fuel transfer lift shield plug
FTP	Fuel transfer port
F-C	Frequency-consequence
GAO	Government Accountability Office
GDC	General design criteria
GEN	Generator system
GFIMS	Gas filtration impurities monitoring and sampling
GGRB	Greater green river basin
GL	Generic letter
GMC	Ground motion characterization
GMPM	Ground motion prediction model
GMRS	Ground motion response spectra
GP	Great Plains
GPCPD	Gallons per capita per day
GPS	Global positioning system
GSi	Geologic strength index
GUI	Graphical user interface
GV	Guard vessel
GVDS	Group-view display system
GVH	GE Vernova Hitachi Nuclear Energy
GWUS	Generic Western United States
HA	Human action
HAA	Head access area
HALEU	High-assay low-enriched uranium
HAZMAT	Hazardous materials
HCF	Hot channel factor
HCl	Hydrochloric acid
HEC-HMS	Hydrologic Engineering Center's Hydrologic Modeling System
HEC-RAS	Hydrologic Engineering Center's River Analysis System
HEPA	High efficiency particulate air
HEU	Highly enriched uranium
HF	Human factors
HFE	Human factors engineering

HFEITS	Human factors engineering issue tracking system
HFEPP	Human factors engineering program plan
HFWUA	Hams Fork Water Users Association
HIC	High integrity container
HMI	Human-machine interface
HMR	Hydrometeorological report
HPM	Human performance monitoring
HRA	Human reliability analysis
HRCS	Heat rejection control system
HRR	Heat rejection radiator
HRS	Heat rejection system
HS	Hazards screening
HSI	Human-system interface
HTF	Heat transfer fluid
HTFC	High temperature fission chamber
HTGR	High-temperature gas-cooled reactor
HVAC	Heating, ventilation, and air conditioning
HW	Highly weathered
IAC	Intermediate air cooling system
IAEA	International Atomic Energy Agency
IB	Idaho Batholith
IBC	International building code
ICC	International Code Council
ICRP	International Commission on Radiological Protection
ICS	Integrated control system
IDCOR	Industry degraded core rulemaking
IDLH	Immediately dangerous to life or health
IDP	Integrated decision-making process
IDPP	Integrated decision-making process panel
IDU	Isolation damper units
IE	Initiating event
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IGS	Inert gas system
IHT	Intermediate heat transport system
IHTCS	Intermediate heat transport control system
IHTS	Intermediate heat transport system
IHX	Intermediate heat exchanger
IMS	Inventory management system
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IO	Input-output
IPCC	Intergovernmental Panel on Climate Change
ISA	International Society of Automation
ISB	Intermountain seismic belt
ISG	Interim staff guidance
ISO	International Organization for Standardization
ISP	Intermediate sodium pump
ISRM	International Society for Rock Mechanics
ISRS	In-structure response spectra

IST	In-service testing
ISU	Isolation spring units
IVDC	In-vessel drive controller
IVS	In-vessel storage
IVTM	In-vessel transfer machine
I&C	Instrumentation and control
KM	Kemmerer Mine
KU1	Kemmerer Power Station Unit 1
LANL	Los Alamos National Laboratory
LBB	Leak before break
LBE	Licensing basis event
LCI	Lettis Consultants International
LCO	Limiting condition for operation
LCS	Local control station
LDA	Lead demonstration assembly
LEL	Lower explosive limit
LEU	Low-enriched uranium
LFL	Lower flammable limit
LFRS	Lateral force resisting system
LiDAR	Light detection and ranging
LIP	Local intense precipitation
LMP	Licensing modernization project
LOCA	Loss of coolant accident
LOOP	Loss of offsite power
LPSD	Low-power shutdown
LPZ	Low population zone
LRFD	Load and resistance factor design
LSSS	Limiting safety system setting
LTA	Lead test assembly
LTP	Lead test pin
LWA	Limited work authorization
LWR	Light water reactor
L&C	Limitations and conditions
MANDE	Monitoring and non-destructive examination
MANDEEP	Monitoring and non-destructive examination expert panel
MAR	Material at risk for release
MCA	Maximum concentration chemical accidents
MCC	Motor control center
MCE _R	Maximum Considered Earthquake
MCNP	Monte Carlo N-Particle Code
MCR	Main control room
MCT	Maximum cladding temperature
MCV	Movable closure valve module
MC&A	Material control and accounting
MEI	Maximally exposed individual
MHA	Maximum hypothetical accident
MIRSS	Modular isolated reactor support structure
MPS	Multi purpose sampler
MRD	Modulus reduction and damping
MRE	Most-recent earthquake

MSL	Mean sea level
MSRE	Molten salt reactor experiment
MSS	Manufacturers Standardization Society
MST	Mechanistic source term
MW	Moderately weathered
NAAQS	National ambient air quality standards
NACP	Sodium leak detection control panel
NAD	North American Datum
NAP	North Ash Pond
NAPS	New assembly preconditioning station
NAVD	North America Vertical Datum
NBSIR	National Bureau of Standards
NCA5	Fifth National Climate Assessment
NCB	Nuclear island control building
NCDC	National Climatic Data Center
NCEI	National Centers for Environmental Information
NCH	Nuclear island cranes and hoists
NDC	Nuclear island DC power supply system
NEA	Nuclear island ancillary electrical system
NEB	Nuclear island electrical building
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NEPA	National Environmental Protection Act
NES	Nuclear island auxiliary electrical system
NESC	National Electrical Safety Code
NFA	Nuclear island fire detection and alarm system
NFD	Nuclear island fire water distribution system
NFP	Nuclear island fire protection system
NFPA	National Fire Protection Association
NFS	Nuclear island fire suppression system
NG	Noble gas
NGA	Nuclear island argon gas distribution and storage
NGA-East	Next Generation Attenuation for Central and Eastern North America
NGI	Norwegian Geotechnical Institute
NGL	Nuclear island grounding and lightning protection
NGS	Nuclear island air and gas distribution system
NGVD 29	National Geodetic Vertical Datum of 1929
NHT	Nuclear island heat tracing system
NHV	Nuclear island heating, ventilation, and air conditioning
NI	Nuclear island
NIC	Nuclear island control system
NIOSH	National Institute for Occupational Safety and Health
NLS	Nuclear island lighting system
NLV	Nuclear island AC electrical power low voltage
NME	Nuclear island major maintenance equipment
NMFS	New Madrid Fault Source
NMV	Nuclear island AC electrical power medium voltage
NNA	Sodium leak detection, collection, and containment system
NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission

NRCS	Natural resources conservation service
NRGR	Northern Rio Grande Rift
NSC	Nuclear island security system
NSHMP	National Seismic Hazard Mapping Project
NSHM23	National seismic hazard model 2023
NSRST	Non-safety-related with special treatment
NSS	Nuclear island salt system
NST	Non-safety-related with no special treatment
NSTF	Natural convection shutdown heat removal test facility
NUMARC	Nuclear Management and Resources Council
NUP	Nuclear island uninterruptible AC power supply system
NWS	Nuclear island water system
NYS	Nuclear island yard system
N/A	Not Applicable
OBE	Operating basis earthquake
OCC	Outage control center
ODCM	Offsite dose calculation manual
OE	Operating experience
OER	Operating experience review
OL	Operating license
OQE	Other quantified event
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
OW	Observation well
PAC	Protective action criteria
PAM	Post-accident monitoring
PAV	Pool auxiliary vault
PBR	Pebble bed reactors
PCO	Plant communication system
PCP	Process control program
PCS	Plant control system
PDC	Principal design criteria
PDR	Public document room
PEER	Pacific earthquake engineering research
PEM	Pin extraction machine
PEP	Plume exposure pathway
PFD	Probability of failure on demand
PFM	Probabilistic fracture mechanics
PGA	Peak ground acceleration
PGM	Plant General Manager
PGS	Power generation system
PHF	Plug handling fixture
PHM	Pool handling machine
PHMSA	Pipeline and Hazardous Materials Safety Administration
PHSS	Pebble handling and storage system
PHT	Primary heat transport system
PHTCS	Primary heat transport control system
PI	Plasticity index
PIC	Pool immersion cell
PIE	Postulated initiating event

PITAP	Post-construction inspection, testing, and analysis program
PLA	Penalized-likelihood approach
PLOF	Protected loss of flow
PLTMS	Primary loop thermal management system
PM	Particulate matter
PMC	Plant monitoring and control system
PMF	Probable maximum flood
PMP	Probable maximum precipitation
PMWP	Probable maximum winter precipitation
PNL	Pacific Northwest Laboratory
PNPP	Pacific Naughton Power Plant
POS	Plant operating state
PRA	Probabilistic risk assessment
PRC	Pin removal cell
PRISM	Power Reactor Innovative Small Module
PRM	Process radiation monitoring
PSAR	Preliminary safety analysis report
PSF	Probabilistic risk assessment safety function
PSHA	Probabilistic seismic hazard analysis
PSP	Primary salt pump
PSRA	Probabilistic site response analysis
PTI	Plugging temperature indicator
PZGW	Piezometer (groundwater)
PZSW	Piezometer (surface water)
P-S	Compressional (P) and shear (S) wave downhole
QA	Quality assurance
QAP	Quality assurance program
QAPD	Quality assurance program description
QATR	Quality assurance topical report
QFFD	Quaternary fault and fold database
QHO	Quantitative health objective
RAB	Reactor auxiliary building
RAC	Reactor air cooling system
RAHS	Reactor auxiliary heating system
RAI	Request for additional information
RAP	Reliability assurance program
RBHVAC	Reactor building heating, ventilation and air conditioning
RC	Radiological consequence
RCA	Radiologically controlled area
RCACS	Reactor coolant auxiliary control system
RCC	Reactor core system
RCI	Request for confirmation of information
RCP	Representative concentration pathways
RCS	Reactor control system
RCSS	Reactivity control and shutdown system
REMP	Radiological environmental monitoring program
RES	Reactor enclosure system
RFDC	Required functional design criteria
RFI	Reactor fixed internals
RG	Regulatory guide

RH	Reactor head
RI	Risk Integration
RIC	Radiation indicator controller
RIL	Research Information Letter
RIM	Reliability and integrity management
RIMEP	Reliability and integrity management expert panel
RIPB	Risk-informed, performance-based
RIS	Reactor instrumentation system
RN	Radionuclide
RMBS	Remote manual bypass switch
RMC	Rod monitoring and control system
RMR	Rock mass rating
RMS	Radiation monitoring system
ROSP	Remote onsite shutdown panel
RQD	Rock quality designation
RP	Radiation protection
RPA	Rotatable plug assembly
RPD	Rotatable plug drive
RPJ	Rotatable plug jack
RPP	Radiation protection program
RPS	Reactor protection system
RQD	Rock quality designation
RSB	Reactor support block
RSC	Remote shutdown complex
RSF	Required safety functions
RSS	Reactor support structure
RTB	Reactor trip breaker
RTMS	Reactor thermal management system
RTS	Reactor trip system
RV	Reactor vessel
RVACS	Reactor vessel auxiliary cooling system
RVC	Reticulated vitreous carbon
RVH	Reactor Vessel Head
RVSS	Reactor vessel support system
RVT	Regulating voltage transformer
RWG	Gaseous radwaste processing system
RWL	Liquid radwaste processing system
RWS	Solid radwaste processing system
RXB	Reactor building
R&D	Research and development
SAMG	Severe accident management guideline
SAP	South Ash Pond
SARA	Superfund Amendments and Reauthorization Act
SARRDL	Specified acceptable system radionuclide release design limit
SASW	Spectral analysis of surface waves
SC	Success criteria
SCADA	Supervisory, control and data acquisition
SCG	Sodium cover gas system
SD	Standard deviation
SDC	Seismic design category

SDG	Standby diesel generator
SDOE	Secure development and operational environment
SE	Safety evaluation
SEFOR	Southwest Experimental Fast Oxide Reactor
SEI	Structural Engineering Institute
SF	Spent fuel
SFCS	Spent fuel cooling system
SFP	Spent fuel pool
SFR	Sodium-cooled fast reactor
SGI	Safeguards Information
SGS	Steam generation system
SH	SASW High
SHT	Sodium holding tank
SHX	Sodium-salt heat exchanger
SIS	Seismic isolation system
SL	Safety limit
SMACNA	Sheet Metal and Air Conditioning Contractors' National Association
SMR	Small modular reactor
SMS	Seismic monitoring system
SNM	Special nuclear material
SOV	Solenoid-operated valve
SPDS	Safety parameter display system
SPID	Screening, prioritization, and implementation details
SPRA	Seismic probabilistic risk assessment
SPS	Sodium processing system
SPS-E	Ex-vessel storage tank sodium processing system
SPS-I	Intermediate Sodium Processing System
SPS-P	Primary Sodium Processing System
SPT	Standard penetration test
SR	Safety-related
SRC	Stress relaxation cracking
SRDC	Safety-related design criteria
SRE	Sodium reactor experiment
SRP	Standard review plan
SRSS	Square root of the sum of squares
SS	Safety-significant
SSC	Structures, systems, and components
SSE	Safe shutdown earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil-structure interaction
SSSI	Structure-soil-structure interaction
STP	Standard temperature and pressure
STS	Steam turbine system
SVDU	Special video display unit
SW	Slightly weathered
SWUS	Southwestern United States
SWUS-DCPP	Southwestern United States for Diablo Canyon
SWUS-PV	Southwestern United States for Palo Verde
SY	Systems Analysis
TA	Task Analysis

TATNF	Time-at-temperature no-failure
TEDE	Total effective does equivalent
TEEL	Temporary emergency exposure limits
TEMA	Tubular Exchanger Manufacturers Association
TFB	Turbine Facility Building
TFF	Sodium Test and Fill Facility
TI	Technical Integration
TI-RIPB	technology-inclusive, risk-informed, and performance-based
TMI	Three Mile Island
TMP	Technology maturation plan
TMS	Tritium management system
TNT	Trinitrotoluene
TR	Topical report
TRA	Technology readiness assessment
TRISO	Tristructural isotropic
TRL	technology readiness level
TS	Technical specification
TSS	Thermal Salt Storage System
TSTF	Technical Specification Task Force
TVA	Tennessee Valley Authority
TWA	Time Weighted Average
TVSS	Transient Voltage Surge Suppressor
TWR®	Traveling Wave Reactor
UCO	Uranium oxycarbide
UFC	Unified facilities criteria
UFL	Upper flammability limit
UGRB	Upper Green River Basin
UHRS	Uniform Hazard Response Spectra
UHS	Ultimate Heat Sink
UIS	Upper Internal Structure
UL	Underwriters Laboratories
UMC	Utility monitoring and control system
UPS	Uninterruptible power supplies
URD	Utility requirements document
USACE	United States Army Corps of Engineers
USCB	United States Census Bureau
USCS	Unified Soil Classification System
USGCRP	United States Global Change Research Program
USGS	United States Geological Survey
USI	Unresolved safety issues
USL	Upper subcritical limit
USO	US SFR Owner, LLC
UTM	Universal Transverse Mercator
UU	Unconsolidated-undrained
UV	Undervoltage
VCE	Vapor cloud explosion
VDU	Visual display unit
VHRA	Very high radiation area
VRLA	Valve-regulated lead-acid
V _s	Shear-Wave Velocity

V _s 30	Time-averaged shear-wave velocity in the upper 30
VTC	Vapor trap cell
V&V	Verification and validation
WGUEP	Working Group on Utah Earthquake Probabilities
WOHS	Wyoming Office of Homeland Security
WSEO	Wyoming State Engineer's Office
WUS	Western United States
WWDO	Wyoming Water Development Office
X/Q	Atmospheric dispersion factor
XIS	Nuclear instrumentation system
YC	Yellowstone Caldera
YGA	Yellowstone Geoid Anomaly

1 THE FACILITY

This chapter of the safety evaluation (SE) provides a general introduction to the facility and an overview of the topics covered in detail in other chapters of this SE, including areas of review, regulatory criteria and guidance, review procedures and findings, and conclusions.

1.1 Introduction

This SE documents the results of the U.S Nuclear Regulatory Commission (NRC) staff's (staff) technical review of the construction permit (CP) application submitted by TerraPower, LLC (TerraPower) on behalf of U.S. SFR Owner, LLC (USO), under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," for the Kemmerer Power Station Unit 1 (KU1) power reactor proposed to be built in Lincoln County, Wyoming. The proposed reactor is a non-light-water reactor (non-LWR). An environmental review was also performed for the KU1 CP application, and its evaluation and conclusions are documented in a Final Environmental Impact Statement, published on October 21, 2025, as NUREG-2268 "Environmental Impact Statement for the Construction Permit Application for Kemmerer Power Station Unit 1" to the Federal Register (FR) (90 FR 48507).

The staff acknowledged receipt of USO's application for a CP by letter on May 14, 2024 (ML24127A183) and published notice in the FR on May 21, 2024 (89 FR 44715). The KU1 CP application, including the environmental report, was submitted by letter dated March 28, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24088A060). Among other things, the following were included in the KU1 CP application:

- A preliminary safety analysis report submitted to meet the requirements of 10 CFR 50.34, "Contents of applications; technical information," paragraph (a), "Preliminary safety analysis report."
- Environmental report submitted to meet the requirements of 10 CFR 50.30, "Filing of applications for licenses; oath or affirmation," paragraph (f).
- General information submitted to meet the requirements of 10 CFR 50.33, "Contents of applications; general information."

The staff conducted an acceptance review for docketing of USO's KU1 CP application and, by letter dated May 21, 2024 (ML24135A109), determined that USO's KU1 CP application was complete and acceptable for docketing. The application was assigned Docket Number (No.) 50-613. A notice of docketing of USO's KU1 CP application was published in the FR on June 4, 2024 (89 FR 47997).

The staff's SE of the CP application to construct the 10 CFR Part 50 utilization facility is based on information in the application, as revised and supplemented. Unless otherwise stated, this SE evaluates the information contained in:

- The original application dated March 28, 2024 (ML24088A060); as supplemented by Revision (Rev.) 1, dated October 3, 2025 (ML25276A288)

- Supplements to the original application, dated:
 - June 17, 2025 (ML25171A021) “Transmittal of Probabilistic Site Response Analysis Calculation of Ground Motion Response Spectra and Safe Shutdown Earthquake Spectra;”
 - September 9, 2025 (ML25251A127) “Revision to Exemption Request from 10 CFR 50.33(f) and 10 CFR 50 Appendix C Financial Qualification Documentation Requirements and Revision to the General and Financial Information;”
 - September 15, 2025 (ML25259A175) “Intermediate Heat Exchanger Tube-to-Tubesheet Welds Design Information;”
 - September 16, 2025 (ML25259A180) “Request for Confirmation of Information;”
 - September 17, 2025 (ML25260A002) “Response to NRC Audit Question 3-85;”
 - October 3, 2025 (ML2576A027) “DOE Confirmation of Active and Good Faith Negotiations for Disposal Contract Letter to USO;”
 - October 1, 2025 (ML25274A130) “Natrium Demonstration Plant Long-Lived Passive Structural Materials of Construction Selection and Development;”
 - October 1, 2025 (ML25274A124) “Research and Development Supplemental Information;”
 - September 10, 2025 (ML25253A386) “Preventive Measures Classification Methodology and Preliminary Results;”
 - July 23, 2025 (ML25205A087) Transmittal of TerraPower, LLC, “Natrium Demonstration DID Evaluation Report,” NAT-4770 Rev. 1;
 - April 7, 2025 (ML25108A080) “Transmittal of Response to NRC Audit Question 3-77 on KU1 PSAR;”
 - January 23, 2025 (ML25028A117) “Transmittal of Responses to NRC Audit Questions on KU1 PSAR;”
 - January 3, 2025 (ML25003A162) “Regulatory Interpretation of the Applicability of 10 CFR 50.10 and 10 CFR 51.4 Definitions of Activities Constituting Construction to the Installation of Conduit and Cable Trays;”
 - December 19, 2024 (ML25016A155) Transmittal of Responses to NRC Audit Questions on KU1 PSAR;”
 - October 28, 2024 (ML24310A087) Transmittal of Responses to NRC Audit Questions on KU1 PSAR”; and

- September 6, 2024 (ML24253A220) “Transmittal of Responses to NRC Audit Questions on KU1 PSAR.”

1.1.1 Areas of Review

The KU1 CP application review consisted of two concurrent reviews: (1) a safety review of the KU1 preliminary safety analysis report (PSAR) and supporting technical information, and (2) an environmental review of the KU1 Environmental Report. The staff reviewed the PSAR and supporting technical information against applicable regulatory requirements using appropriate regulatory guidance and standards, as discussed below, to assess the sufficiency of the preliminary design of the KU1 power reactor. As part of this review, the staff evaluated descriptions and discussions of KU1’s structures, systems, and components (SSCs), with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. The staff also evaluated the preliminary design of KU1 to ensure the sufficiency of principal design criteria (PDC), design bases, and information relative to materials of construction, general arrangement, and approximate dimensions to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. In addition, the staff reviewed USO’s identification and justification for the selection of those variables, conditions, or other items that USO determined to be probable subjects of technical specifications for the facility in accordance with 10 CFR 50.34(a)(5). The staff also reviewed USO’s evaluation of the SSCs to ensure that they would adequately provide for the prevention of accidents and the mitigation of consequences of accidents. The staff considered the preliminary analysis and evaluation of the design and performance of the SSCs of the KU1 facility which the applicant prepared with the objective of assessing the risk to public health and safety resulting from operation of the facility.

1.1.2 Regulatory Basis, Acceptance Criteria, and Exemptions

The staff reviewed the PSAR and supporting technical information against applicable regulatory requirements, using appropriate regulatory guidance and standards, to assess the sufficiency of the preliminary facility design and analysis for the issuance of a CP.

In accordance with paragraph (a) of 10 CFR 50.35, “Issuance of construction permits,” the Commission may issue a CP if the Commission finds that:

1. the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;
2. such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report (FSAR);
3. safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that

4. on the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, "Reactor Site Criteria," the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

As provided in 10 CFR 100.2, "Scope," the siting requirements in 10 CFR Part 100 "apply to applications for site approval for the purpose of constructing and operating stationary power and testing reactors pursuant to the provisions of [10 CFR Part 50]." The KU1 application is for a CP for a stationary power reactor. Therefore, the staff evaluated the characteristics of the proposed KU1 site using the applicable criteria in 10 CFR Part 100, in addition to those in 10 CFR Part 50. The staff's review evaluated the geography and demography of the site; nearby industrial, transportation, and military facilities; site meteorology; site hydrology; and site geology, seismology, and geotechnical engineering to determine whether issuance of the CP would be inimical to the public health and safety. The staff's review also evaluated SSCs and equipment designed to ensure safe operation, performance, and shutdown when subjected to extreme weather, floods, seismic events, missiles (including aircraft impacts), chemical and radiological releases, and loss of offsite power.

The CP, if issued, would constitute an authorization for USO to proceed with construction but would not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated into the permit. USO did not request such approval here. Such approval, if appropriate, would be made following the evaluation of the final design of the facility, as described in the FSAR as part of USO's operating license (OL) application for KU1, should the applicant apply for an OL¹.

In addition to the findings listed in 10 CFR 50.35, a CP application must also provide sufficient information to allow the Commission to make the following determinations in accordance with 10 CFR 50.40, "Common standards," and 10 CFR 50.50, "Issuance of licenses and construction permits":

1. There is reasonable assurance: (i) that the construction of the facility will not endanger the health and safety of the public, and (ii) that construction activities can be conducted in compliance with the Commission's regulations.
2. The applicant is technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
3. The applicant is financially qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.

¹ When this safety evaluation references an operating license application, review of an operating license application, operating license stage of the review, or other similar term that involves an application for or review of an operating license for KU1, it should be understood that this review will only be conducted if a KU1 operating license is applied for and accepted for review by the NRC. USO indicated in enclosure 1 to its application that it intends to apply for an OL.

4. The issuance of a permit for the construction of the facility would not be inimical to the common defense and security or to the health and safety of the public.
5. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering reasonable available alternatives, the issuance of this CP, subject to the conditions for protection of the environment set forth herein, is in accordance with Subpart A of 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
6. The application meets the standards and requirements of the Atomic Energy Act and the Commission's regulations, and that notifications, if any, to other agencies or bodies have been duly made.

The staff's evaluation of the KU1 preliminary design and analysis was based primarily upon the following 10 CFR requirements:

- 10 CFR 50.30, "Filing of application; oath or affirmation;"
- 10 CFR 50.33, "Contents of applications; general information;"
- 10 CFR 50.34, "Contents of applications; technical information," particularly 10 CFR 50.34(a), "Preliminary safety analysis report;"
- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors;"
- 10 CFR 50.35, "Issuance of construction permits;"
- 10 CFR 50.40, "Common standards;"
- 10 CFR 50.50, "Issuance of licenses and construction permits;"
- 10 CFR 50.55, "Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses;"
- 10 CFR 50.55a, "Codes and standards;"
- 10 CFR 50.150, "Aircraft impact assessment;"
- 10 CFR Part 20, "Standards for Protection against radiation;"
- 10 CFR Part 26, "Fitness for duty programs," Subpart K, "FFD [Fitness for Duty] Programs for Construction;" and
- 10 CFR Part 100, "Reactor site criteria."

The regulations of 10 CFR 50.40 require that:

the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in part 20 of this chapter, and that the health and safety of the public will not be endangered.

With respect to 10 CFR Part 20, which is referred to in 10 CFR 50.40, the staff assessed whether USO had identified the relevant requirements for an operating facility and provided descriptions of the preliminary facility design to determine whether the PSAR provides an acceptable basis for the development of SSCs, and whether there is reasonable assurance that USO will comply with the regulations in 10 CFR Part 20 during KU1 facility operation. Because USO has not applied for licenses to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material in accordance with 10 CFR Part 30, “Rules of general applicability to domestic licensing of byproduct material,” 10 CFR Part 40, “Domestic licensing of source material,” and 10 CFR Part 70, “Domestic licensing of special nuclear material,” respectively, or a license to operate a production or utilization facility under 10 CFR Part 50, the requirements of 10 CFR Part 20 do not apply at this time. As such, the staff did not evaluate the application against the requirements in 10 CFR Part 20, but such evaluation would occur for the OL application, should the applicant apply for an OL.

As required by 10 CFR 50.34(a)(3)(i), USO must describe the PDC for KU1 in the PSAR. PDC for a Sodium Advanced Reactor were developed by TerraPower in topical report (TR) NATD-LIC-RPRT-0002-A, which the NRC approved (ML24283A066). This TR is incorporated by reference in PSAR section 5.3 which repeats the PDC from the approved TR and provides additional contextual information on how the PDC are implemented in the KU1 design. The staff’s evaluation of this incorporation by reference is in section 5.3 of this SE.

USO used the licensing modernization project (LMP)² methodology described in Nuclear Energy Institute (NEI) 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development” (ML19241A472), as endorsed by the NRC in Regulatory Guide (RG) 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform, the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors” (ML20091L698). These guidance documents define risk-informed, performance-based, and technology-inclusive processes for the selection of licensing basis events (LBEs); safety classification of SSCs; and the determination of defense-in-depth (DID) adequacy for non-light-water reactors (non-LWRs). NEI 18-04 provides a frequency consequence target curve that is used to assess events, SSCs, and programmatic controls. LBEs are categorized by the frequency of occurrence, separated into anticipated operational occurrences, design-basis events (DBEs), and beyond-design-basis events. Because the LMP methodology is a novel approach used for the first time for a commercial power reactor in this application, an orientation to the NEI 18-04 process and how it is reflected in the structure of the PSAR is provided in SE section 1.3.3.3.1 below.

USO followed the format of NEI 21-07, “Technology Inclusive Guidance for Non-Light Water Reactors Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology,”

² This methodology is referred to interchangeably throughout this SE as the NEI 18-04 process, the LMP process, the NEI 18-04 methodology, or the LMP methodology

Rev. 1 (ML22060A190) to develop portions of the PSAR. NEI 21-07 describes the scope and level of detail in specific portions of the first eight chapters of a PSAR that are associated with LMP-based safety analysis. The staff endorsed NEI 21-07 as one acceptable approach to develop portions of the first eight chapters of the PSAR in RG 1.253 “Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Advanced Reactors,” Rev. 0 (ML23269A222).

DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications – Roadmap” (ML23277A139) provides guidance for the staff review of new non-LWR applications following the LMP methodology. Specifically, it includes (1) a general overview of the information that should be included in a non-LWR application submitted under 10 CFR Part 50 or 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants;” (2) a review roadmap for NRC staff with the principal purpose of ensuring consistency, quality, and uniformity of the staff reviews; and (3) a well-defined base from which the staff can evaluate proposed differences in the scope of reviews (e.g., CP versus OL). RGs 1.233 and 1.253 as well as other key guidance documents are referenced in DANU-ISG-2022-01.

As appropriate, the staff used additional guidance and codes and standards (e.g., NRC RGs, Institute of Electrical and Electronics Engineers standards, and American National Standards Institute/American Nuclear Society (ANSI/ANS) standards) in its review of the KU1 CP application.

Exemptions

Enclosure 4 of USO’s KU1 CP application included four exemption requests associated with 10 CFR Part 50 regulations. The requested exemptions are described and addressed in appendix B and chapter 14 of this SE. An additional staff-initiated exemption is also included in appendix B of the SE.

1.1.3 Review Procedures

USO’s KU1 CP application only seeks authorization to construct the proposed KU1 facility. Accordingly, the KU1 design may be adequately described at a functional or conceptual level in the PSAR. As stated throughout the PSAR, USO will include additional design and analysis details with its FSAR as part of its OL application, should it apply for an OL.

The objective of the staff’s evaluation was to assess the sufficiency of information contained in the KU1 application for the issuance of a CP in accordance with the requirements of 10 CFR Part 50. An in-depth evaluation of the KU1 design will be performed following the docketing of an OL application and its accompanying FSAR, should USO apply for an OL.

1.1.4 Resolving Technical Issues

The staff may use a variety of approaches to resolve technical issues (e.g., regulatory audit, requests for confirmation of information (RCIs), requests for additional information (RAIs)). An applicant may also voluntarily supplement the application on its own initiative to provide additional information.

For this review, the staff conducted a general audit to enhance the staff's understanding of USO's application enabling the timely identification of supplemental information that required docketing to support the staff's review. During the audit, USO provided clarifications through its responses to the staff's questions and submitted updates to the PSAR and docketed supplements to the application. The results of the staff's audit of the KU1 CP application are available at ML25302A443.

During its review of the KU1 CP application, the staff also prepared and issued three RCIs (ML25261A106) and the applicant responded to them (ML25259A180).

1.1.5 Ongoing Research and Development

The provisions of 10 CFR 50.34(a)(8) allow for ongoing research and development (R&D) to confirm the adequacy of the design of SSCs to resolve safety questions prior to the completion of construction. In accordance with 10 CFR 50.34(a)(8), USO identified several R&D activities, which are described in PSAR chapter 13. Chapter 13 of this SE evaluates these activities.

1.1.6 Advisory Committee on Reactor Safeguards Review

To support the Advisory Committee on Reactor Safeguards (ACRS) in providing an independent review and report to the Commission regarding the KU1 CP application, the staff presented the results of its SE to the ACRS subcommittee on October 8-9 and 22-23, 2025. The ACRS full committee met on November 5-6, 2025, to complete its review. After the meetings, to meet the requirements of 10 CFR 50.58, "Hearings and report of the Advisory Committee on Reactor Safeguards," the ACRS issued a letter report to the Commission with its conclusions and recommendations regarding the KU1 CP application. The ACRS letter is provided in appendix C of this SE.

1.1.7 Application Availability

Publicly available documents related to the KU1 CP application may be obtained online in the ADAMS public documents collection at <https://www.nrc.gov/reading-rm/adams.html>.³ For assistance with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209 or by email to PDR.Resource@nrc.gov.

The versions of the PSAR are publicly available in ADAMS. Other public documents and correspondence related to this application may be found by searching KU1's Docket Number, 05000613, in ADAMS. Portions of the application or correspondence containing sensitive information (e.g., proprietary information) are withheld from public disclosure pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

1.1.8 NRC Staff Contact Information

The project manager for this SE was Mallecia Sutton-Padmore, Senior Project Manager, Division of Advanced Reactors and Non-power and Utilization Facilities, U.S. Nuclear Regulatory Commission. Ms. Sutton-Padmore may be contacted regarding this SE at 301-415-

³ This is the new link that replaces Web Based ADAMS (WBA) (<https://adams-search.nrc.gov/home>). WBA is scheduled to be disabled in early December 2025.

0673 or via email at Mallecia.Sutton@nrc.gov. Appendix E to this SE provides a listing of principal contributors, including areas of technical expertise and chapters of authorship.

1.2 Summary and Conclusions on Principal Safety Considerations

The staff evaluated the descriptions and discussions of the proposed KU1 facility, as described in USO's CP application, as supplemented. Based on its review, the staff makes the following findings:

1. Applicable standards and requirements of the Atomic Energy Act and Commission regulations have been met.
2. The application generally conforms to the applicable guidance for a CP applicant for a commercial non-LWR, with acceptable departures as noted in this SE.
3. Required notifications to other agencies or bodies related to this licensing action have been duly made.
4. Based on the preliminary design of the facility, there is reasonable assurance that the final design will conform to the design basis with adequate margin for safety.
5. There is reasonable assurance that the facility can be constructed in conformity with the permit, the provisions of the Atomic Energy Act, and the Commission's regulations.
6. The staff has evaluated the accident analyses presented by USO in the PSAR and determined that the calculated potential radiation dose consequences outside the KU1 site from postulated accidents are not likely to exceed the dose guidelines of 10 CFR 50.34(a)(1)(ii)(D) using site atmospheric dispersion characteristics as required by 10 CFR Part 100. Furthermore, SSCs have been designed to provide for the prevention of accidents and the mitigation of consequences of accidents.
7. Releases of radioactive materials and wastes from the facility are not expected to result in concentrations outside the limits specified by 10 CFR Part 20, Subpart D, "Radiation Dose Limits for Individual Members of the Public," and are as low as is reasonably achievable.
8. The financial information, technical analyses, programs, and organization described in the application, as supplemented, demonstrate that USO is financially and technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
9. The preliminary emergency plan provides reasonable assurance that USO will be prepared to assess and respond to emergency events.
10. The application presents information at a level of detail that is appropriate for general familiarization and understanding of the proposed facility.
11. The application describes the relationship of specific facility design features to reactor operation.

12. Issuance of the CP will not be inimical to the common defense and security or to the health and safety of the public.

Therefore, the staff finds that, subject to certain conditions, the preliminary design and analysis of KU1, as described in the PSAR, is, where relevant, consistent with guidance and is sufficient and meets the applicable regulatory requirements for the issuance of a CP in accordance with 10 CFR 50.35.

The staff notes that there are some remaining areas of uncertainty in the KU1 design and analysis, as discussed throughout this SE. This is acceptable under the requirements of 10 CFR 50.34(a) and 10 CFR 50.35, which state that the expected design and analysis is preliminary and that outstanding safety questions may be resolved through R&D efforts. The staff also notes that, per 10 CFR 50.35(b), a CP does not constitute Commission approval of any design feature unless specifically requested and incorporated into the permit; no such approval was requested in the application. As stated in this SE where applicable, further technical information or design information required to complete the safety analysis can reasonably be left for later consideration in the FSAR. USO stated in the PSAR and supplements where they intend to provide such information at the OL stage of the review. The staff notes that resolution of these items is not necessary for the issuance of a CP.

In PSAR chapter 13, USO identified several ongoing R&D activities to confirm the adequacy of the design of SSCs to resolve safety questions prior to the completion of construction. The staff is tracking these activities and, if an OL application is submitted, will ensure they are appropriately completed pursuant to 10 CFR 50.34(b)(5). Additionally, appendix A to this SE identifies certain permit conditions that the staff recommends the Commission include if the CP is issued, which are intended in part to ensure the staff has access to information on certain areas where the design and analysis is expected to mature between the CP and OL stages.

Based on these findings as documented in this SE, and subject to the permit conditions identified in appendix A of this SE, the staff recommends that the Commission make the following conclusions for the issuance of a CP for the KU1 facility in accordance with 10 CFR 50.35, 10 CFR 50.40, and 10 CFR 50.50:

1. USO has described the proposed design of KU1, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.
2. Such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the FSAR.
3. Safety features or components that require R&D have been described by USO and an R&D program will be conducted that is reasonably designed to resolve any safety questions associated with such features or components.
4. On the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be

constructed and operated at the proposed location without undue risk to the health and safety of the public.

5. There is reasonable assurance: (i) that the construction of KU1 will not endanger the health and safety of the public, and (ii) that construction activities can be conducted in compliance with the Commission's regulations.
6. USO is technically qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
7. USO is financially qualified to engage in the construction of its proposed facility in accordance with the Commission's regulations.
8. The issuance of a permit for the construction of KU1 would not be inimical to the common defense and security or to the health and safety of the public.
9. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering reasonable available alternatives, the issuance of the CP, subject to the conditions for protection of the environment set forth therein, is in accordance with Subpart A, "National Environmental Policy Act—Regulations Implementing Section 102(2)," of 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
10. The application meets the standards and requirements of the Atomic Energy Act and the Commission's regulations, and notifications to other agencies or bodies have been duly made.

1.3 General Description

1.3.1 Introduction

PSAR chapter 1 provides an overview of the facility, including a brief description of the proposed plant location, the type of reactor being proposed, the intended use of the reactor, and a summary description of the overall plant configuration. PSAR section 1.1.4 includes a summary description of plant SSCs, including reactor systems and components, secondary systems and components, support systems and components, and the major structures of the nuclear island (NI).

Safety-significant SSCs (i.e., those that are assigned a safety classification of safety-related (SR) or non-safety-related with special treatment (NSRST)) are described in further detail in PSAR chapter 7 and evaluated in chapter 7 of this SE. The seismic monitoring system is evaluated in section 6.4.1.1 of this SE. Liquid and solid radwaste processing (RWL and RWS) are described in more detail in PSAR chapter 9 and evaluated in chapter 9 of this SE. However, many non-safety related with no special treatment (NST) SSCs are primarily or only described in PSAR section 1.1.4, including:

- secondary systems and components
 - salt systems

- power cycle systems
- main power system
- support systems and components
 - NI ancillary electrical system
 - NI plant communication system
 - NI cranes and hoists
 - energy island (EI) auxiliary electrical system
 - NI air and inert gas distribution systems
 - NI major maintenance equipment
 - NI emergency operating lighting
- support buildings
 - NI electrical equipment modules (e-modules)
 - EI facilities

These systems and their classification as NST are evaluated in this section of the SE.

The discussion on secondary systems and components includes a reference to NATD-LIC-RPRT-0001-A, “Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces” (ML24011A321). Specifically, as stated in section 1.1.4.4.9 of the PSAR, “The EI facilities support thermal energy storage and steam generation plant operations that are independent from reactor power operations due to the TSS [thermal salt storage system] as described in NATD-LIC-RPRT-0001.” This NRC-approved TR describes an evaluation of NRC regulations pertinent to the design interface of Natrium NI and EI systems. As confirmed through audit, although the applicant mentioned this TR in its application, the applicant is not attempting to rely on any of the analysis within. Further, the staff’s analysis of the application does not rely on the information, analysis, or conclusions in the TR or the staff’s SE approving the TR. The staff additionally notes that the NRC has issued an exemption to USO excluding certain NST EI SSCs from the definition of construction (ML25119A331). In its SE approving this exemption, the staff stated that it had “reviewed the design described in the exemption and the proposed PSAR and found it to be consistent with the design features supporting the NI-EI independence discussed in NATD-LIC-RPRT-0001-A and the associated NRC staff SE.” The staff has reviewed the current version of the PSAR and confirmed that that statement is still true.

PSAR section 1.2 provides a site description overview. Section 1.3 of the PSAR summarizes approaches described in NEI 18-04 as applied to KU1, for the selection of LBEs, safety classification of SSCs, and determination of DID adequacy. Section 1.3 of the PSAR also provides an overview of how the design addresses the fundamental safety functions (FSFs) of retaining radionuclides, controlling heat generation, and controlling heat removal.

PSAR section 1.4 includes tables that provide:

- a discussion of the facility's conformance with regulatory guides;
- a listing of the TRs and technical reports that are incorporated by reference into the PSAR;
- a discussion of the facility's conformance with generic safety issues (GSIs), unresolved safety issues (USIs), and Three Mile Island (TMI) action items;
- the consensus codes and standards used in the design.

1.3.2 Regulatory Evaluation

Regulations applicable to the information included in PSAR chapter 1 include 10 CFR 50.34. In particular, 10 CFR 50.34(a)(1)(ii) requires a description and safety assessment of the site and a safety assessment of the facility, and 10 CFR 50.34(a)(2) requires a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

Section 1 of DANU-ISG-2022-01 provides guidance for, in part, the review of chapter 1 of an LMP-based PSAR. This section references RG 1.253 which endorses, with clarifications and additions, NEI 21-07.

NEI 21-07 indicates that PSAR section 1.1 content, which provides the summary description of the plant and its SSCs, should reflect the preliminary nature of the design, as appropriate, and should be sufficient to "permit the reader to understand fundamental concepts of the plant and how it operates," "support reader understanding of the design and how the LMP-based affirmative safety case will be developed," and "understand the initial plant functionality." The guidance for PSAR section 1.2 states that the section should provide a high-level overview of the site and general vicinity of the licensed activities. The guidance for PSAR section 1.3 states that the section should provide "a high-level overview of the LMP-based affirmative safety case methodology," focused on the FSFs and the DID aspects of the design.

DANU-ISG-2022-01 section 1.1.5 provides additional guidance including that the following summary tables should be included in chapter 1 of a PSAR:

- The GSIs, USIs, and TMI action items technically relevant to the design. Section 1.1.5 of DANU-ISG-2022-01 notes that appendix B of the ISG provides useful information on the applicability of NRC regulations to non-light-water power reactors that should be considered when reviewing such tables.
 - DANU-ISG-2022-01 appendix B notes that while 10 CFR 50.34(f), "TMI-related requirements," does not apply to the design, the Commission direction in SECY-15-0002 (ML15266A023) confirmed that early directions for 10 CFR Part 52 new power reactor applications should be applied consistent to 10 CFR Part 50 new power reactor applications. Based on this, DANU-ISG-2022-01 appendix B states that the staff should ensure that an applicant addresses the technically relevant

TMI-related items during the review process and propose license conditions requiring the appropriate item in the interim.

- RGs directly applicable to the design, and whether the applicant proposes an alternative approach to satisfy a regulation rather than following the guidance in one of the RGs.
- The consensus codes and standards used in the design, and whether the applicant proposes to request an exemption from or alternative to such standards that are incorporated by reference into 10 CFR 50.55a.

1.3.3 Technical Evaluation

1.3.3.1 Description of Plant SSCs

The summary design information provided in PSAR section 1.1.4 addresses the summary description required by 10 CFR 50.34(a)(2). For safety-significant SSCs, more detailed information and the safety assessment required by 10 CFR 50.34(a)(1)(ii) is provided elsewhere in the PSAR, but for NST SSCs this assessment is provided in PSAR section 1.1.4 and evaluated below.

1.3.3.1.1 Secondary Systems and Components

Section 1.1.4.2 of the PSAR describes the secondary systems and components. The KU1 design includes a separate NI and EI. Secondary systems and components are contained on the EI, while the NI contains the reactor and associated sodium and fuel handling systems. The interface between the NI and EI is shown in PSAR figure 1.1-1. PSAR section 1.1.4.2 states that the NI boundary conditions have been designed so the interface with the EI does not impact the KU1 safety analysis, and all EI systems and functions are classified as NST. The summary description provided in PSAR section 1.1 is discussed below; more detail on the interfaces between the EI and NI (particularly through the molten salt and electrical systems) are discussed in SE chapter 7.

Salt and Power Cycle Systems

Section 1.1.4.2.1 of the PSAR describes the salt systems, which consist of the nuclear island salt system (NSS), energy island salt system (ESS), and the TSS. The section also describes the role of the salt systems in transferring energy from the intermediate heat transfer system (IHT) to the TSS salt storage tanks and beyond to the steam generation system (SGS); how the NSS isolation valves provide the boundary between the NSS and ESS (and thus the NI and EI); and how the salt system is monitored from the main control room during normal operations using the NSS molten salt flow rate and hot and cold salt temperatures and pressures. The sodium-salt heat exchanger (SHX) provides the interface between the salt-based NSS and the sodium-based IHT; IHT and SHX are described in more detail in section 7.1.4 of this SE.

Section 1.1.4.2.2 of the PSAR describes the power cycle systems, which consist of the SGS, condensate and feedwater system (CFW), steam turbine system, generator system (GEN), and heat rejection system. These systems are used to generate electricity and reject waste heat to the atmosphere via cooling towers.

As described in chapter 3 of the PSAR, the salt systems and the downstream power cycle systems are not relied on for decay heat removal following a plant transient or accident. Transients initiated by a failure in these systems are evaluated in PSAR chapter 3 as “energy island transients” and result in either a reactor runback (an automatic, controlled reduction in power and flow) or scram. DBEs involving EI systems are described in PSAR sections 3.7.2.3 and 3.7.2.4, a beyond design basis event is described in PSAR section 3.8.2.3, and a design basis accident (DBA) is described in PSAR section 3.9.2.2. For all transients, the reactor remains in a safe condition with no fuel failure and with heat removal provided by NI SSCs. As discussed in PSAR section 1.1.4.2 and confirmed by the staff’s review of chapter 5, the preliminary classification of all EI systems is NST. Based on the description provided in PSAR chapter 1 and the disposition of EI transients in PSAR chapter 3, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the salt and power cycle systems and the role of these systems in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR information regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

Main Power System

Section 1.1.4.2.3 of the PSAR describes the main power system. This system includes a normal main power subsystem that supplies power from the generator to the 230 kilovolt switchyard and plant auxiliary loads through the unit auxiliary transformer, and an alternate main power supply subsystem that transfers power from the switchyard to plant auxiliary loads through the reserve auxiliary transformer when the main power subsystem is unavailable. Power is supplied to plant loads through these transformers from the switchyard during startup and shutdown. The unit auxiliary transformer and reserve auxiliary transformer supply the NI and EI auxiliary electrical systems, which are described in more detail in PSAR sections 1.1.4.3.6 and 1.1.4.3.14, respectively, and will be discussed in the following section of this SE. Safety-significant power systems are discussed in more detail in PSAR section 7.7.

While electric power is needed for some NSRST functions, as described in PSAR section 7.7, these loads are supplied via the NI uninterruptible alternating current (AC) power supply system (NUP) and NI direct current (DC) power supply system (NDC) for up to 72 hours using batteries. These systems are evaluated in section 7.7 of this SE. The 72 hour mission time is consistent with the station blackout coping period specified in the Standard Review Plan (SRP) section 8.4, “Station Blackout” (ML100740424) and SRP 19.4, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors” (ML13081A756). While aspects of this guidance are LWR-specific, the concepts surrounding the 72 hour coping time for loss of offsite power are not and in the staff’s judgment this coping time is appropriate to apply to KU1. Based on the adequacy of the NSRST systems to support the 72 hour mission time, the staff determined that it is reasonable for the main power system to be classified as NST. Considering the summary-level information provided in PSAR chapter 1 and the more detailed information provided in PSAR chapter 7, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the main power system and the role of these systems in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR information regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

1.3.3.1.2 Support Systems and Components

NI Ancillary Electrical System, Plant Communication System, and Emergency Operating Lighting System

Section 1.1.4.3.11 of the PSAR describes the NI ancillary electrical system, which includes NI lighting systems (NLS) and NI grounding and earthing and lighting protection (NGL). Power for NLS is provided by the NI AC electrical power low voltage system and backed up for certain key lighting systems by self-contained batteries, diesel generators, or NUP. NGL protects facility staff and equipment from transient over-voltages, gives a ground reference for instrumentation signals, and provides protection for lightning strikes and switching surges.

Section 1.1.4.3.12 of the PSAR describes the NI plant communication system, which is used to provide internal and external communications during normal and emergency plant operations. The PSAR states that it consists of diverse and independent communication subsystems that will be designed, fabricated, erected, constructed, and tested in accordance with industry standards. Emergency communication capabilities are described in additional detail in PSAR section 11.3, which notes that additional information will be provided in the OL application.

Section 1.1.4.3.17 of the PSAR describes the NI emergency operating lighting system. Emergency lights are used to support fire suppression and recovery actions in the NI control building (NCB), in and between the main control room (MCR), and remote shutdown complex (RSC). The lights are DC self-contained battery-operated units, and when in proximity to safety-significant components are appropriately designed to mitigate impacts due to seismic events; this is consistent with the seismic interaction design requirements described in PSAR section 6.4.1.5 and evaluated by the staff in section 6.4.1.1.5 of this SE.

The staff did not identify any SR functions that rely on the NI ancillary electrical, NI plant communication, or NI emergency operating lighting systems, though the staff noted that lighting may be needed for manual actions that are NSRST for DID adequacy. The applicant stated that lighting is evaluated as part of the human factors engineering (HFE) program, as discussed in PSAR section 11.2. The staff evaluated the HFE program in section 11.2 of this SE but noted that a complete evaluation of the adequacy of HFE within the KU1 design can reasonably be left to the OL application. The NSRST manual actions include manual scram, manual primary sodium pump trip, manual intermediate sodium pump trip, and manual sodium processing system pump trip, as described in PSAR chapter 5. These functions are performed in the MCR and RSC, and communications systems are not needed to perform these functions. The MCR and RSC are discussed in PSAR section 7.6.7. Based on the summary-level information provided in PSAR chapter 1, as augmented by information on safety classifications, manual actions, and emergency communications needs as discussed in PSAR chapters 5, 7, and 11, the staff determined that the preliminary information was sufficient to understand the fundamental design and operation of the lighting and communications systems and the role of these systems in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR information regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

NI Cranes and Hoists

Section 1.1.4.3.13 of the PSAR describes the NI cranes and hoists (NCH) system, which consists of the bridge cranes and monorail hoists in the reactor building (RXB), fuel handling

building (FHB), and reactor auxiliary building (RAB). The RXB and FHB cranes lift and move critical heavy loads. These cranes are designed to be single-failure proof and to the standard of ASME NOG-1 as Type I cranes. Other monorail hoists in FHB and RAB do not carry critical loads and are not single failure proof. Cranes used to handle fuel are not part of NCH system and are described in additional detail in PSAR section 7.3. Loads from NCH are accounted for in the building structural designs as described in PSAR sections 6.4 and 7.8. Based on the summary-level information provided in PSAR chapter 1, as augmented by more detailed information on how loads are accounted for in structural design in PSAR chapters 6 and 7, the staff determined that the preliminary information was sufficient to understand the fundamental design and operation of the NCH systems. Because of this, this aspect of the information provided is consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01 and as such the staff determined it is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2). The role of NCH cranes in preventing the occurrence of postulated initiating events (PIEs) is not evaluated in the PSAR as part of the LMP-based safety analysis. The applicant indicated in a supplement to the CP application (ML25253A386) that it will be assessed using a new process for evaluating preventative measures, and would be fully addressed in the KU1 OL application, should the applicant apply for an OL. The staff evaluation of this process is in SE section 5.1. Given the RXB and FHB bridge cranes are being designed to the ASME NOG-1 standard as a single-failure proof (Type I) crane, which will make them highly reliable, the staff determined the design information was adequate for a CP application and thus acceptable. The staff expects to review the design and safety classification of NCH in more detail during the OL application review.

El Auxiliary Electrical System

PSAR section 1.1.4.3.14 describes the EI auxiliary electrical system, which consists of medium voltage and low voltage systems serving the EI. The EI auxiliary electrical system also includes the standby diesel generators (SDGs), which are available to provide backup power to the NI AC electrical power medium voltage system for asset protection purposes.

Like other EI systems, the EI auxiliary electrical system is NST. Despite this, there are some analyzed LBEs in PSAR chapter 3 where power from the SDGs is modeled (e.g., the loss of offsite power with non-passive intermediate air cooling (IAC) described in PSAR section 3.6.1.2). However, for each of these LBEs there are also similar LBEs without power from the SDGs that demonstrate the fuel acceptance criteria are satisfied (e.g., the loss of offsite power with passive IAC described in PSAR section 3.7.1.2); this shows that the SDGs are not needed to maintain events within the frequency-consequence target curve and confirms that they are acceptable to classify as NST.

Based on the above, and the discussion provided in chapters 1, 3, and 7 of the PSAR relative to the EI auxiliary electrical system, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the EI auxiliary electrical system and the role of these SSCs (in particular, the SDGs) in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR information regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

NI Major Maintenance Equipment

PSAR section 1.1.4.3.19 describes the NI major maintenance equipment (NME), which consists of a cask, a transfer adapter with closure valve, an inert gas management system, and a component lift system with a grapple. The NME provides a temporary inert gas environment to prevent excess air ingress into the primary cover gas and leakage of primary cover gas to the RXB during installation and removal of in-vessel fuel handling equipment. The installation and removal of the in-vessel fuel handling equipment occurs during operating mode 4, which specifies that the reactor is shutdown with no more than one control rod assembly coupled to its drive (i.e., the reactor is subcritical) and a sodium temperature of less than 400° F. The in-vessel equipment used to handle fuel is not part of NME and is described in additional detail in PSAR section 7.3.3. PSAR section 1.1.4.3.19 also states that details regarding refueling operations, including PRA evaluation of the installation and removal of the in-vessel fuel handling equipment, would be developed in support of the OL review. Based on the information provided in PSAR chapter 1, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the NI major maintenance systems, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such the staff determined it is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2). With respect to the LMP-based safety-analysis, the NME performs no identified role in preventing or mitigating any LBE. However, it maintains an inert gas environment connected to the reactor head component nozzles while they are open for installation of the in-vessel equipment. Design features of the in-vessel fuel handling equipment, as described in PSAR section 7.3.3, provide assurance that an NME handling event that causes the in-vessel equipment to drop during installation does not result in damage to fuel. The staff expects to review the design and safety classification of the NME, as well as NME operations during refueling, in more detail during the OL application review.

1.3.3.1.3 Support Buildings

Support buildings described in PSAR chapter 1 consist of the NI e-modules and EI facilities. The e-modules contain electrical systems and components. As described in PSAR section 7.7, safety-significant NI electrical components are housed in the NCB substructure and are thus not contained within e-modules. EI facilities include all the buildings on the EI, which as previously discussed, are classified as NST along with the rest of the EI SSCs. Based on the information provided in PSAR chapter 1, as augmented by the discussion in chapter 7 clarifying the role of the e-modules, the staff determined that the preliminary information provided was sufficient to understand the fundamental design and operation of the support buildings and their role in the LMP-based safety analysis, consistent with the guidance in RG 1.253, NEI 21-07, and DANU-ISG-2022-01. As such, the staff determined that the PSAR information regarding these systems is acceptable to support 10 CFR 50.34(a)(1)(ii) and (a)(2).

1.3.3.2 Site Description

The staff reviewed site description information in section 1.2 of the PSAR and site information provided in PSAR chapter 2. The staff review is documented in SE chapter 2.

1.3.3.3 Plant Safety Overview

The staff reviewed information in section 1.3 of the PSAR, including how FSFs are addressed. The staff review is documented in SE chapters 3, 4, and 5.

1.3.3.3.1 Licensing Basis Methodology

As described in PSAR section 1.3.1, the KU1 licensing basis was developed using the process described in NEI 18-04. This process is referred to interchangeably throughout this SE as the NEI 18-04 process, the LMP process, or the NEI 18-04 or LMP methodology. As described earlier, NEI 18-04 was found to be acceptable, with certain clarifications, in RG 1.233. Guidance for the content of applications using this process is provided in NEI 21-07, which was found to be acceptable with certain clarifications and additions in RG 1.253, as well as DANU-ISG-2022-01 and references therein.

PSAR section 1.3.1 states that conformance with the NEI 18-04 process is demonstrated in PSAR chapters 3 through 8. The staff's detailed evaluation of the applicant's implementation of the NEI 18-04 process is documented throughout the corresponding chapters of this SE. Because this application is the first time LMP is used for a commercial power reactor, an orientation to the NEI 18-04 process and how it is reflected in the structure of the PSAR is provided below.

From a safety analysis perspective, the LMP process has the overall process flow shown below. First, an applicant develops an initial list of PIEs and a PRA. Development of the PRA necessitates supporting information, including elements such as initiating event identification, event sequence analysis, and human reliability analysis. The PRA is used to refine and further develop the initial set of LBEs, which are then categorized based on their event sequence frequency. The LBEs are then analyzed for their radiological consequences, which involves analyzing system transients (with nuclear and thermal hydraulic analyses), source term generation, and atmospheric transport (i.e., consequence analysis). LBE frequencies and consequences are evaluated against a frequency-consequence (F-C) target curve, which is also used to determine which events are risk significant based on their location relative to the target.



Figure 1.3-1: General flow of NEI 18-04 process

Once events have been selected, an applicant would evaluate the safety functions modeled in the PRA (also referred to as PRA safety functions, or PSFs) involved in those events to determine if they are safety-significant by assessing the role they play in maintaining LBEs below the F-C target curve, preventing and mitigating DBAs, and their contributions to integrated risk metrics and meeting DID adequacy. DID adequacy is evaluated in an integrated process that ensures multiple functions are available to perform each of the generically applicable FSFs described in NEI 18-04. PSFs that are determined to be safety-significant because they are needed to maintain LBEs below the F-C target curve or because of their contribution to integrated risk metrics are also designated risk-significant. The SSCs are then assigned a safety classification and risk significance based on the function(s) they perform. Design requirements for each SSC are then determined based, in part, on the safety classification and what is needed to ensure the SSC can appropriately perform its safety

function(s). The same process is used to identify programmatic special treatments applicable to each SSC.

This process flow is reflected in the structure of the PSAR for the KU1 CP application. In the PSAR, the PRA is described in section 3.1, source terms in section 3.2, and the methodologies for consequence and other supporting analyses in section 3.3. The overall set of selected LBEs and descriptions of each LBE, including frequencies and radiological consequences, consistent with the NEI 21-07 guidance, are provided in PSAR sections 3.4 through 3.9. Integrated risk metrics and DID evaluation are discussed in PSAR chapter 4. The methodology for identifying safety-significant PSFs is in section 5.1 of the PSAR, with results in section 5.2 of the PSAR. In the SE, the staff discusses the functions and their safety classifications in sections 5.2, 5.4, and 5.5 to align more with the NEI 21-07 guidance. The design requirements and methodology that are applied based on those safety classifications and functions are provided in PSAR chapter 6, and descriptions of the design and how it meets those requirements to perform the safety-significant PSF are in PSAR chapter 7. Finally, the programs discussed in PSAR chapter 8 serve as programmatic special treatments for the SSCs described in PSAR chapter 7.

The staff identified that the content of the PSAR differs slightly from NEI 21-07. PSAR chapter 2 includes site information, as discussed in DANU-ISG-2022-01. The supporting analyses discussed in NEI 21-07 chapter 2, like fuel design, nuclear and thermal-hydraulic design, and criticality safety analyses, are described in PSAR chapter 3. Where NEI 21-07 chapter 6 includes the design requirements and descriptions of SR SSCs and chapter 7 includes design requirements and descriptions of NSRST SSCs, PSAR chapter 6 provides the design requirements for SR and NSRST SSCs and PSAR chapter 7 provides a description of all SR and NSRST SSCs. The staff determined that this reorganization of the information in the PSAR relative to the guidance does not affect the intended scope of the information.

1.3.3.3.2 Fundamental Safety Functions

As described in PSAR section 1.3.2, the FSFs for KU1 include control of heat generation, control of heat removal, and radionuclide retention. The staff determined these FSFs are consistent with those described in NEI 18-04 section 3.3.4 which are intended to be generically applicable to all reactors.

PSAR section 1.3.2.2 explains how control of heat generation is established using two different banks of control rods, which can be inserted using both active (power runback, which inserts control rods using the control rod drive system) and passive (gravity scram) means. DID for the gravity scram is provided by a control rod drive (CRD) driveline scram follow feature, in which the protection system commands the CRD to insert control rods into the core coincident with a scram signal. Though not described in PSAR chapter 1, the alternative shunt trip discussed briefly in PSAR chapters 3 and 5 provides a DID pathway for tripping the reactor trip breakers and initiating a gravity scram. In addition to the active and passive means of controlling reactivity, there is inherent reactivity feedback provided by the reactor core. Design features related to control of heat generation are described in additional detail in chapter 7 of the PSAR, with additional discussion on their performance in transient analyses in chapter 3.

PSAR section 1.3.2.3 explains how the IAC and reactor air cooling system (RAC) satisfy the FSF of controlling heat removal. The PSAR states that IAC provides active and passive means of controlling heat removal by pulling heat from the reactor core using the intermediate heat exchangers into the IHT and discharging them to the atmosphere through sodium-air heat

exchangers. The active means of IAC relies on the intermediate sodium pumps and IAC blower, while the passive means relies on natural circulation. The RAC provides heat removal by directing air from outside the reactor building past the reactor and out through an outlet stack. Motive force for the air is provided by natural circulation. The PSAR states that the RAC is an inherent means of controlling heat removal; however, the staff considers it to be passive because it relies on maintaining an open flow path. Both passive IAC and RAC cooling rely on natural circulation of sodium in the reactor vessel to draw heat from the core. IAC and RAC and supporting features in the IHT and primary heat transport system are described in detail in chapter 7 of the PSAR, with additional discussion on their performance in transient analyses in chapter 3.

PSAR section 1.3.2.1 describes the applicant's approach to ensuring radionuclide retention, which is achieved using a functional containment strategy. A high-level definition of functional containment is provided in SECY-18-0096, which was approved by the Commission, as "a barrier, or set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment." Under this definition, compared to the containments of the operating light-water reactor fleet, a functional containment is not necessarily comprised of a single pressure-retaining containment structure. Instead, in the approach described in SECY-18-0096, functional containment performance criteria are developed to address each barrier's role in mitigating releases to meet plant-level performance criteria (i.e., regulatory dose requirements). Enclosure 2 to SECY-18-0096 describes a proposed risk-informed, performance-based, technology-inclusive approach to derive these functional containment performance criteria. The staff notes that, while functional containment is commonly associated with the development of the Next Generation Nuclear Plant (NGNP), a gas-cooled reactor, and the use of tristructural isotropic (TRISO) fuel, the methodology described in SECY-18-0096 and Enclosure 2 that was approved by the Commission is applicable to any non-LWR technology.

Under the NEI 18-04 process, mechanistic source term and radiological consequence analyses are used to determine offsite doses for the LBEs and thus evaluate the adequacy of functional containment barriers. The applicant determines the adequacy of the functional containment barriers based on doses meeting the F-C target curve provided in NEI 18-04. As stated in SECY-18-0096 enclosure 2:

[P]erformance criteria for the design features associated with retaining radioactive materials within a facility will be established based on the range of event categories and the related success criteria for each category... Success criteria for AOOs and DBEs include a graded scale for potential offsite doses based on event sequence frequencies (i.e., below an F/C target) and demonstration that prevention features such as cooling systems and fuel system boundaries limit the migration of fission products within the facility.

These considerations are included in the NEI 18-04 process, which provides a process for categorizing accidents based on their frequencies, conservative treatment of DBAs, and an F-C target curve, consistent with table 1 of enclosure 2 to SECY-18-0096. NEI 18-04 also includes the DID adequacy evaluation discussed in SECY-18-0096 enclosure 2. Based on the above, the staff determined that the NEI 18-04 methodology is consistent with the approach described in the enclosure to SECY-18-0096 and provides the necessary information to develop the functional containment performance criteria.

In NEI 18-04, the functional containment performance criteria are then used to inform reliability and capability targets, which are intended to ensure successful completion of PRA safety functions, including radionuclide retention. This overall process is reflected in the PSAR and the staff's SE. The LBEs that result in releases for which functional containment performance must be analyzed are described in PSAR sections 3.5 through 3.9, and evaluated by the staff in sections 3.4 through 3.8 of this SE. PSAR section 3.2 describes the mechanistic source term methodology and the source terms analyzed for the KU1 CP application. The staff evaluates source term methodology and analyses in section 3.2 of this SE. PSAR sections 3.3.1.4 and 3.3.2.2 describe the methodology to calculate radiological consequences for the non-DBA LBEs and DBAs, respectively. The staff evaluates radiological release consequences methodology and analyses in section 3.3.1.5 of this SE. In the CP application analyses, functional containment barrier performance is based on assumptions, which feed into the capability targets; the design will be fully evaluated by the staff at the OL. PSAR section 6.2 describes the applicant's assessment of reliability and capability targets for safety-significant SSCs, which is evaluated by the staff in section 6.2 of this SE.

Section 1.3.2.1 provides a list of SSCs associated with the functional containment strategy. The staff reviewed this list and determined it was consistent with the identification of radionuclide retention functions for SSCs as reflected through the safety analysis, safety classification, and plant design information provided in chapters 3, 5, 6, and 7 of the PSAR.

With respect to functional containment, the staff concluded that the overall approach is acceptable to apply to the KU1 design based on the use of the NEI 18-04 methodology for licensing basis development, which is technology-inclusive and consistent with the process described in SECY-18-0096 Enclosure 2, and the reasonable identification of those barriers responsible for radionuclide retention. However, the staff did not come to a final determination on the adequacy and acceptability of functional containment performance due to the preliminary nature of the design and analysis as discussed in chapters 3, 4, 5, 6, and 7 of this SE. The staff will confirm the acceptability of the functional containment performance criteria and the associated design and performance evaluation at the OL.

1.3.3.4 Conformance with Regulatory Criteria and Referenced Material

Table 1.4-1 of the PSAR provides a list of applicable RGs. The table includes a discussion of whether the design is in full conformance or partial conformance with the RG. As appropriate, the RG conformance is evaluated in the applicable sections of this SE. The staff noted that table 1.4-1 also provides a listing of RGs which are not considered by USO to be applicable to KU1 at the CP phase and will be addressed as part of the OL application.

Applicable TRs and technical reports are identified in table 1.4-2 and table 1.4-3 of the PSAR, respectively. Table 1.8-1 of this SE provides a listing of the TRs that have been incorporated by reference into the PSAR and where in this SE the staff evaluated the applicability of the TRs and any associated limitations and conditions (L&Cs) associated with these reports. Table 1.8-2 of this SE provides a listing of the technical reports that are incorporated by reference into the PSAR and submitted as part of the CP application and where in this SE report these technical reports are evaluated.

PSAR table 1.4-4 provides a listing of TMI-related items that USO identified as either fully or partially applicable to the KU1 design. DANU-ISG-2022-01, appendix B, table 4 provides generic applicability determinations of the TMI items for non-LWRs, with entry conditions for

technical relevancy listed for some items. The discussion associated with table 4 states the TMI items are only requirements applicable to 10 CFR Part 52 applications, but as discussed in section 1.3.2 of this SE, DANU-ISG-2022-01, appendix B also states that the staff should ensure that an applicant addresses the technically relevant TMI-related items during the review process and propose license conditions requiring the appropriate items in the interim. The staff compared the TMI related items in table 1.4-4 of the PSAR against the information found in DANU-ISG-2022-01 table 4. Table 1.8-3 of this SE provides a listing of the TMI items that are either fully or partially applicable, where in this SE these requirements are evaluated, and the staff's disposition of the applicability of the items. Because USO committed to addressing the technically relevant TMI-related items in PSAR section 1.4.3 and table 1.4-4, the staff determined it was not necessary to propose a license condition as discussed in DANU-ISG-2022-01.

Regarding GSIs and USIs, the applicant determined in PSAR section 1.4.3 that none of the non-TMI items provided in appendix B of NUREG-0933, Resolution of Generic Safety Issues (formerly titled "A Prioritization of Generic Safety Issues") applied to the KU1 facility. The staff notes that there is no requirement in 10 CFR Part 50, similar to the 10 CFR Part 52 requirements, to address USIs and GSIs in NUREG-0933. Although not a requirement, guidance in section 1.1.5 of DANU-ISG-2022-01 suggests that the technically relevant USIs and medium and high priority GSIs be evaluated for non-LWR applicants. The staff performed an independent review of the USIs and GSIs to determine if any of these items are technically relevant to the KU1 facility. The staff identified several items with technical relevance to the KU1 design but concluded that they are either appropriately addressed through the design process (including by conformance to applicable guidance or consensus codes and standards) as described in various sections of this SE or can be assessed at the OL stage of the review. Examples include Task Action Plan Item A-25, which concerns non-safety related loads on Class 1E power sources and is addressed through conformance with RG 1.75, and GSI-186, which concerns heavy load drops and is addressed through design of cranes to the ASME NOG-1 standard and implementation of a heavy load program in conformance with RG 1.244 (which will be implemented at the OL stage). As such, the staff determined that the USIs and GSIs are appropriately addressed for the CP application.

Table 1.4-5 of the PSAR provides a listing of the codes and standards used in the design. Providing a listing of the codes and standards used in the design is consistent with the guidance found in DANU-ISG-2022-01 section 1.1.5. As applicable, these codes and standards are evaluated as part of the SSC evaluations found in this SE.

Evaluation of Topical Reports

TRs incorporated by reference are listed in table 1.4-2 of the PSAR. The applicability of the TRs and any associated L&Cs are evaluated in the portion of the SE that corresponds to where they are primarily referenced in the PSAR. Table 1.8-1 of this SE details where each TR is evaluated by the staff.

1.3.4 Conclusion

The staff reviewed the information on the plant SSCs provided in PSAR section 1.1.4, the plant safety overview information provided in PSAR section 1.3, the conformance with regulatory criteria and reference material provided in PSAR section 1.4, and other information from elsewhere in the PSAR as pertinent, as discussed above, and determined it is acceptable to

support 10 CFR 50.34(a)(1)(ii) and (a)(2) because it is consistent with the applicable guidance on content of applications from RG 1.253, NEI 21-07, and DANU-ISG-2022-01 and the NEI 18-04 process. Several aspects of the information provided in chapter 1, including the site description provided in PSAR sections 1.2 are evaluated in more detail in other portions of the staff's SE as described above.

1.4 Shared Facilities and Equipment

The staff determined that this section is not applicable because KU1 is a single-unit site, as stated explicitly in PSAR section 5.3.1.5.

1.5 Comparison with Similar Facilities

The applicant did not provide a comparison with similar facilities, because it is not requested by RG 1.253 or NEI 21-07. However, the staff includes a brief discussion of similar facilities and relevant licensing history here. KU1 is a pool-type sodium-cooled fast reactor (SFR) using metallic uranium-zirconium alloy fuel. This configuration has similarities to several different reactors that have operated in the US and internationally. Information on prior experience in operating and licensing similar facilities is available in NUREG/KM-0007, "NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors" (ML14128A346). While this discussion focuses on domestic operating and design experience since many international reactors differ substantially in the fuel or primary coolant system design from the KU1 design, further information on international reactor designs is provided in NUREG/KM-0007. The international operating experience, particularly regarding sodium fires and design of intermediate and secondary systems, was considered by the staff during the review.

The US operating experience with liquid-metal cooled reactors began with the construction of the Experimental Breeder Reactor I (EBR-I) at Argonne West (now Idaho National Laboratory) in Idaho, a test reactor which operated from 1951–1963. Subsequent test reactors include the Sodium Reactor Experiment (SRE), which operated from 1957–1964 at the Santa Susana Field Laboratory in California; the Experimental Breeder Reactor II (EBR-II), which operated from 1964–1994 at Argonne West (now Idaho National Laboratory, or INL) in Idaho; the Southwest Experimental Fast Oxide Reactor (SEFOR), which operated from 1969–1972 in northwest Arkansas; and the Fast Flux Test Facility (FFTF), which operated from 1980–1993 at Pacific Northwest Laboratory (PNL, now Pacific Northwest National Laboratory) in Washington. The FFTF design was reviewed by the NRC and an SE on the FSAR was issued in 1978 as NUREG-0358. One commercial reactor, Enrico Fermi Nuclear Generating Station, Unit 1 (Fermi 1), operated from 1963–1966 in Michigan.

Other SFRs were proposed in the 1970s through the present, and several of them went through various stages of the licensing process with the NRC; most notably, the Clinch River Breeder Reactor Project (CRBRP) and General Electric's PRISM reactor. CRBRP, a loop-type, oxide-fueled SFR similar to FFTF, applied for a CP and submitted a PSAR to the NRC in 1975. The NRC issued an SE report for CRBRP in 1983 as NUREG-0968 (ML082381008), but due to project cancellation a CP was never issued. Regulatory engagements regarding PRISM, a pool-type, metal-fueled SFR, began in 1986 with the submittal of a preliminary safety information document. The NRC issued a pre-application safety analysis report for PRISM in 1994 as NUREG-1368 (ML063410561).

Of these reactors, all used sodium coolant except EBR-I, which used a sodium-potassium eutectic (NaK). EBR-I, SRE, EBR-II, and Fermi 1 all primarily used metallic fuel like KU1, though EBR-II was also used to test a wide variety of different fuels. SEFOR and FFTF primarily used oxide fuel, though FFTF also later tested a number of metal fuel pins. Of the reactors that operated in the US, the KU1 overall system design is the most similar to EBR-II because it was a pool-type reactor, though there are substantial differences in size and the detailed design between the two plants, including EBR-II's use of a z-pipe to connect the core outlet to the hot pool rather than the open configuration used in KU1. The size and core configuration of FFTF was also somewhat similar to KU1, in that it had similarly sized fuel assemblies and used a similar core restraint system to KU1, but it had coolant loops that circulated primary coolant outside of the reactor vessel. As such, though most key features of KU1 have been demonstrated through operating experience, there is no single facility that fully captures all of them.

The staff's review was, in particular, informed by SFR operating experience where fuel was damaged. At EBR-I, fuel damage occurred during a coolant flow test due to thermal expansion of the core components that had not been accounted for by design. At SRE, an organic lubricant infiltrated the primary system and reacted with the sodium, forming blockages that led to fuel damage. At Fermi 1, fuel damage was caused by a flow blockage when a metal plate detached from the bottom of the reactor and covered a fuel assembly inlet. This operating experience has informed the subsequent licensing history of SFRs, which has typically required evaluation of flow blockages.

Of the conceptual designs that were not operated, the PRISM design is most similar to KU1. Both are pool-type, metal-fueled SFRs, with seismic isolation systems and SR decay heat removal provided by an air-based cooling system (called RAC for KU1 and reactor vessel auxiliary cooling system (RVACS) for PRISM). However, there are still substantial differences between the facilities. KU1 uses high-assay low enriched uranium (HALEU) fuel, while PRISM was proposed to use uranium-plutonium-zirconium (U-Pu-Zr) alloy fuel with depleted uranium blankets. KU1 has a molten salt energy storage system separating the sodium from the water-based power conversion system, where PRISM ran steam generators directly off the intermediate loop. KU1 has mechanical primary coolant pumps, while PRISM had electromagnetic pumps.

These differences in key design features are reflected as differences in the overall safety strategy for the two plants. PRISM's neutronic and thermal-hydraulic design characteristics led to consideration of accident scenarios involving coolant boiling leading to significant core melt and the potential for a hypothetical core disruptive accident. This accident involves the relocation of the fuel into a more reactive configuration and subsequent energetic disassembly of the core and has historically been used to define the forces that must be withstood by the reactor vessel, reactor head, and containment. To address these concerns, the PRISM design adopted a leaktight containment dome covering the reactor head.

By contrast, the KU1 use of HALEU fuel leads to a softer neutron spectrum, high negative Doppler worth compared to other metallic-fueled SFRs, high delayed neutron fraction, and core-wide negative void worth (see SE section 3.11). Combined with mechanical pumps with inherent coastdown characteristics provided by rotating machinery (see SE section 7.1.3), and a highly reliable scram system (summarized in SE section 1.3.3.3.2), transients that could lead to substantial fuel melt or even coolant boiling have a low enough frequency of occurrence that they are not identified as LBEs under LMP (see SE section 3.4). These same physical

characteristics, coupled with the use of LMP, enable the use of a functional containment concept with a variety of barriers, including the NSRST head access area as a functional containment boundary, as summarized in SE section 1.3.3.3.2. Finally, the molten salt system used at KU1 removes the potential for sodium-water reaction and enables a separate decay heat removal pathway through the IAC, which is on the IHT system. This provides an important DID decay heat removal system that was not present in PRISM.

1.6 Summary of Operations

Plant operations are addressed to the extent necessary for a CP application in chapters 8, 9, 10, 11, and 12 of this SE.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

The Nuclear Waste Policy Act of 1982 (42 USC § 10101) provides that the U.S. Government is responsible for the permanent disposal of high-level radioactive waste and spent nuclear fuel, but the cost of disposal should be the responsibility of the generators and owners of such waste and spent fuel. Guidance for the staff to evaluate compliance with the Nuclear Waste Policy Act is provided in DANU-ISG-2022-01.

The applicant's letter submitting the CP application stated that "TerraPower is in good faith negotiations with the Department of Energy to enter into a contract for the disposal of high-level waste and nuclear fuel under section 302(b) of the Nuclear Waste Policy Act of 1982, as amended." The staff notes that while USO is a wholly owned subsidiary of TerraPower, the expectation is that the permit holder, USO, will have a contract for waste disposal with the DOE as described above. By letter dated October 3, 2025 (ML25276A027), USO provided documentation from DOE that USO is actively and in good faith negotiating on a contract under section 302(b) of the Nuclear Waste Policy Act. Because USO has provided documentation of good faith negotiations with the Department of Energy, the staff finds that USO is in compliance with the Nuclear Waste Policy Act at the CP stage, consistent with DANU-ISG-2022-01.

1.8 Tables

Table 1.8-1: Topical Reports Incorporated by Reference into the Application

Topical Report Number	Topical Report Title	ADAMS Accession Number	Safety Evaluation Report Section
TP-QA-PD-0001	Quality Assurance Program Description Topical Report	ML23116A179	8.2
NAT-3056-A	Plume Exposure Pathway Emergency Planning Zone Sizing Methodology	ML25104A001	11.3
NAT-2806-A	Fuel and Control Assembly Qualification	ML25083A296	3.10
NATD-LIC-RPRT-0002-A	Principal Design Criteria for the Sodium Advanced Reactor	ML24283A066	5.3
NAT-3226-A	An Analysis of Potential Volcanic Hazard at the Proposed Sodium Site Near Kemmerer, Wyoming	ML24303A409	2.7
NAT-2965-A	Human Factors Engineering Program Plan and Methodologies	ML25211A232	11.2
NAT-9392-A	Radiological Source Term Methodology	ML25211A271	3.2
NAT-9390-A	Design Basis Accident Methodology for in-vessel events without Radiological Release	ML25211A127	3.3
NAT-9391-A	Radiological Release Consequences Methodology	ML25211A267	3.3
NAT-9393-A	Stability Methodology	ML25211A276	3.11
NAT-9395-A	Partial flow Blockage	ML25251A084	3.3
NAT-4950-A	Instrumentation and Control Architecture and Design Basis	ML25232A241	7.6.3, 7.6.5, and 7.6.7
NAT-8922	Reactor Seismic Isolation System Qualification	ML25195A156	7.1.2
NAT-9394-A	DBA Transient Methods for Events with Radiological Release	ML25251A090	3.3

Table 1.8-2: Technical Reports Incorporated by Reference in the Application

Technical Report Number	Technical Report Title	Safety Evaluation Report Section
TP-LIC-RPT-0011	Core Design and Thermal Hydraulic Technical Report	3.11, 3.12
TP-LIC-RPT-0012	Preliminary Emergency Planning Zone Determination Analysis	11.3

Table 1.8-3: Three Mile Island Requirements

Regulation	Description	Safety Evaluation Report Section or Comment
10 CFR 50.34(f)(1)(i)	PRA to seek improvements in reliability of heat removal systems	3.1.1
10 CFR 50.34(f)(1)(iii)	Reactor coolant pump seal damage	Listed in DANU-ISG-2022-01, appendix B table 4 as applicable only for reactor designs that have a coolant pump with seals that retain inventory credited for core cooling. This is not applicable to the KU1 design because it does not have coolant pumps with seals that retain inventory credited for core cooling.
10 CFR 50.34(f)(1)(xii)	Perform an evaluation of alternative hydrogen control systems	Not listed in DANU-ISG-2022-01, appendix B table 4 as being generically applicable but it is listed in KU1 table 1.4-4 PSAR as being applicable. See section 7.2.4 of this SE.
10 CFR 50.34(f)(2)(i)	Control room simulator	11.2
10 CFR 50.34(f)(2)(ii)	Plant procedure improvement program	11.2
10 CFR 50.34(f)(2)(iii)	Control room human factors	11.2
10 CFR 50.34(f)(2)(iv)	Safety parameter display system	11.2
10 CFR 50.34(f)(2)(v)	Automatic indication of status of safety systems	11.2
10 CFR 50.34(f)(2)(vi)	High point venting of reactor coolant system (RCS)	Listed in DANU-ISG-2022-01, appendix B table 4 as applicable only if reactor coolant flow is credited for core cooling and coolant flow can be impeded by non-condensable gases. KU1 table 1.4-4 PSAR notes that this is not applicable to the design. Based on the design of the KU1 primary coolant system, which is arranged in a pool-type configuration with a liquid surface level to which non-condensable gases can migrate, the staff determined that

Table 1.8-3: Three Mile Island Requirements

Regulation	Description	Safety Evaluation Report Section or Comment
		coolant flow cannot be impeded by non-condensable gases and as such concluded that this requirement is not applicable to the design.
10 CFR 50.34(f)(2)(vii)	Radiation shielding design review	10.1
10 CFR 50.34(f)(2)(viii)	Post-accident sampling	KU1 table 1.4-4 notes that the design is partially compliant with this item. The staff's evaluation is in section 7.2.4 of this SE.
10 CFR 50.34(f)(2)(x)	Relief and safety valves – provide a test program and associated model development and conduct tests to qualify RCS relief and safety valves.	Listed in DANU-ISG-2022-01 table 4 as applicable only if RCS has relief valves and failure of these valves would lead to core cooling challenges. KU1 table 1.4-4 notes that the design is partially compliant with this item. The staff's evaluation is in section 7.2.3 of this SE
10 CFR 50.34(f)(2)(xi)	Relief and safety valves – provide direct indication of relief and safety valve position in the control room	11.2
10 CFR 50.34(f)(2)(xiv)	Containment isolation	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. This is not applicable to the KU1 design because it uses a functional containment.
10 CFR 50.34(f)(2)(xv)	Containment purging	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. This is not applicable to the KU1 design because it uses a functional containment.
10 CFR 50.34(f)(2)(xvii)	Control room instrumentation for containment functions	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. Although KU1 uses a functional containment, PSAR table 1.4-4 notes that portions of this item (i.e., those related to containment pressure, containment radiation intensity, and providing for effluent monitoring) are applicable. See section 11.2 of this SE.
10 CFR 50.34(f)(2)(xviii)	Coolant instrumentation	7.1.3, 11.2

Table 1.8-3: Three Mile Island Requirements

Regulation	Description	Safety Evaluation Report Section or Comment
10 CFR 50.34(f)(2)(xix)	Post-accident monitoring	7.6.5, 11.2
10 CFR 50.34(f)(2)(xxv)	Provide an onsite technical support center and onsite operational support center	PSAR table 1.4-4 notes that the new emergency preparedness rule does not require these specific facilities (see SE 11.3).
10 CFR 50.34(f)(2)(xxvi)	Leakage control outside containment	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. PSAR table 1.4-4 notes that the design uses a functional containment and not a single containment structure and that barriers and boundaries classified as SR will be leak testable to demonstrate their performance. However, the PSAR also states in table 1.4-4 that SR barriers will be leak-testable and will be designed for leakage control and detection, and a leakage control program will be provided. See SE section 8 and 11.2.
10 CFR 50.34(f)(2)(xvii)	In-plant Radiation Monitoring	10.1
10 CFR 50.34(f)(2)(xviii)	Preclude control room habitability issues during accidents	10.1
10 CFR 50.34(f)(3)(i)	Industry experience	11.2
10 CFR 50.34(f)(3)(ii)	Quality Assurance (QA) list includes all SSCs important to safety	The QA list is discussed in section 2 of TR TP-QA-PD-0001 which is incorporated by reference into the PSAR as shown in PSAR table 1.4-2. As stated in the QA TR, the list includes all SR and NSRST SSCs, which as discussed in staff position C.6.a.(2) of RG 1.253 includes both SR and NSRST SSCs. Based on this, the staff determined that 10 CFR 50.34(f)(3)(ii) is met as discussed in the QA TR, which is incorporated by reference into the PSAR. Further evaluation of USO's QA program is provided in chapter 8 of this SE.
10 CFR 50.34(f)(3)(iii)	QA program	8.1
10 CFR 50.34(f)(3)(iv)	Dedicated containment penetrations	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. Not applicable to KU1

Table 1.8-3: Three Mile Island Requirements

Regulation	Description	Safety Evaluation Report Section or Comment
		design because it uses a functional containment.
10 CFR 50.34(f)(3)(vi)	Containment	Listed in DANU-ISG-2022-01 table 4 as applicable only for designs that use a traditional containment rather than a functional containment approach. Not applicable to KU1 design because it uses a functional containment.
10 CFR 50.34(f)(3)(vii)	Management plan for design and construction activities	11.1

2 SITE CHARACTERISTICS

This chapter of the U.S. SFR Owner, LLC (USO) Kemmerer Power Station Unit 1 (KU1) construction permit (CP) safety evaluation (SE) describes the U.S. Nuclear Regulatory Commission (NRC) staff's (the staff's) technical review and evaluation of the preliminary information on site characteristics provided in chapter 2 of the KU1 preliminary safety analysis report (PSAR). Site characteristics include geography and demography; nearby industrial, transportation, and military facilities; meteorology; hydrology; and geology, seismology, and geotechnical engineering.

The applicable regulatory requirements for the evaluation of site characteristics are as follows:

- Title 10 of the *Code of Federal Regulations* (CFR) 50.34(a)(1)(ii),
- 10 CFR 100.3, "Definitions,"
- 10 CFR 100.20(b),
- 10 CFR 100.21(a), (b), (d), (e) and (h), and
- 10 CFR 100.23, "Geologic and seismic siting criteria."

The applicable guidance for the evaluation of the site characteristics is as follows:

- DANU-ISG-2022-02, "Chapter 2, 'Site Information,'" March 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23277A140)

The regulations in 10 CFR 50.34(a)(3)(i), require a CP applicant to include in the PSAR the PDC for the facility. The PDC for KU1 are incorporated by reference into the PSAR via topical report (TR), NATD-LIC-RPRT-0002-A, "Principal Design Criteria for the Sodium Advanced Reactor," Rev 1 (ML24283A066). PSAR section 5.3, lists the PDC from the TR and provides a brief description of how each PDC is addressed by the design. The staff found the PDC for the Sodium plant to be acceptable, as documented in its SE included with the approved version of NATD-LIC-RPRT-0002-A. The PDC that apply to the evaluation of site characteristics and site parameters is:

- PDC 2, "Design Bases for Protection Against Natural Phenomena"

2.1 Site Characteristics and Site Parameters

PSAR section 2.1 summarizes the site-specific external Design Basis Hazard Levels (DBHLs) and site characteristics related to geography, demography, meteorology, hydrology, geology, seismology, geotechnical engineering, potential volcanic activities, and turbine missiles. PSAR table 2.1-1 summarizes the site-specific external DBHLs, and table 2.1-2 summarizes the site characteristics. These DBHLs and site characteristics, based on analyses in PSAR sections 2.2 through 2.7, are those necessary to establish the findings required by 10 CFR 50.34, "Contents of applications; technical information," and 10 CFR 100, "Reactor Site Criteria." The tables include references to sections with additional information.

The staff reviewed the information in PSAR section 2.1 to determine the sufficiency of the site characteristics and postulated external DBHLs for CP issuance. This evaluation is conducted in accordance with 10 CFR 50.34, 10 CFR 100, and the acceptance criteria provided in DANU-ISG-2022-02 section 2.1. The staff determined that the applicant characterized applicable site-specific natural phenomena consistent with PDC 2, as described in PSAR section 5.3.1.2 and that the external DBHLs are established using this site characterization information. Based on review of PSAR tables 2.1-1 and 2.1-2, and PSAR sections 2.2 through 2.7, the staff determined the application provides and substantiates sufficient information to establish the actual characteristics of the proposed site, and that the facility meets the underlying regulatory requirements. The staff determined that the information in PSAR chapter 2 is sufficient to evaluate egress pathways needed for: (i) emergency plans following or in anticipation of a release of radioactive material, and (ii) establishing effective security measures. The staff found that the information provided in PSAR chapter 2 is sufficient to demonstrate low radiological risk to the public.

In summary, the staff determined that the information provided in PSAR section 2.1 adequately characterizes the site and establishes appropriate external DBHLs for the proposed facility. The siting information is sufficient to support further safety analyses and emergency and security plans.

2.1.1 Conclusion

The staff has reviewed PSAR section 2.1 and confirmed that the applicant addressed the required information relating to site characteristics. The staff's review of tables 2.1-1 and 2.1-2 verified that the site information is sufficient to support the further safety analyses and emergency and security plans. Therefore, the staff concludes that the applicant has met the requirements of 10 CFR 50.34(a)(1)(ii) and 10 CFR 100.3, 100.20, "Factors to be considered when evaluating sites," and 100.21, "Non-seismic siting criteria."

2.2 Geography and Demography

PSAR section 2.2 provides: 1) a description of the site location in PSAR section 2.2.1; 2) a description of the exclusion area including the USO's legal rights with respect to all areas that lie within the exclusion area in PSAR section 2.2.2; and 3) a discussion of the population distribution around the site in PSAR section 2.2.3.

Section 2.2 of DANU-ISG-2022-02 provides the applicable guidance for the review. The application should include sufficient information to understand the site geography and demography. The regulations in 10 CFR 100.21(a) require an exclusion area and low population zone (LPZ) as defined in 10 CFR 100.3 to be established.

2.2.1 Site Location and Distribution

PSAR section 2.2.1.1 states that the KU1 site is located in the United States Geological Survey (USGS) 7.5-minute Elkol quadrangle in Lincoln County, Wyoming. The proposed facility and its supporting infrastructure will be located on an area of approximately 178 acres. The center point of the Reactor Building (RXB) will be at 41.705841° N latitude and 110.560547° W longitude, as given in PSAR table 2.2-1. PSAR section 2.6.1.3.3 states that the proposed site is located within

the Cumberland flats, a broad relatively flat north–south trending valley. The nearest ridge, Oyster Ridge, is approximately 0.5 miles (mi) (0.8 kilometers (km)) east of the site, as shown in PSAR figure 2.4-63. An unnamed ridge is the western boundary of the Cumberland Flats. As stated in PSAR section 2.2.1.1, North Fork Little Muddy Creek flows through eastern portion of the property.

PSAR section 2.2.1.1 states that the site lies south of the intersection of US 189 and US 30 highways and east of the intersection of US 189 and Union Pacific Railway line. At the closest approach, the Union Pacific rail line is approximately 0.15 mi (0.24 km) from the proposed site. PSAR section 2.2.1.2 states that there are minor roads, such as County Road 325, Elkol Road, and site access and service roads of the Kemmerer Mine, in the immediate vicinity of the proposed site.

PSAR section 2.2.1.1 states that the site is approximately 3 mi (4.8 km) south of the Kemmerer City municipal boundary and approximately 4.2 mi (6.8 km) southwest of the town of Diamondville. The Pacific Naughton Power Plant (PNPP) is approximately 3.8 mi (6.1 km) away from the site. The power plant operates two coal-fired units (Naughton 1 and 2) and a natural gas-fired unit (Naughton 3). The Kemmerer Mine (KM) extracts coal using the open pit method. Its permit boundary is approximately 2.2 mi (3.5 km) west of the proposed site. The Kemmerer Municipal Airport (EMM) is approximately 7.1 mi (11.4 km) north of the proposed site.

The staff reviewed the information presented in PSAR section 2.2.1 to understand the site geography. The staff notes that USO provided the specific 7.5-minute quadrangle of the USGS survey map in which the proposed site is located in addition to Federal, State, and County jurisdictions in PSAR figures 2.2-22 and 2.2-23. The staff also notes that prominent natural features such as ridges and creeks have been identified. In addition, prominent human-made features near the proposed site, such as the PNPP, KM, and EMM, are also located in these figures with respect to the proposed reactor location. Nearby highways, rail lines, and other local roads are identified in PSAR figures 2.2-22 and 2.2-23. These figures contain sufficient information to identify types and locations of both natural and human-made features, and any potential hazards associated with them. In addition, these figures indicate local, State, and Federal jurisdictions associated with the proposed site and its surrounding area. Based on the staff's review of the information discussed above, the staff determined that PSAR section 2.2.1 and figures 2.2-22 and 2.2-23 provide sufficient information to locate the proposed site. In addition, the staff determined the description of the proposed site is sufficiently detailed for reviewing other PSAR sections.

2.2.2 Exclusion Area Authority and Control

PSAR section 2.2.1.2 states that the Exclusion Area Boundary (EAB) and the LPZ of the proposed facility coincide and are defined by a circle having a radius of 0.25 mi (0.4 km) from the RXB center point. The EAB and LPZ are shown in PSAR figure 2.2-21. As can be seen in this figure, both EAB and LPZ extend beyond the KU1 property boundary. PSAR section 2.2.2.1 states that USO will have authority and control over all activities in the EAB including exclusion and removal of personnel and property (e.g., during an emergency). USO will maintain agreements with appropriate entities within the EAB but outside the USO ownership boundary to ensure that appropriate plans including protective actions, as warranted, would be in place within these areas outside of the USO ownership, as stated in PSAR section 2.2.2.1.

Although the area in the vicinity of the proposed site has potentially exploitable minerals including coal, bentonite, phosphorus, sulfur, oil, and gas, there are no mines or oil or gas wells within the proposed site or directly adjacent to the site boundary, as stated in PSAR section 2.2.2.1. PSAR section 2.2.2.1 also states that USO will acquire all mineral rights for the site. In addition, PSAR section 2.2.2.2 states that the Sodium Test and Fill Facility, owned by TerraPower, LLC (TerraPower), a 100 percent owned subsidiary of USO, the applicant, is within the EAB but outside the property boundary of the proposed site.

The staff reviewed the description of the EAB in PSAR section 2.2.2.1 including USO's rights to control of all activities in the EAB in PSAR sections 2.2.2.2 through 2.2.2.4 and, additionally, the portion of the EAB that is outside the owner control area shown in PSAR figure 2.2-21. In PSAR section 2.2.2.1, USO commits to maintain agreements with appropriate entities of this portion of the land outside its ownership. As stated in PSAR section 2.2.2.1, the agreements will also include protective actions, as warranted, throughout the operational life of the proposed facility. Because of these agreements, the staff determined that USO will have control over all activities in this area.

PSAR sections 2.2.2.3 and 2.2.2.4 (and as shown in PSAR figure 2.2-21) state that there are no Federal, State, or County roads traversing the exclusion area. Consequently, no public roads need to be relocated or abandoned. The staff determined, based on its independent review using Google Maps, that no roads traverse the EAB. Consequently, the staff determined that no arrangements would be necessary to control traffic within the EAB and no relocation or abandonment of public roads would be necessary.

As previously described, the Sodium Test and Fill Facility located in the EAB is owned by TerraPower who fully owns USO. As stated in PSAR section 2.2.2.2, USO has control of all activities within the EAB. Based on this discussion, the staff determined that USO will have control of all activities in the Sodium Test and Fill Facility including removal of personnel, if warranted.

As stated in PSAR section 2.2.2.1, the proposed site and its vicinity have exploitable minerals and USO plans acquire all mineral rights for the site. The staff reviewed the supporting documentation during the audit (ML25302A443) which confirmed the statements in the PSAR regarding the mineral rights within the EAB but outside the property boundary. Since there are no mines or oil and gas wells directly adjacent to the site, as described in section 2.2.2.1 of the PSAR, and USO plans to acquire all mineral rights the staff determined that USO will maintain appropriate control within the EAB.

In summary, based on the staff's review, the staff determined that USO will have control over all activities within the EAB during the operating life of the proposed facility.

2.2.3 Population Distribution

PSAR section 2.2.3 provides the estimated resident and transient population within a 50-mi (80 km) radius from the center point of the RXB based on the 2020 decennial census data from the US Census Bureau. The population distribution has been presented in 10-year increment starting at 2030, the projected date of facility approval, to year 2090 by calculating an annualized growth rate using the 2020 census data and projection of county-wise population data from the Wyoming Department of Administration, Economic Analysis Division, and the

Kem C. Gardner Policy Institute of the University of Utah. The transient population was estimated and included with the resident population for the band 0 to 10 mi (0 to 16 km) from the RXB center point.

PSAR tables 2.2-2 and 2.2-3 provide the total resident and transient population information within 10 mi (16 km) and only resident population from 10 to 50 mi (16.1 to 80 km), respectively, in 10-yr increment starting at 2020 and ending in 2090. PSAR table 2.2-4 shows the cumulative population distribution within 0 to 5 mi (0 to 8 km) in 1-mi (1.6-km) increment, and also within 5 to 10 mi (8 to 16 km) and 10 to 20 mi (16.1 to 32.2 km) from RXB center point as given in 2020 Census data and projected to 2030 and 2090. This table also compares the current and projected population densities (number of individuals per square mile) with the 500 individuals per square mile criterion of Regulatory Guide (RG) 4.7, "General Site Suitability Criteria for Nuclear Power Stations," (ML12188A053).

PSAR figures 2.2-2 through 2.2-18 show the population distribution divided into ten concentric bands of 0 to 1 mi (0 to 1.6 km), 1 to 2 mi (1.6 to 3.2 km), 2 to 3 mi (3.2 to 4.8 km), 3 to 4 mi (4.8 to 6.4 km), 4 to 5 mi (6.4 to 8 km), 5 to 10 mi (8 to 16.1 km), 10 to 20 mi (16.1 to 32.2 km), 20 to 30 mi (32.2 to 48.3 km), 30 to 40 mi (48.3 to 64.4 km), and 40 to 50 mi (64.4 to 80.5 km) from the RXB center point and within sixteen equal directional sectors, each sector consisting of 22.5°.

PSAR figure 2.2-22 shows that Kemmerer is the largest community within 10 mi (16.1 km) of the proposed site. The town of Diamondville is also within the 10 mi (16.1 km) region of the proposed site. PSAR figure 2.2-10 shows the area within 50 mi (80.5 km) of the RXB center point. This zone includes Lincoln, Sublette, Sweetwater, and Uinta counties of Wyoming; Cache, Morgan, Rich, and Summit counties of Utah; and Bear Lake County of Idaho. Only the resident population is presented PSAR figures 2.2-10 through 2.2-18.

Transient population near the proposed site includes individuals working in the nearby industries, recreational areas, hotels and motels, recreational vehicle parks. It also includes seasonal residents, and migrant populations, as described in PSAR section 2.2.3.3. Major employers near the proposed site include the PNPP, Kemmerer Operations LLC coal mine, Cowboy State Trucking, South Lincoln Medical Center, and South Lincoln Nursing Center, which account for 263 transient individuals within 10 mi (16.1 km) radius from the RXB center point. Recreational opportunities account for 1,190 transient population within 10 mi (16.1 km) radius, approximately 1,000 of them are visitors attending the Oyster Creek Music Festival, held annually in Kemmerer. Other recreational opportunities near the proposed site include the JC Penney Historic District National Historic Landmark, the Fossil County Frontier Museum, and the Fossil Island Gold Club. In addition, hotels, motels, and recreational vehicle parks account for another 548 transient population within the 10 mi (16.1 km) area. The Census Bureau estimated that 64 individuals reside seasonally in this area.

Although the University of Wyoming–Lincoln County Extension and the Wyoming Department of Workforce Service state that there are no crop farms or livestock ranches within 10 mi (16.1 km) of the proposed site, USO identified a parcel, owned by Aggies Grazing LLC, 2.8 mi (4.5 km) away. In addition, local ranchers employ migrant workers to move their livestock (generally sheep) between winter and summer ranges. PSAR section 2.2.3.3 estimates that a maximum of approximately 15 sheepherders could be present at one time. Given the mobile nature of sheepherding operation, USO counted 15 transient migrant workers within each sector of the

population distribution grid, as described in PSAR figure 2.2-1, along the observed trails of sheep movement.

PSAR section 2.2.3.4 states that there is no residual or transient population within the LPZ. No public road traverses the LPZ, as shown in PSAR figure 2.2-21. In addition, no towns, recreational facilities, hospitals, schools, prisons, or beaches are within the LPZ.

PSAR section 2.2.3.5 states that the nearest population center with population greater than 25,000 residents is the city of Logan, Utah, approximately 63 mi (101.4 km) west of the proposed site. Based on census, Logan had a resident population of 52,778 in 2020. The city of Rock Springs, Wyoming, which is located approximately 66 mi (106.2 km) to the east of the site and had a 2020 population of 23,526.

PSAR section 2.2.3.6 describes the variation of population density in terms of radial distance from the RXB. PSAR figure 2.2-19 shows the population density using the 2020 Census data up to 20 mi (32.2 km) from the site along with the density of projected population in 2030, approximate date of facility approval, and in 2090, assumed date of end of operations. These population variation data are compared with the regulatory guidance of cumulative 500 persons per square mile up to 20 mi (32.2 km) from the proposed reactor location, as given in RG 4.7. PSAR figure 2.2-19 shows that the current and projected population densities are significantly smaller than those given in RG 4.7.

The staff has reviewed the information presented in PSAR section 2.2.3, PSAR tables 2.2-2 through 2.2-4, PSAR figures 2.2-2 through 2.2-18, and PSAR figure 2.2-21. The staff notes that the population distribution information surrounding the proposed site is based on the latest 2020 US Census data. The staff also notes that USO has used appropriate and authoritative sources, such as the Economic Analysis Division of the Wyoming Department of Administration and the Kem C. Garner Policy Institute of the University of Utah, to project the county-wise population growth during the operational life of the proposed facility. Additionally, the staff notes that USO has studied the potential sources of transient population and appropriately distributed it to the sector(s) shown in PSAR figure 2.2-1.

The staff further notes that for any county with a negative population growth rate, USO assumed a growth rate of 1.0, neither a decrease nor an increase of population. The staff determined this is a conservative approach as it overestimates the population in future. The staff notes that the population of Kemmerer, the largest community within 10 mi (16.1 km) of the proposed site, was 2,651 in 2000. The population decreased to 2,451, a reduction of 9.1 percent, in 2020. As stated, in the analysis of future projection of population, USO has assumed no decline of population.

The staff observed from PSAR table 2.2-2 and PSAR figures 2.2-2 through 2.2-9 that approximately 4 percent of the total population within 10 mi (16.1 km) of the proposed facility reside within 1 mi (1.6 km) radius. Approximately, 79 percent reside beyond 5 mi (8 km), most of them in Kemmerer and Diamondville. Proportion of population remains almost same in each mile band (PSAR figures 2.2-2 through 2.2-10) in each 10-yr projection till 2090. In addition, the staff observes that majority of the population live north of the proposed site. Only 15 to 18 residents live in each mile band in the southern side. Consequently, population increase projected in future is significantly large in the north of the proposed site.

The staff determined that USO has adequately identified the LPZ in PSAR sections 2.2.2 and 2.2.2.4, and PSAR figure 2.2-21. In addition, USO has adequately described the protective measures that can be implemented in the LPZ for the population, if warranted. The staff notes, as shown in PSAR section 2.2.2.5, that the nearest population center with at least 25,000 residents, Logan, Utah, is located more than 250 times the KU1 LPZ distance from the proposed site. This exceeds the minimum requirement of 1.33 times the LPZ distance, as specified in 10 CFR 100.21(b). Consequently, the staff concludes that the proposed site meets the regulatory criterion for minimum distance to the nearest population center and is appropriately located away from densely populated areas. The staff also notes, based on PSAR figure 2.2-19, that the population density in 2020 and projected to 2030 and 2090 are substantially smaller than the reference values in RG 4.7. Therefore, the staff determined that the proposed reactor site is suitably distant from densely populated centers.

2.2.4 Conclusion

Based on its review, the staff determined that the geographical and demographical information provided in the PSAR is sufficiently detailed and accurate to provide the necessary bases to allow accurate assessments of the potential radiological impact on the public resulting from the siting and operation of the proposed KU1 facility, including analysis (e.g., dose calculations) presented in other PSAR chapters. The staff also determined that no geographic or demographic characteristics of the KU1 site render the site unsuitable for operation of the KU1 facility, and that the information provided meets the applicable acceptance criteria of DANU-ISG-2022-02, section 2.2. Accordingly, the staff determined that the level of detail provided on geography and demography demonstrates an adequate design basis for the KU1 facility.

2.3 Nearby Industrial, Transportation, and Military Facilities

PSAR section 2.3 provides a description of potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports located near the site to establish whether the effects of potential hazards onsite or in the vicinity of the site should be considered as design basis hazards in accordance with 10 CFR 100.20(b). USO evaluated each of the identified hazards following RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Rev. 2 (ML21253A071), and RG 1.91, "Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes near Nuclear Power Plants," Rev. 3 (ML21260A242). Section 2.3 of DANU-ISG-2022-02 provides the applicable guidance for the review. Under the regulations in 10 CFR 100.20(b) and 10 CFR 100.21(e), the applicant must evaluate potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports. The applicant should also determine whether bulk storage or transportation of hazardous materials may occur at or near the site and should assess the impact of potential explosions (see: RG 1.91). If applicable, the applicant should assess aircraft hazards associated with nearby airports, Federal airways, holding and approach patterns, military airports, training routes, and training areas in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," section 3.5.1.6 (ML100331298).

2.3.1 Identification of Nearby Facilities and Transportation Routes

In PSAR sections 2.3.1.1 and 2.3.3, USO identified the potential hazards from nearby facilities and major transportation routes within 10 mi (16.1 km) of the proposed facility using the guidance in RG 1.78, RG 1.91, and RG 4.7, and presented them in PSAR figures 2.3-1 through 2.3-3. Additionally, facilities and activities greater than 5 mi (8 km) were assessed if they could affect the safety-significant structures, systems, and components (SSCs). Special attention was given to activities within 0.6 mi (1 km) that could pose credible hazards to safety-significant SSCs of the proposed facility.

Three industrial facilities within 10 mi (16.1 km) include the PNPP, KM, and Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant. Highways within 5 mi (8 km) of the proposed site include US 30, US 189, and Wyoming State highways WYO 240 and WYO 412. Minor roadways in the immediate vicinity of the proposed site include County Road 325, Elkol Road, site access and service roads to the KM, Alleman Road, North Alleman Road, Naughton Road, and access road to the proposed site.

As US 189 is the closest and most important roadway within 5 mi (8 km) of the proposed site, USO has concluded that the effects of hazardous chemicals transported on US 189 would bound the effects from other highways and roads at larger distances from the proposed site. The staff determined this assessment is acceptable. The nearest railroad is the Union Pacific Railroad approximately 0.15 mi (0.24 km) from the proposed site.

PSAR sections 2.2.1.2 and 2.3.1.3.1 state that the proposed site is located south of the intersection of US 189 and US 30, and east of the intersection of US 189 and the Union Pacific Railroad. Wyoming State Highways WYO 240 and WYO 412 are within the site vicinity. Minor roadways in the immediate vicinity of the proposed site include County Road 325, Elkol Road, access and service roads to KM, Alleman Road and North Alleman Road, Naughton Road, and the access road to the proposed facility. As US 189 is not only the most trafficked highway within 5 mi (8 km) of the proposed site but also the nearest highway, USO concluded that hazardous materials transported on US 189 would produce the bounding hazards. The staff determined this acceptable as US 189 is the nearest highway appropriately 0.2 mi (0.3 km) west of the boundary of the proposed site and is also the most important highway in terms of cargo and vehicular traffic.

PSAR section 2.3.1.2.3 identified five natural gas pipelines within 5 mi (8 km) of the proposed site: (1) Ignacio-Sumas Loop, a 24-inches (in.) (61-centimeters (cm)) diameter pipeline; (2) Ignacio-Sumas, a 22-in. (59-cm) diameter pipeline; (3) JL95, a 10-in (25-cm). diameter pipeline; (4) FL91, a 3-in. (8-cm) diameter pipelines; and (5) FL91, a 4-in. (10-cm) diameter pipeline. PSAR figure 2.3-2 shows the location of first three large diameter pipelines. The Williams Company operates two of these pipelines: the Ignacio-Sumas Loop pipeline located approximately 2.44 mi (3.9 km) northeast of the RXB center point, and Ignacio-Sumas pipeline approximately 2.57 mi (4.1 km) northeast of the RXB center point. Dominion Energy Questar operates the JL95 pipeline located approximately 2.82 mi (4.5 km) northeast of the RXB center point, as stated in PSAR section 2.3.2.2.

The FL91 pipelines are connection pipelines to the nearby PNPP. Due to their small diameter and short lengths, USO concluded that an evaluation of hazards from these two pipelines would not be needed. The staff agrees because a complete rupture of these pipelines would only

release a relatively small quantity of natural gas which would dilute to below the low flammability limit of natural gas (i.e., below 5 percent) before reaching the KU1 site, approximately 3.8 mi (6.1 km) away.

2.3.2 Hazards from Industrial Facilities

2.3.2.1 Pacific Naughton Power Plant

PSAR section 2.2.1.1 states that PNPP in Lincoln County, Wyoming, is approximately 4.05 mi (6.5 km) away from the reactor center point for the proposed KU1 and has three electrical generating units. Naughton Unit 1 and Unit 2 currently use coal to produce electricity. PSAR sections 2.3.2.1.2 and 2.3.2.7 state that Unit 1 and Unit 2 are planned to be converted to use natural gas in 2026. Unit 3 was converted to use natural gas in 2019. These power generating units are expected to continue operating through 2036. Unit 1 currently produces 156 MW and Unit 2 201 MW of electricity. Capacity of Unit 3 is 247 MW.

The staff reviewed the information provided in PSAR sections 2.2.1.1, 2.3.2.1.2, and 2.3.2.7, regarding the hazards that PNPP may pose to the proposed facility. Potential hazards posed by coal used to generate electricity in Naughton Unit 1 and Unit 2 are evaluated below by the staff. In addition, as stated in PSAR section 2.3.1.2.3, two small diameter natural gas pipelines, FL91 and FL91, at the PNPP are connection pipelines. Due to their short lengths, small diameter, and distance to the proposed site, the staff determined that any potential hazards to the proposed facility would be bounded by the hazards posed by three significantly larger diameter pipelines described in PSAR section 2.3.1.2.3.

2.3.2.2 Kemmerer Coal Mine and Other Coal Mines

As stated in PSAR section 2.6.3.1.2, several coal mines exist near the proposed site. The KM is an active open pit (surface operation) coal mine. There are numerous abandoned coal mines in the area. Many of these mines operated in the coal rich Dry Holbro member of the Frontier Formation. Several of them are located near the proposed site. Nearest abandoned mine to the proposed site is approximately 0.6 mi (1 km) east of the site. It was a small operation with areal extent a few tens of meters. PSAR section 2.6.3.1.2 states that no abandoned coal mine underlies the proposed site and that these abandoned mines were surface operations and do not extend to much depth. Additionally, no mine shaft was located nearby which extends under the site. Based on the staff's review of the PSAR, the staff concludes no mining operation existed before or currently exists to extract coal from underneath the proposed site. Consequently, any potential hazards to the proposed facility from collapse of existing or old mine workings is not a credible hazard. Therefore, the staff review focused on other hazards associated with the mining operations at the KM to the proposed facility. In addition, as discussed in SE section 2.2.2, potential for opening of new coal mines is negligible as USO will control all activities within the EAB.

PSAR section 2.3.2.1.3 states that the KM, owned by Kemmerer Operations LLC, is located approximately 3.11 mi (5 km) away from the center point of the proposed KU1 site. This is a surface coal mine, which extracts coal by open pit method using drilling and blasting. PSAR section 2.3.2.1.3 also states that processed coal at KM is stored in concrete silos and in open air stockpiles. Hazards from fugitive coal dust clouds from open air stockpiles and coal fires are mitigated/controlled using water sprays, chemical dust suppression, and windbreaks. In

addition, a high rate of coal turnover reduces the possibility of coal undergoing autoignition in storage. PSAR section 2.3.2.1.3 also states that release of ignitable coal dust clouds is rare in KM. Based on this information, USO concluded in PSAR section 2.3.2.1.3 that potential combustible dust clouds or autoignition of coal from KM are not credible hazards to the proposed facility.

The staff reviewed the information in PSAR section 2.3.2.1.3 on handling of extracted coal at KM and mitigation/controlling measures taken to minimize the potential for generating combustible coal dust clouds. The staff determined that the mitigative measures, such as water sprays, chemical dust suppression, and windbreaks, used by KM reduce the potential for fire hazard and generation of coal dust clouds minimizing dust explosions. In addition, KM reduces the potential for autoignition of coal by increasing the turnover rate at the stockpile. The staff also determined that the large intervening distance is expected to eliminate any potential hazards associated with a coal dust explosion and autoignition of the stockpiled coal. Therefore, based on the preceding discussion, the staff determined that handling of coal at nearby KM would not pose a credible hazard to the proposed facility.

The staff also determined that KM uses surface drilling and blasting to extract coal from the coal seams. The staff audited USO's internal documentation related to evaluating the hazards associated with use of explosives at KM. The staff audited the transportation of explosives to the KM and confirmed that the applicant did not use US 189. The staff determined that USO's conclusion that an accidental explosion of the explosives while on transit to the KM is not a credible hazard is acceptable.

The staff considered other hazards that might be created to the proposed facility from blasting at the KM. A typical mine blast at the KM will not generate ground vibration that can be damaging to the reinforced concrete structures at the proposed facility more than 3 mi (4.8 km) away. In addition, the distance of the proposed sites to the locations of mine blasts ensure that any flying rock is not a credible hazard to the safety-significant SSCs. Therefore, the staff concludes that explosives used in blasting at the KM would not produce any credible hazards to the proposed facility.

2.3.2.3 Oil and Gas Wells

USO states in PSAR section 2.6.3.1.6 that two nearest active gas wells are approximately 3 mi (4.8 km) and 5 mi (8 km) away from the proposed site. In addition, there are five abandoned gas wells and one abandoned oil well nearby. The nearest abandoned gas well is approximately 2.5 mi (4 km) southwest. PSAR section 2.6.3.1.7 states that no fracking operations take place nearby. In addition, no wastewater injection wells were observed nearby the site. This was confirmed by the Wyoming Oil and Gas Commission and Wyoming Department of Environmental Quality Underground Injection Control reports (PSAR references 2.6-89 and 2.6-91). Therefore, the staff determined that nearby oil and gas extraction operations do not pose a credible hazard to the proposed facility.

2.3.3 Hazards from Transportation Routes

2.3.3.1 Highways

Although explosives for use at the KM are not transported using US 189, the staff did not determine any information in PSAR that would exclude transportation of explosives using US 189 for blasting at other mines and construction sites. As US 189 is the nearest highway, the overpressure generated from accidental detonation of the explosives while on transit on this highway would be bounding for other highways near the proposed site.

As Ammonium Nitrate Fuel Oil (ANFO) is the most commonly used explosive in mine and construction blasts, the staff assumed that 50,000 pounds (lb) (22679 kilograms (kg)) of ANFO, the maximum load carried by a highway truck, as per RG 1.91, would be transported on US 189 near the proposed site. The staff determined using the trinitrotoluene (TNT) Equivalence method that the safe distance at which an overpressure of 1 pounds per square inch (psi) (6.9 kilopascal (kPa)) would be generated from the 50,000 lb of ANFO blast is smaller than the distance of the safety-significant SSCs at the proposed facility from US 189. Therefore, the staff concludes transportation of commonly used explosive ANFO on US 189 would not be a credible hazard to the proposed facility. The staff also notes that detonation of 50,000 lb of ANFO while transiting US 189 is a conservative assumption as ammonium nitrate prill by itself does not burn easily. Safety regulations would not allow carrying ready-to-use ANFO in a truck on the highway. At the blasting site, ammonium nitrate prills would be mixed with fuel oil (FO) to manufacture ANFO. Each blasthole would be loaded with ANFO at a predetermined quantity. Primers (boosters) along with initiators (e.g., NONEL) would be attached to ANFO in each blasthole to conduct the blasts. Therefore, the staff concludes that 50,000 lb of ANFO detonating while on transit on US 189 is a conservative scenario and a potential explosion of ANFO while transiting US 189 would not pose a credible hazard to the proposed facility.

2.3.3.2 Railroads

PSAR section 2.3.2.5 states that the nearest railroad to the proposed site is approximately 0.15 mi (0.24 km) away and is operated by the Union Pacific Railroad, Rocky Mountain Division. Near the proposed site, coal is transported from the KM using this railroad. As sufficient mitigation measures are taken at the KM to prevent autoignition of coal and formation coal dust clouds, USO has not evaluated any accident scenarios on the Union Pacific Railroad near the proposed site.

The staff reviewed the discussion provided in PSAR section 2.3.2.5. As documented above in this SE, the staff's review of potential hazards posed by coal dust and autoignition of soil coal, determined that the mitigation measures taken at the KM are appropriate. Therefore, the staff determined that the coal while transported near the proposed site on rail cars does not pose a credible hazard to the proposed facility.

2.3.3.3 Pipelines

The staff reviewed USO's assessment of overpressure and thermal hazards from rupture of nearby pipelines in PSAR section 2.3.1.2.3 and table 2.3-12. In addition, the staff audited USO's internal documentation supporting the information in the PSAR. As stated in PSAR section 2.3.1.2.3, there are three buried natural gas pipelines within 5 mi (8 km) of the proposed

site that may have the potential to affect safe operations of the proposed facility. PSAR figure 2.3-2 shows their locations. In PSAR section 2.3.3.1, USO selected the Ignacio-Sumas Loop pipeline for assessing both overpressure and thermal radiation hazards to the proposed facility from a hypothetical guillotine rupture. The staff determined that the 24-in diameter Ignacio-Sumas Loop pipeline is located closest to the RXB center point, as shown in PSAR figure 2.3-2. Additionally, this pipeline operates at the highest pressure of 850 psi (5,861 kPa) and has the largest diameter of all the nearby pipelines. Therefore, the staff determined that the Ignacio-Sumas Loop pipeline would bound the hazards to the proposed facility from a hypothetical guillotine rupture of the nearby pipelines.

USO analyzed the potential overpressure and thermal radiation from an explosion of the natural gas vapor cloud resulting from a hypothetical rupture of a pipeline. USO used the ALOHA program to assess the potential overpressure and thermal radiation on the safety-significant SSCs at the proposed facility from a complete (guillotine) rupture of the Ignacio-Sumas Loop pipeline. In this scenario, natural gas will be released from both ends of the rupture. The flow out of the ruptured zone would be double the flow through the pipeline before the rupture.

USO estimated the distance from the release point (rupture zone of the pipeline assumed closest to the proposed facility) to the point where the concentration of the released natural gas in the vapor cloud is equal to the LFL or LEL (same limit for both methane⁴ and natural gas). Alternatively, per the guidance in RG 1.91, the distance between the rupture zone and the point where the overpressure generated by the exploding vapor cloud is 1 psi (6.9 kPa) can be used as the 'safe distance.' If the distance of the proposed facility is more than this safe distance, all structures would be safe from explosion overpressure hazard per RG 1.91. In addition, potential damages to the safety-significant SSCs need not be analyzed for any explosion-related missile impacts if the overpressure is 1 psi (6.9 kPa) or less.

USO assumed a smooth inner surface of the Ignacio-Sumas Loop pipeline. As internal roughness of the pipeline will induce more turbulence reducing the flow out of the ruptured zone. USO also assumed that the unbroken end of the pipeline is connected to a very large capacity reservoir of natural gas. This assumption therefore credits no action taken by the Control Room operator to detect the rupture and isolate the ruptured zone of the pipeline. The staff determined that USO's assumptions are conservative.

The staff notes that a rupture of a high pressure natural gas pipeline rapidly releases the gas through the break of the pipeline in the form of a jet. In case of a buried pipeline, the rupture of the pipeline may eject the overburden and form a crater. The consequences from a guillotine break of a natural gas pipeline can be divided into (Sluder et al., 2022):

- Missile generation
- Flash fire, in case of delayed ignition
- Jet fire, in case of immediate ignition

⁴ Methane is the primary component of natural gas, making up approximately 95 percent of its composition.

- Overpressure from explosion.

USO did not consider the potential consequences from a missile generated from a hypothetical rupture of the pipeline. The staff determined that the potential rupture points of the pipeline are too far away from the proposed site for the potential missiles to be a credible hazard.

In the case of delayed ignition, a flash fire results from relatively slow (less than the speed of sound) burning (or, deflagration) of the released natural gas plume in open air. A flash fire may burn back to the source (the ruptured section of the pipeline) becoming a jet fire. The overpressure generated from a flash fire of natural gas is insignificant because it is occurring in an unconfined (open) space. USO did not analyze the consequences of a flash fire as the consequences of a jet fire would be bounding. The staff determined it is acceptable to not analyze the consequences of the flash fire after rupture of the Ignacio Sumas Loop pipeline because a jet fire has more severe consequences.

A jet fire results from rapid release of natural gas through the ruptured area of the pipeline in the form of a momentum jet. Sparks generated by the fragments of the ruptured pipe and/or rock particles ejected can potentially ignite the released natural gas and start a jet fire. Within a very short time after ignition, the jet fire reaches its full intensity. The heat flux density decreases as soon as the pipeline isolation valves are closed as the flow of natural gas declines. Fire will continue until the gas released from the pipeline is consumed.

A vapor cloud explosion (VCE) occurs when a cloud of flammable gas ignites, and the flame speed accelerates to sufficiently high velocities to develop significant overpressure. The conditions necessary for a VCE of the released natural gas to occur are:

1. The ignition must be delayed so that the natural gas-air cloud can form an ignitable mixture with concentration within the flammable range of natural gas, i.e., between 5 percent (the lower flammability limit or LFL) and 15 percent (the upper flammability limit or UFL).
2. Availability of confinement with significant congestion in the flame propagation path where the flammable natural gas is released.
3. The ignition source must have sufficient energy to ignite the natural gas-air mixture.

Release of natural gas from a transmission pipeline, if ignited, typically results in a rapid burning fire or deflagration, rather than a detonation or explosion. Peak overpressure developed in a deflagration event is lower than that in a similar detonation event.

The staff notes that the availability of an ignition source with sufficient energy to initiate a direct detonation of the released natural gas cloud is extremely rare. Ignition energy needed to deflagrate methane-air mixture is in the order of 10^{-7} Btu (10^{-4} J); however, direct initiation of detonation requires approximately 105 Btu (108 J) of energy, an increase of 12 orders of magnitude. Such concentration of energy may be available from high explosives; however, common ignition sources, such as sparks from ejected pipeline fragments or electrical apparatus, hot steam lines, open furnaces, heaters, and moving parts in machinery, do not generally possess such energy concentration. Therefore, direct initiation of detonation of natural gas vapor cloud even near the release point is an extremely unlikely phenomenon.

A review of major vapor cloud incidents in the world (Atkins et al., 2017) did not identify any historical records of VCEs of liquefied natural gas or methane in open areas with sufficient severity to cause damage. The Pipeline and Hazardous Materials Safety Administration (PHMSA) pipeline accident investigators confirmed this in response to an inquiry by the NRC Expert Evaluation Team (ML20100F635). In addition, the NRC Expert Evaluation Team did not find, “any record of dense methane gas clouds. igniting or exploding at a location away from the initial pipe rupture.” Based on the preceding discussion, the staff determined that assessment of the potential consequences of only jet fire and overpressure from VCE events after a hypothetical rupture of the Ignacio-Sumas Loop pipeline is appropriate.

USO conducted an assessment of the overpressure generated from an explosion of the released gas and thermal radiation from the jet fire from a rupture of the Ignacio-Sumas Loop. Distance to the 1-psi (6.9 kPa) overpressure limit from immediate and delayed ignitions of the released gas and 5 kW/m² thermal load are given in PSAR table 2.3-12. The results for the double flow scenarios (flow from both ends of a ruptured pipeline) show that the distance to the LFL (or LEL) is significantly smaller than the distance of the proposed facility from the assumed ruptured point of the Ignacio-Sumas Loop pipeline. Therefore, USO concluded that overpressure generated from released natural gas from the Ignacio-Sumas Loop pipeline would not be a credible hazard to the proposed facility and the staff determined this conclusion was acceptable based on the discussion above.

USO also reported in PSAR table 2.3-12 that a potential jet fire of the natural gas released from the Ignacio-Sumas Loop pipeline would produce a threshold heat load of 5 kW/m² to a distance of 1,641 feet (ft) (500 meters (m)) from the ruptured point. The staff has independently estimated the distance of the double flow scenario using the ALOHA program. The distance to the threshold heat load is smaller than the distance to the nearest safety-significant SSCs at the proposed facility. Therefore, the staff determined that potential jet fires at the closest Ignacio-Sumas Loop pipeline would not pose a thermal hazard to the proposed facility.

The staff's review determined that the analyses presented by USO has several conservatisms. For example, based on staff's experience in the review of transmission pipelines, a rupture of a natural gas transmission pipeline is an extremely rare event because this event generally would be preceded by a detectable leak. The analysis assumes a worst-case release scenario from the pipeline which bounds any consequences of leaks. Additionally, the atmosphere at the proposed site is typically not calm as assumed in the analysis. Although the TNT Equivalency method was used to estimate the 1-psi (6.9 kPa) overpressure distance, the staff determined that it is a conservative scenario as high explosives would be needed to initiate and sustain the flame front at the detonation velocity. In summary, the Ignacio-Sumas Loop pipeline is 2.44 mi (3.9 km) away from the proposed site. USO used the ALOHA computer program to estimate the LFL distance, the 1-psi (6.9 kPa) distance, and distance to the thermal threshold of 5 kW/m². Based on the results given in PSAR table 2.3-12 and its review, the staff determined that the safety-significant SSCs at the proposed facility would not be affected by a potential release of natural gas from the nearby pipelines.

2.3.4 Hazards from Aircraft Operations

In PSAR section 2.3.1.2.4, USO identified the EMM, located approximately 8 mi (12.9 km) north of the proposed site. USO did not identify any other civilian or military airports within 10 mi (16.1 km) radius of the proposed site. In addition, PSAR section 2.3.1.2.4 has not identified any

Federal airway within 2 mi (3.2 km) and any military training routes within 5 mi (8 km) of the proposed site. In addition, no holding patterns were identified near the proposed site in the Visual Flight Rules sectional chart for the Salt Lake City.

USO used the Federal Aviation Administration (FAA) projected operational information for the EMM up to 2050 using historical data from 1990 through 2022. In addition, USO used the average rate of change from 2022 through 2050 to estimate the number of operations at the EMM in 2051 through 2090.

USO used the guidance given in section 3.5.1.6 of NUREG-0800 to estimate the annual crash frequency of aircraft flying in the vicinity of the proposed site. As the annual number of aircraft operations (includes landing and taking off) at EMM projected at 2090 did not exceed 16,000 (500 times square of the distance of 8 mi (12.9 km)) annually, based on Screening Criterion A of section 3.5.1.6 of NUREG-0800, USO concluded that aircraft operations at EMM would not pose a credible hazard to the proposed facility and the proposed site has satisfied Criterion A. As USO did not identify any Federal airways or military training routes near the proposed facility, both criteria B and C are also satisfied. Consequently, USO concluded that the annual frequency of aircraft crashes on to the proposed facility would be less than an order of magnitude of 10^{-7} per year by inspection and, therefore, aircraft crash is not a credible hazard of the proposed site.

The staff reviewed the information and assessment presented in PSAR sections 2.3.1.2.4, 2.3.2.6, and 2.3.3.1.6. The staff determined that EMM is approximately 8 mi (12.9 km) north of the proposed site. This airport is owned by the City of Kemmerer and has three runways based on AirNav.com:

1. Runway 16/34: in good condition and 8,203 ft (2500 m) long with asphalt surface
2. Runway 10/28: in fair condition and 3,271 ft (997 m) long with grass surface
3. Runway 04/22: in fair condition and 2,671 ft (814 m) long with concrete surface.

A flight landing on Runway 16 (landing from south) will bring the aircraft over the proposed site.

Based on the data from the FAA (https://www.faa.gov/air_traffic/flight_info/aeronav/aero_data/), this airport is mostly used by general aviation aircraft. From 7/1/2021 to 6/30/2022, a total of 3,158 operations took place at this airport, out of which 3,100 operations (98 percent) are by general aviation aircraft. Only 12 operations are by air taxis and 42 operations are by military aircraft. No air carrier operations took place at this airport in this time period.

As the combined number of operations at this airport is small compared to Criterion A of section 3.5.1.6 (3,158 annual flights compared to the threshold of 32,000) the staff determined that flights at this airport, even those using Runway 16 and thereby coming close to the proposed site, will not pose a credible crash hazard. In addition, the staff determined that the Victor Airway V4 is several miles north of EMM. As a result, the airway is much further away than 2 statute miles (3.2 km) from the site and does not pose a credible crash hazard to the proposed site, as per Criterion C of section 3.5.16 of NUREG-0800. The staff did not identify any military training routes near the proposed site. Therefore, based on the above discussion the staff determined aircraft flying in the vicinity would not be a credible hazard and need not be accounted for in designing the safety-significant SSCs at the proposed facility.

2.3.5 Hazards from Chemical Storage and Transport Explosion of Flammable Chemicals Stored Off-site or On-site and in Transit on Nearby Highways

The staff reviewed PSAR section 2.3.3.1.4 and the results presented in PSAR tables 2.3-9, 2.3-10, 2.3-11, 2.3-13, and 2.3-14. In addition, the staff audited USO's internal documentation on the details of flammability and explosion hazards assessment of the chemicals stored in nearby facilities and on-site including while in transit on US 189.

USO used the TNT Equivalency method, given in RG 1.91, to estimate the overpressure generated from immediate ignition of the released chemicals and used the 1 psi (6.9 kPa) criterion to assess whether the generated overpressure could damage the safety-significant SSCs at the proposed facility. USO used the ALOHA software, developed jointly by the National Oceanic and Atmospheric Administration (NOAA) and Environmental Protection Agency (EPA), to calculate peak overpressure from an explosion of the vapor cloud (delayed ignition) assuming sufficient congestion and confinement would be available of its propagation path and used the 1 psi (6.9 kPa) criterion to assess damage potential.

For analyzing a boiling liquid expanding vapor explosion (BLEVE) event, USO used a threshold of 5 kW/m² for thermal radiation limit and calculated the safe distance when the thermal flux equals the selected thermal criterion. USO used the same thermal hazard criterion for heat load from a jet fire. The staff notes that a person may get burn injuries if exposed to a 2.5 kW/m² heat load with prolonged exposure. More than twice of this level of thermal radiation is necessary to ignite wood with flame or melt plastic tubes.

Results of a screening analysis of the chemicals stored in nearby facilities are given in PSAR table 2.3-2 identifies the chemicals that are stored at nearby facilities and identifies the flammable materials and their storage locations and capacities:

1. Acetylene at KM: 6 cylinders, each with a capacity of 300 ft³ (8.5 m³)
2. Airline System Antifreeze at KM: 156 gallons (gal) (590 Liters (L))
3. Unleaded gasoline at KM: 10,000 gal (37,854 L)
4. Unleaded gasoline at PNPP: 29,190 lb (13,240 kg)
5. Propane at KM: 100 gal (379 L)
6. Propane at PNPP: 4,300 lb (1,950 kg)
7. Ammonium hydroxide at PNPP: 2,064 lb (936 kg).

Chemicals to be stored on-site of the storage facility (PSAR table 2.3-7) include

1. 6,480 ft³ (183 m³) of hydrogen from a coupled hydrogen rack and
2. 5,000 gal (18,927 L) of ammonium hydroxide.

In addition, hazards associated with rupture of a natural gas pipeline on the proposed facility are evaluated in PSAR section 2.3.3.1.4.

2.3.5.1 Acetylene

Acetylene is stored as a gas at the KM in six independent cylinders, each having a capacity of 300 ft³ (8.5 m³). USO assumed that only one cylinder would be used at a time. Following 40 CFR 68.25, "Worst-case release scenario analysis," USO also assumed that acetylene

would escape from the cylinder at a rate of 30 ft³/minute (min) (0.85 m³/min) over a 10 min period. ALOHA results determined that the distance to the Lower Explosive Limit (LEL) is significantly smaller than the distance to the safety-significant SSCs of the proposed facility. Therefore, even if an explosion occurs, it would be far away from the safety-significant SSCs. The staff independently repeated the calculation assuming all six cylinders release acetylene simultaneously. This is an extremely rare possibility as all cylinders have to fail simultaneously and release gas from an event such as common cause failure. The staff determined that even with six cylinders releasing simultaneously, the safety-significant SSCs at the proposed facility remain safe from the overpressure hazard.

2.3.5.2 Antifreeze

USO modeled the KM Airline System Antifreeze as 100 percent methanol using the ALOHA program. The staff determined this a reasonable approach because the Airline System Antifreeze is 98 percent methanol. As the Airline System Antifreeze is stored in a standard size vessel, USO assumed that the same vessel size would be used for transportation. The Airline System Antifreeze is expected to form a pool near the release point from the tank as it is stored as a liquid. Methanol will evaporate from the pool as time progresses and form a vapor cloud. ALOHA results show that explosion of the vapor cloud will not occur if the atmosphere is unstable, given by atmospheric stability classes A through E. If the atmosphere is very stable and can be characterized as stability Class F, an explosion may occur; however, both distances to the LFL and the 1-psi (6.9-kPa) overpressure limit are small compared to the distance to the safety-significant SSCs at the proposed facility. Therefore, USO concluded an accidental rupture of the vessel containing the Airline System Antifreeze in storage at the KM or while on transit on US 189 would not produce a credible hazard to the proposed facility. The staff determined that USO's evaluation of this hazard is acceptable because the concentration of the Airline System Antifreeze will be below the LFL at the proposed site location, and therefore cannot be ignited to explode. Additionally, the overpressure at the proposed site would be too low to be a hazard from an explosion of the released vapor cloud at KM or while on transit on US 189.

2.3.5.3 Ammonium Hydroxide

Ammonium hydroxide is stored at the PNPP and also on-site in the Turbine Facility Building (TFB) as a liquid in standard size vessels. USO assumed that same quantity of ammonium hydroxide would be involved while in transit through US 189. As it is stored as liquid, a release will form a pool near the release point from the vessel. Ammonium hydroxide from the pool will evaporate as time progresses to form a vapor cloud.

The vapor cloud formed after a release of ammonium hydroxide at the PNPP or in transit on US 189 will not explode when the atmosphere is unstable due to rapid mixing with air. Explosion of released ammonium hydroxide while in transit on US 189 will only occur if the atmospheric is stable. However, ALOHA results show that the LEL distance and the safe distance to 1-psi (6.9 kPa) overpressure limit are small compared to the distance of the safety-significant SSCs at the proposed facility.

In its analysis, USO assumed a pool would form if ammonium hydroxide was released from the container due to a leak or a rupture. Ammonium hydroxide would evaporate from the pool and form a vapor cloud. Immediate ignition of the released ammonium hydroxide would result in

detonation generating overpressure. The calculated safe distance to the 1-psi (6.9-kPa) overpressure limit is significantly smaller than the distance to the safety-significant SSCs from storage locations of ammonium hydroxide.

ALOHA results show that a delayed ignition of the vapor cloud formed from evaporation of the pooled material will only generate an explosion (deflagration) in more stable atmosphere; however, distance to the safety-significant SSCs is larger than the distance at which 1-psi (6.9-kPa) overpressure is estimated to occur.

Therefore, USO concluded that ammonium hydroxide while in storage at PNPP or on-site or while in transit will not be a credible hazard to the proposed facility. The staff determined that USO's evaluation of this hazard is acceptable because the overpressure developed by both immediate and delayed ignition would be below the threshold of 1 psi (6.9 kPa) due to large distance to the proposed site.

2.3.5.4 Gasoline

Gasoline is stored at PNPP as a liquid at ambient pressure and temperature. Gasoline is not a pure chemical; rather, a solution of several hydrocarbons. As the current version of the ALOHA program cannot model gasoline as a solution, USO modeled gasoline as n-Heptane using the ALOHA computer program. The staff determined that using n-Heptane as surrogate for gasoline in ALOHA analysis is reasonable as it is a common practice. The staff notes that it may be conservative to use n-Heptane instead of gasoline as n-Heptane evaporates at a faster rate forming the vapor cloud than gasoline below 80° C because many components of gasoline boil only at temperatures higher than 80° C. The boiling temperature of gasoline is a range from 50° C to 190° C.

In its analysis USO assumed that the total 29,190 lb (13,240 kg) of gasoline stored at PNPP and 10,000 gal (37,854 L) at KM is spilled in an event which forms pools near the release points. USO assumed that gasoline is transported in trucks of 9,000 gal (34,068 L) capacity (equivalent to 50,000 lb (22,679 kg) capacity of highway trucks on single axle, as per RG 1.91) through US 189.

Results using the ALOHA program show that, for all combinations of atmospheric conditions assumed in the assessment, the distances to the LFL and 1-psi (6.9 kPa) overpressure limit are significantly smaller than the distances of the safety-significant SSCs at the proposed facility from either PNPP or KM. USO also analyzed the overpressure generated from release of 9,000 gal (13,240 kg) of gasoline while transiting the point closest to the proposed facility on the nearest highway US 189. Distances to the LFL or the 1-psi (6.9 kPa) limit are smaller than the distances of the safety-significant SSCs from highway US 189.

Based on the above discussion, the staff determined that transportation of gasoline on nearby highways or storage at the nearby facilities would not create any credible hazard to the proposed facility.

2.3.5.5 Hydrogen

At the TFB on-site, USO stated that 6,680 standard cubic foot (scf) (189 standard cubic meter (Sm³)) or 34.5 lb (15.6 kg) of hydrogen would be stored as gas in two coupled racks, each rack

storing 12 bottles and each bottle containing 270 scf (7.6 Sm³) of hydrogen. USO assumed that release of hydrogen will take place over a period of 10 min if the coupled rack fails, following 40 CFR 68.25 (e)(1). Results, audited by the staff, using the ALOHA program showed that the distance to the safety-significant SSCs is greater than the LFL distance. Based on its review of USO's analysis and further confirmation during the audit, the staff determined that accidental release of hydrogen stored on-site would not pose a credible overpressure hazard to the safety-significant SSCs at the proposed facility.

2.3.5.6 Propane

The staff reviewed USO's calculation of the distance to the 1-psi (6.8 kPa) overpressure and 5 kW/m² thermal radiation limits from a hypothetical explosion of the entire 6,000 gal (22,712 L) of propane while in transport on US 189 in PSAR section 2.3.3.1.4 and table 2.3-11. The distance to the 1-psi (6.8 kPa) limit was calculated using the TNT Equivalence method. The calculated distances of 2,761 ft (841 m) for overpressure and 1,443 ft (440 m) for thermal radiation exceeds the distance between US 189 and the proposed facility, which is 1,035 ft (315 m). Therefore, there is a potential for damage of the safety-significant SSCs at the proposed facility when propane is transported on US 189 under a stable atmospheric condition (stability classes E and F).

USO investigated the allowable annual frequency of propane shipments being transported on US 189 without exceeding the overpressure and thermal radiation hazard limits. Based on discussions with the PNPP and KM, USO stated that propane is currently delivered to the PNPP once a month and the KM once a year, a total of 13 shipments in a year. During the audit, USO stated this assumption will be verified during the OL stage.

USO assumed that propane in a tanker is HM Class 3 liquid and used Battelle (2001) "Comparative Risks of Hazardous Materials and Non-Hazardous Materials Truck Shipment Accidents/Incidents," for assessing the potential hazards associated with propane transport on US 189. Table 24 of Battelle (2001) gives 4.96414×10^{-7} accidents/mile of travel of HM Class 3 liquids. In addition, table 9 of Battelle (2001) shows that 1,379 HM Class 3 accidents took place in 1997 while the cargo was enroute. Additionally, table 10 of Battelle (2001) shows that out of these 1,379 HM Class 3 accidents, only 50 and 22 accidents resulted in either a fire or an explosion, respectively. Therefore, only 5.22 percent of the accidents of HM Class 3 cargo on highways would result in a fire or an explosion. In addition, table 11.4 of Federal Emergency Management Agency (FEMA) (1989), "Handbook of Chemical Hazard Analysis Procedures," shows that approximately 20 percent of all spills resulted in entire cargo loss. Therefore, only 1.04 percent (= 5.22 percent * 20 percent) of the 1,379 accidents released all liquids from the tanker and resulted in either a fire or an explosion. Using RG 1.91, the exposure distance of US 189 would be approximately 1 mi (1.6 km) within which an accident of the propane tanker truck releasing all cargo would produce a credible overpressure hazard to the proposed facility. Consequently, USO estimated that 19 trips of propane tanker trucks would be permissible annually using Equation (6) of RG 1.91 if the threshold hazard is 10^{-7} per year. If the threshold hazard is set at 10^{-6} per year, 194 trips by propane tankers would be acceptable.

USO stated that the estimated annual allowable trips for propane tankers on US 189 near the proposed facility is conservative and that most of the safety-significant SSCs would be below ground level. The only above ground safety-significant SSCs will be the stacks of the Reactor Air Cooling System. The staff agrees that SSCs below ground level would have significant

protection against explosion pressure waves propagating on the surface; however, as one component of a safety-significant SSCs might be affected, overpressure and thermal radiation from propane tanker accident would be a credible hazard to the proposed facility if the annual number of shipments exceeds the threshold values.

Although USO has accounted for the shipments to the nearby KM and PNPP, propane shipments to other facilities far away can use US 189. In addition, propane is also used for domestic consumption and in farms. Therefore, the margin available before propane transport using US 189 may be considered a potential hazard may not be large enough. Therefore, the staff conducted an independent assessment of the hazard posed by propane transport through US 189.

The staff determined that propane is classified as HM Class 2, Division 2.1 (Flammable Gas) as per 49 CFR 173.115, “Class 2, Divisions 2.1, 2.2, and 2.3—Definitions.” Propane is colorless gas shipped as a liquefied gas (liquefied petroleum gas) under its vapor pressure. Therefore, the staff determined that the statistics used from Battelle (2001) for HM Class 3 liquids are not relevant for assessing hazards from transport of propane through highways. Table 24 of Battelle (2001) shows 3.42784×10^{-7} accidents/mile of travel of HM Class 2.1. In addition, table 9 of Battelle (2001) 276 HM Class 2.1 accidents took place in 1997 while the cargo was enroute. Additionally, table 10 of Battelle (2001) shows that only 7 fires and 2 explosions resulted in those 276 accidents. Therefore, only 3.3 percent accidents of HM Class 2.1 resulted in either a fire or explosion. Taking only 20 percent of all spills resulted in total loss of the cargo following FEMA (1989), only 0.65 percent of 276 HM Class 2.1 accidents released the entire load and resulted in a fire or an explosion. The staff determined that 45 trips by propane tanker trucks can be made annually before it will be considered a credible hazard to the proposed facility if the threshold hazard is set at 10^{-7} per year. Assuming the threshold hazard is 10^{-6} per year, 449 trips by propane tanker trucks can use US 189 near the proposed facility before it becomes a credible hazard.

Additionally, the staff determined that although the approach used by USO to estimate the annual crash rate of propane tanker truck on highway US 189 is reasonable, the statistics used may be out of date (based on 1997 Commodity Flow Survey of the Bureau of the Census). The staff independently estimated the annual crash rate of propane tankers using “Large” truck accident statistics, published yearly by the Federal Motor Carrier Safety Administration (FMCSA) (<https://www.fmcsa.dot.gov/safety/data-and-statistics/large-truck-and-bus-crash-facts>). The staff used statistics from 2016 through 2021 in the estimation as the reporting requirements of large truck crashes have changed in 2016.

FMCSA defines “Large” trucks as trucks with a gross vehicle weight greater than 10,000 lb (4,536 kg); however, any vehicle carrying HM requiring a HM placard is considered a large truck irrespective of the truck weight. Therefore, the staff notes that a 6,000 lb (2,722 kg) propane tanker would be classified as a large truck in the FMCSA’s databases.

The staff used the single truck crash statistics for propane tanker truck crashes. Based on FMCSA Trends table 19 the staff determined that large “single trucks” traveled a total of 724,354 million miles in 2016 through 2021. FMCSA Vehicles table 11 tabulates the number of fatal and non-fatal crashes involving HM Class 2 (all types of gases) resulting in either a release or no release of the HM for each year. In 2016 through 2021, 29 fatal large truck crashes transporting HM Class 2 cargo released the cargo whereas no releases took place in

16 fatal crashes. In these years, 60 non-fatal crashes released Class 2 HM being transported. No Class 2 HM was released in 400 non-fatal crashes. It was unknown whether a release took place in 44 non-fatal crashes.

To be conservative, the staff assumed that there was a release even if it was not certain (i.e., all unknowns are assumed that a release took place). Therefore, in 2016 through 2021, a total of 133 large truck crashes released Class 2 HM whereas 416 crashes did not. Therefore, a HM truck carrying Class 2 material would be involved in 1.836×10^{-10} accidents per mile of travel per year.

USO estimated the exposure distance on US 189 for a propane explosion to be approximately 1 mi (1.6 km), following RG 1.91. The exposure distance for thermal radiation would be smaller and, therefore, assessment of overpressure would be bounding. The staff estimates that approximately 1.836×10^{-10} releases would occur per year near the proposed facility from propane tanker trucks transiting US 189 and experiencing accidents. The staff assumed that the entire 6,000 gal (2,722 kg) of propane is released in every accident and all HM Class 2 accidents were involved with HM Class 2, Division 2.1 (Flammable Gas), which is conservative. Following similar steps, a threshold annual hazard of 10^{-7} would allow 545 trips by propane tanker trucks on US 189 near the proposed facility in a year. However, a hazard threshold of 10^{-6} per year would increase the annual number of tanker trips to 5,447 before it becomes a credible hazard.

Based on above discussion, the staff determined that significantly more trips on US 189 in a year need to take place by propane tanker trucks before propane transport becomes a credible hazard to the proposed facility. Therefore, the staff determined that propane transport using US 189 does not pose a credible hazard to the proposed facility.

2.3.6 Toxicity Hazards

The staff reviewed PSAR section 2.3.3.1.5 and analysis results presented in PSAR tables 2.3-15 through 2.3-18. In addition, the staff audited USO's internal documentation supporting the information in the PSAR. PSAR section 2.3.3.1.5 describes the toxicity hazards from an accidental spill of hazardous chemicals both off-site and on-site of the proposed facility on habitability of the Main Control Room (MCR). USO used the ALOHA computer program to assess a spill of a specific chemical would create a toxicity hazard at the proposed facility.

The ALOHA program estimates three parameters for each scenario:

1. Distance from the hazard source (i.e., spill location) to the point where the toxicity reaches the limit, either the immediately dangerous to life or health (IDLH) or the asphyxiation limit.
2. Maximum concentration of the spilled chemical in the atmosphere at the outdoor of the MCR.
3. Maximum concentration of the spilled chemical in the atmosphere inside the MCR assuming MCR has an air change rate (ACR) of 1.2; i.e., 1.2 times the Control Room air by volume is replaced by atmospheric ambient air every hour.

The staff determined that the assumed ACR of 1.2 for the MCR facility is conservative following RG 1.78 as the MCR is assumed to possess an inefficient leakage control without any isolation

capabilities. The staff expects that the actual MCR will have a lower ACR and the assumed conditions would bound the performance of the actual MCR during operation.

An asphyxiation concentration limit is used for chemicals which have no established IDLH values. A neutrally buoyant chemical concentration of 67,000 ppm is taken as the level at which an oxygen-deficient environment (19.5 percent of oxygen in air in contrast to normal 21 percent) would develop in the MCR because of the released chemical. As per 29 CFR 1910.134(d), an oxygen-deficient atmosphere shall be considered as IDLH. Similarly, if oxygen is spilled near the MCR, an oxygen-enriched atmosphere would develop in the MCR if the oxygen concentration in the confined space exceeds 23.5 percent, as per 29 CFR 1919.146(b). USO estimated a concentration of 31,600 ppm of released oxygen would develop an oxygen-enriched atmosphere inside the MCR.

The National Institute for Occupational Safety and Health (NIOSH) develops the IDLH values of different chemicals. NIOSH has not listed the IDLH value of sodium hypochlorite; however, sodium hypochlorite would generate toxic chlorine gas in contact with acid. Chlorine gas can irritate eyes and make breathing difficult. USO has taken the IDLH value of chlorine gas (10 ppm) as the IDLH value of sodium chlorite. The staff determined the use of the IDLH value of chlorine gas for sodium hypochlorite to be acceptable.

Within 10 mi (16.1 km) of the proposed site, there are four industrial facilities: (1) KM, (2) PacifiCorp Naughton Power Plant (PNPP), (3) Sodium Test and Fill Facility (TFF), and (4) Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant. The Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant is more than 5 mi (8 km) from the MCR. Therefore, the staff concludes, based on RG 1.78, that if a release occurs at the Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant, atmospheric dispersion will dilute and disperse the plume of hazardous chemicals to such a degree that either toxic limits will never be reached or there would be sufficient time for the MCR operators to take appropriate actions. However, KM, PNPP, and TFF are within 5 mi (8 km) of the MCR. Therefore, the staff reviewed the analysis of potential hazards for each chemical stored at these facilities.

As discussed previously, the nearest railroad to the proposed site is operated by Union Pacific. This line terminates at the crossing with US 30, approximately 0.5 miles (0.8 km) west of the City of Kemmerer. Coal is the only material transported on this line. As reviewed above by the staff, transportation of coal would not pose a credible hazard (including toxicity hazard) to the proposed facility. Therefore, the staff determined that hazardous chemicals transported on highways would be the only transportation sources to be evaluated for toxicity hazards. Although there are several other roads near the proposed site, US 189 is the nearest highway running along the west side. US 189 at its closest point to the MCR is approximately 1,350 ft (411 m) away. Therefore, analyses using the hazardous materials on US 189 would be bounding as USO assumed all chemicals would travel along US 189.

PSAR table 2.3-6 lists three natural gas (methane) pipelines within 5 mi (8 km) of the proposed site. As methane does have a toxicity limit such as the IDLH value or the time weighted average (TWA) 8-hour work shift limit, USO has screened out methane in the nearby pipelines for further consideration. The staff independently used ALOHA to assess the potential hazards to the habitability of the MCR in case of a rupture of the Ignacio-Sumas Loop pipeline carrying natural gas. To assess the potential toxicity of released methane, ALOHA software adopts the

Protective Action Criteria (PAC) values for methane to assess toxicity hazards: (1) PAC-1 equal to 65,000 ppm, (2) PAC-2 equal to 230,000 ppm, and (3) PAC-3 equal to 400,000 ppm. In ALOHA these values are Temporary Emergency Exposure Limits (TEELs) for methane and represent the concentration at which most people will begin to experience health effects when exposed to more than an hour. At a concentration of methane above PAC-1 (65,000 ppm) in air, general population including susceptible individuals could experience notable discomfort or irritation; however, these effects are not disabling and are reversible upon cessation of exposure. For a complete rupture of the Ignacio-Sumas Loop pipeline connected to an infinite source, ALOHA estimated that the methane concentration would be below the PAC-1 (65,000 ppm) concentration approximately 400 ft (122 m) from the rupture location, compared to the 12,890 ft (3,929 m) distance from the proposed facility to the nearest pipeline, as given in PSAR table 2.3-6. Therefore, the staff determined that methane in these pipelines would not pose a credible toxicity hazard to the habitability of the MCR.

The staff notes, as per RG 1.78, all scenarios considered for spillage of hazardous chemicals on-site, at off-site facilities, and during highway transport on US 189 by USO are from maximum concentration chemical accidents (MCAs), as opposed to average concentration duration chemical accidents (ACAs). ACAs produce continuous releases lasting for a long duration with low leakage rate, such as leakage from a valve. In contrast, MCAs result in instantaneous release or a short-term puff of a large quantity of hazardous chemicals. MCAs occurs from an outright failure of the container itself. The staff determined that USO considered the largest storage container of a hazardous chemical located on-site or at a nearby facility. Similarly, the largest shipping container to transport frequently the hazardous chemicals near the proposed site was considered by USO. In addition, the staff determined that following RG 1.78 USO appropriately analyzed scenarios with multiple containers of equal size assuming failure of just one container unless the containers were physically connected. USO assumed that none of the chemical storage tanks are connected. Therefore, the largest vessel quantity/volume was analyzed for each chemical.

In addition, USO assumed that the capacity of the gasoline tanker trucks on US 189 to be 9,000 gal (34,069 L) and propane tanker trucks to be 6,000 gal (22,712 L). The capacity of the propane tanker truck is based on permitted maximum fill volume of 88 percent for propane. The staff determined this assumption is acceptable as USO used the maximum highway truck capacity of 50,000 lb (22,690 kg), given in RG 1.91, to derive the volume capacity of the tanker truck (rounded up).

As discussed previously, USO assumed that the entire quantity of gasoline is in Heptane to assess the habitability of the MCR from a spill of the entire capacity of a gasoline tanker transiting US 189 using ALOHA. To assess the toxicity hazard, the IDLH value of n Heptane equal to 750 ppm was used in the ALOHA analysis. The staff determined that substituting Heptane for gasoline is acceptable as it is a common practice and as gasoline is not a pure chemical; rather a solution of several hydrocarbons. Therefore, the staff concludes that the assumption that the entire content of the gasoline tank in a tanker would be spilled in an accident and forms a vapor cloud, as modeled by USO using the ALOHA program, is conservative.

USO assumed in its analysis that the maximum quantity of ammonium hydroxide transported using US 189 is the total quantity that PNPP can store on-site. Ammonium hydroxide is only stored at PNPP and the chemical is transported to PNPP using US 189 (PSAR section 2.3.3.1.5

and PSAR tables 2.3-1, 2.3-2, and 2.3-4). Based on PSAR table 2.3-4, the maximum quantity of ammonium hydroxide transported via US 189 is 2,064 lb (936 kg).

PNPP is located approximately 21,384 ft (6518 m) away from the MCR and stores six hazardous chemicals, based on PSAR table 2.3-1:

1. Ammonium hydroxide: 2,064 lb (936 kg), as liquid solution,
2. Unleaded gasoline: 29,190 lb (13,240 kg), as liquid,
3. Hydrochloric acid (concentration $\geq$ 35 percent): 541 lb (245 kg), as liquid,
4. Hydrochloric acid (concentration $\leq$ 35 percent): 2,450 lb (1,111 kg), as liquid,
5. Natural gas (methane), as gas (through FL91 3-inch and FL91 4-inch pipelines)
6. Propane 4,300 lb (1,950 kg), as liquefied gas.

KM is located approximately 16,421 ft from the MCR, as given in PSAR table 2.3-1. It stores seven hazardous chemicals in the site (PSAR table 2.3-2):

1. Acetylene: 6 cylinders each with 300 ft³ (8.5 m³) capacity, as gas,
2. Airline system antifreeze: 156 gal (591 L), as liquid,
3. Carbon dioxide mixed with argon: 6 cylinders with 300 ft³ (8.5 m³) capacity, as liquefied gas,
4. Unleaded gasoline: 10,000 gal (37,854 L), as liquid,
5. Nitrogen: 6 cylinders each with volume 300 ft³ (8.5 m³), as liquefied gas,
6. Oxygen: 3 cylinders each with 200 ft³ (5.7 m³) capacity and 6 cylinders with 300 ft³ (8.5 m³) capacity, as liquefied gas, and
7. Propane: 4,300 lb (1,950 kg), as liquefied gas.

As stated in PSAR section 2.3.3.1.5, ALOHA calculations performed by USO concluded that the maximum distance to the toxicity limits was not exceeded for any chemicals stored at the PNPP and KM. Consequently, the maximum chemical concentration outdoor and indoor of the MCR would not be exceeded for these chemicals.

As described in PSAR table 2.3-3, argon and nitrogen are stored in the TFF in different forms.

1. Argon is stored in a 3,000 gal (11,356 L) cryogenic tank as liquid, 400,000 gal (1,514,165 L) as gas in atmospheric tank, 13,000 scf (368 Sm³) in one tank, and 45,000 gal (170,344 L) in a pressure vessel;
2. Nitrogen is stored as gas in a pressure vessel with 30,000 gal (113,562 L) capacity and in a 3,000 gal (11,356 L) cryogenic tank.

USO's analysis as provided in PSAR table 2.3-15 concluded that only a spill of nitrogen from the cryogenic storage tank would exceed the toxicity limit at the MCR. In addition, although the outdoor concentration calculated exceeded the limit as asphyxiant (more than 67,400 ppm), the concentration inside the MCR is below the asphyxiation limit. The staff determined USO's conclusion that none of the hazardous chemicals stored in any of the off-site facilities would produce a credible hazard to the MCR is acceptable.

PSAR table 2.3-7 lists seven chemicals that would be stored on-site at the proposed facility. The Bulk Plant Gases Building would store 1,456 gal (5,512 L) of argon as liquefied gas and 220,110 scf (6233 Sm³) of nitrogen as gas. The Turbine Storage Building would store

5,170 scf (146 Sm³) of carbon dioxide as liquefied gas and 3,120 scf (88 Sm³) of hydrogen as gas. The Water Treatment Plant would have 4,770 gal (18,056 L) of sodium hypochlorite (as chlorine gas). In addition, the TFB stores 110 gal (416 L) of ammonium hydroxide and 265 gal (1,003 L) of hydrazine. PSAR table 2.3-18 gives the results of outdoor toxicity hazard assessment of these chemicals.

USO's analysis provided in PSAR section 2.3.3.1.5 and in PSAR table 2.3-18 determined a spill of ammonium hydroxide, argon, nitrogen, and hydrazine would exceed the minimum toxicity distance and the maximum outdoor concentration would be above the IDLH or asphyxiation limits; however, a spill of only ammonium hydroxide and hydrazine would exceed the toxicity limits indoor at the MCR. In PSAR section 2.3.3.1.5, USO committed to continually sample fresh air intake of the MCR using toxic samplers. PSAR section 2.3.3.1.5 states that the HVAC system to be installed at the Nuclear Island (NI) of the proposed facility will be equipped with sensors to continually sample the MCR fresh air intake. These sensors will warn the MCR operators to take precautionary measures.

The staff notes, following RG 1.78, that most on-site chlorine releases in nuclear power plants have been ACAs resulting in a leakage rate from near zero to less than 1 lb (0.5 kg) of chlorine per second. Given warning, the MCR operators would need only breathing apparatuses to be protected from ACAs.

As discussed in PSAR section 2.3.3.1.5, the release of ammonium hydroxide, propane, and hydrochloric acid being transported US 189 would exceed the maximum outdoor concentration of the MCR; however, the indoor concentration of these chemicals does not exceed the toxicity limits.

In summary, the staff reviewed USO's use of the ALOHA software to assess the potential to pose a toxicity hazard from each chemical to the habitability of the MCR. As the additional information on the analyses provided during the audit used the worst-case scenario or conditions to assess the hazard, the results would likely be conservative. The estimated concentration of each hazardous chemical is compared with the established limits at which individuals could be exposed for 0.5 to 1 hour without serious health concerns. The analyses assumed the worst-case weather conditions: an atmospheric stability of F with wind speed of 1.5 mph (2.4 km/h). The assumed stable atmosphere, results in prolonged periods before the hazardous chemical would be diluted and as a result exposes the MCR to the highest concentration of the chemical spilled. In addition, the analyses assumed that the entire quantity of chemicals in the container is released in the catastrophic event. The probability or possible cause(s) of such a release is not considered. Therefore, based on the preceding discussion, the staff determined that USO's toxicity level estimated from release of each chemical is conservative and acceptable.

2.3.7 Fire Hazards

USO analyzed the thermal flux developed from a rupture of a natural gas pipeline in PSAR section 2.3.3.1.4. In addition, USO discussed the fire hazards from wildfires at the proposed site in PSAR section 2.3.3.1.7. Information on historical wildfires is given in PSAR 2.4.1.3.6 and PSAR tables 2.4-92.

PSAR table 2.4-92 lists 24 wildfires in Lincoln County, Wyoming, during the period from January of 1996 to December 31, 2023. As Lincoln County is much larger than the area 5-mile (8-km) radius from the proposed site, USO considered this a conservative number of wildfires. The return interval of fires in the sagebrush steppe environment of the site is estimated to be from 35 to 125 years, based on information from the Department of Ecosystem Science & Management of the University of Wyoming, as stated in PSAR section 2.4.3.1.6. USO, based on information from the University of Wyoming, does not consider wildfires to be an important factor in the region surrounding the site.

The staff determined that the region surrounding the proposed facility, especially the area surrounding the NI area, is predominantly covered with shrub-scrub vegetation. Additionally, areas close to the NI building will be paved reducing the potential for wildfires significantly. USO concluded in PSAR section 2.3.3.1.7 that the potential for brush fires or wildfires to spread from outside the site boundary into the NI area is minimal. The staff determined, based on its review of USO's analysis, that wildfires do not pose a significant hazard in the region surrounding the KU1 site. The staff identified the potential fire hazards to the safety-significant SSCs from the TFF and the Energy Island was not evaluated by USO. USO committed in PSAR section 2.3.3.1.7 to further evaluate potential fire hazards, including an analysis of the effects of fire originating from the TFF and Energy Island, which will be submitted with the OL application.

2.3.8 Conclusion

The staff evaluated the sufficiency of the proposed site characteristics regarding nearby industrial, transportation, and military facility descriptions, as described in PSAR section 2.3, using the guidance and acceptance criteria from section 2.3 of DANU-ISG-2022-02. The staff also used other NRC guidance documents, as needed and as described above, to supplement DANU-ISG-2022-02.

The staff determined that USO adequately described the facilities, installations, and transportation routes near the proposed site and evaluated the potential hazards posed by them to the proposed KU1 facility. The staff determined that USO has adequately evaluated the potential human-induced hazards from operations at the nearby airports, transportation routes, and facilities to establish the site parameters to ensure that the proposed reactor facility can accommodate the commonly occurring hazards (e.g., explosions, fires, and flammable and toxic vapor clouds). The staff notes that USO plans to install toxicity detectors in the ventilation system for the MCR due to the storage of hazardous chemicals at the proposed site. Based on its determinations above, the staff determined the information on nearby industrial, military, and transportation facilities is sufficient and meets the applicable guidance and regulatory requirements for the issuance of a CP in accordance with 10 CFR 50.35, "Issuance of construction permits," and 50.40, "Common standards."

2.4 Regional Climatology, Local Meteorology, and Atmospheric Dispersion

PSAR section 2.4 provides a description of the general climate of the region around the proposed KU1 site and meteorological conditions relevant to the design and operation of the KU1 facility. PSAR section 2.4 also provides data and information used to determine the atmospheric dispersion conditions in the vicinity of the site. This information includes local and regional airflow and meteorological measurements used for dispersion estimates.

In addition to the guidance provided at the beginning of this chapter, the following guidance are applicable to the evaluation of regional climatology, local meteorology, and atmospheric dispersion is:

- DC/COL-ISG-07, "Interim Staff Guidance [(ISG)] on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures," (ML091490565),
- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Rev. 1 (ML070350028),
- RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Rev. 3 (ML14107A411),
- RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Rev. 1 (ML070360253),
- RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Rev. 1 (ML003740354),
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants," Rev. 1 (ML003740205),
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Rev. 0 (ML031530505),
- RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Rev. 0 (ML110940300),

Other documents the staff considered during the review of PSAR section 2.4 are:

- NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations" (ML12045A149),
- NUREG/CR-2919, "XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations," (ML081360412),

2.4.1 Technical Evaluation

2.4.1.1 Regional Climatology

PSAR section 2.4.1 provides information regarding regional climatic conditions and the occurrence of meteorological phenomena (including both averages and extremes) that could potentially affect the design and operating bases of the Natrium structures, systems, and components at the KU1 site. The applicant provides detailed information about the climate region as it relates to the topics of the following seven subsections of this SE, which correspond to the topics of PSAR sections 2.4.1.2 through 2.4.1.7.

2.4.1.1.1 Data Sources

PSAR section 2.4.1.1 states that several sources of data are used to characterize local and regional climatological conditions pertinent to the KU1 site. Sources include meteorological stations within 50 mi (80 km) of the site to help ensure representativeness of the data for use at the site. PSAR table 2.4-1 identifies the meteorological stations used for all parameters discussed in PSAR section 2.4, the station type, distance from the site, and the parameters observed. PSAR figure 2.4-1 shows the station locations relative to the KU1 site. The staff is not making a determination on the applicability and the adequacy of the data discussed and summarized in PSAR section 2.4.1.1. These data sources are discussed later in the SE for the individual climatological or meteorological hazards for which they were collected.

2.4.1.1.2 General Climate

The applicant described the general climate of the KU1 site by discussing the terrain in the southwestern portion of Wyoming, as well as the general synoptic conditions historically observed. The applicant noted that the KU1 site falls under the Köppen climate classification of Dfb (warm-summer humid continental climate), which is in a region affected by both continental and mountainous influences. The applicant also described the effects of the elevation changes within the area that influences weather patterns near the site and in other portions of the state. The applicant described the region as having generally low humidity levels, especially during the summer months. This leads to the mostly dry condition in this semi-arid climate.

The staff has compared the applicant's general climate description to a similar National Climatic Data Center (NCDC) narrative description of the climate of Wyoming, as well as descriptions from the Wyoming Water Resources Data System and State Climate Office. The staff performed this comparison to determine if the applicant's description was consistent with descriptions from national and state sources. Based on this comparison, the staff determined that the applicant's general climate description is accurate and complete.

2.4.1.1.3 Severe Weather

PSAR section 2.4.1.3.1 discusses severe weather phenomena affecting the site and the surrounding region.

2.4.1.1.4 Thunderstorms, Hail, and Lightning

The following discussion on thunderstorms, hail, and lightning is intended to provide a general understanding of the severe weather phenomena in the proposed KU1 site region but does not result in the generation of site characteristics for use as design or operating bases.

USO used 30-years of data from Lincoln County, WY to examine the thunderstorm activity in the region of the proposed Kemmerer site. The data was collected for the years 1992 through 2022. USO states that Lincoln County, which includes the proposed KU1 site, only reported 22 thunderstorm wind events over the 30-year period from 1992-2022.

Using the same time period as thunderstorm wind events, hail events were found to have occurred 31 times over the 30-year period. PSAR table 2.4-10 provides additional information on the historical hail event data for Lincoln County.

The applicant estimated the number of lightning flashes per square kilometer per year, based on a method attributed to publication titled, "Lightning Protection," (1973, J. Marshall). The applicant estimated that approximately three lightning flashes per year occur in the site vicinity. The staff independently evaluated this estimate based on a Vaisala Xweather Interactive Global Lightning Density Map that shows the average number of lightning events per square kilometer per year. Using these same data sources, the staff independently confirmed that the Kemmerer region in southwestern Wyoming experiences two lightning events per square-km per year.

Therefore, the staff reviewed the applicant's description of thunderstorms, hail, and lightning in the region of the proposed KU1 site and, based on its review, including verification that the applicant obtained the information from reliable and verifiable sources near the site, finds that the applicant's assessment is sufficient for evaluation of potential thunderstorm, hail, and lightning impacts to inform design bases for the facility. Therefore, the staff finds that the applicant's assessment of thunderstorms, hail, and lightning is acceptable.

2.4.1.1.5 Extreme Winds

PSAR section 2.4.1.3.2 states that no hurricanes have been reported within 100-miles of the KU1 site based on data from the NOAA's Office for Coastal Management Historical Hurricane Tracks. The staff reviewed the NOAA data and determined that the applicant's statement is accurate, and that hurricanes do not pose a threat to the KU1 site.

Estimating wind loading on plant structures involves identifying the site's "basic" wind speed. The applicant defined the KU1 site "basic" wind speed by using American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 7-16, "Minimum Design Loads for Buildings and Other Structures." NRC guidance provided in NUREG-0800 section 2.3.1 identifies ASCE/SEI 7-05 as an appropriate standard for use in identifying climatological these hazards. The applicant used a more recent version of this standard that provides data consistent with the 2005 version. Although NUREG-0800 does not apply to non-LWRs, such as the proposed facility, nothing about the proposed facility would affect the applicability of this standard. Thus, the staff finds ASCE/SEI 7-16 acceptable for use in this application. The applicant defined the KU1 site "basic" wind speed using the ASCE/SEI 7-16 definition of "3-second gust speed at 33 ft (10 m) above the ground in Exposure Category C." PSAR section 2.4.1.3.2 states that the applicant determined a "basic" wind speed of 110 miles per hour (mph) and 115 mph wind speed used for wind loading conditions. These wind speeds represent the 3-second gust speed, Exposure Category C, for Risk Category III and IV buildings, respectively.

To assess whether Exposure Category C is an appropriate category for the KU1 site, the staff reviewed the ASCE/SEI 7-16 definition, which states that Exposure Category C relies on the surface roughness categories as defined in ASCE/SEI 7-16 chapter 2. ASCE/SEI 7-16 defines Exposure Category B as "urban and suburban areas, wooded areas, or other terrain with numerous closely spaced obstructions having the size of single-family dwellings or larger" prevailing "in the upwind direction for a distance of at least 2,600 ft (792 m) or 20 times the height of the building, whichever is greater." ASCE/SEI 7-16 defines Exposure Category D as

“flat, unobstructed areas and water surfaces” prevailing “in the upwind direction for a distance greater than 5,000 ft (1,525 m) or 20 times the building height, whichever is greater.” ASCE/SEI 7-16 states that Exposure Category C shall apply for all cases for which neither Exposure Category B nor D applies.

The staff reviewed the site description in PSAR sections 2.4.1.3.2 and 2.4.1.2 and determined that neither Exposure Category B nor Exposure Category D accurately describes the conditions at the KU1 site. Using the ASCE/SEI online hazard tool⁵, the staff reviewed the “basic” 110 and 115 mph wind speed used by the applicant. The staff reviewed the exposure categories as defined by ASCE/SEI 7-16 and determined that Exposure Category C is acceptable at the KU1 site because the KU1 site is inconsistent with the ASCE/SEI 7-16 definition of Exposure Category B and D and, therefore, consistent with the ASCE/SEI 7-16 definition of Exposure Category C. Because Exposure Category C is appropriate for use at the KU1 site and the staff verified the applicant’s wind speeds using the ASCE/SEI online hazard tool, the staff determined that it is acceptable that the applicant used the “basic” 110 and 115 mph wind speed used for wind loading conditions which is the 3-second gust speed, Exposure Category C, for Risk Category III and IV buildings, respectively.

2.4.1.1.6 Tornadoes

PSAR 2.4.1.3.3 states that the applicant used RG 1.76 to choose the tornado site characteristics. RG 1.76 provides design-basis tornado characteristics for three tornado-intensity regions throughout the U.S., each with an exceedance frequency of 10^7 per year. PSAR 2.4.1.3.3 states that the KU1 site is located within Tornado-Intensity Region III, as described in RG 1.76. The applicant’s proposed tornado site characteristics are listed in PSAR table 2.1 1. PSAR table 2.4-12 presents statistics from NUREG/CR-4461, “Tornado Climatology of the Contiguous United States,” Rev. 2 (ML070810400), on tornadoes that have occurred within 2 degrees (of longitude and latitude) surrounding the KU1 site.

The staff reviewed the applicant’s tornado site characteristics by using RG 1.76 and NUREG/CR-4461 to confirm that the tornado site characteristics described by the applicant are consistent with the KU1 site. Based on the staff’s review, the staff determined that the applicant’s description of the frequency of storms that have been recorded near the KU1 site is accurate, as presented in PSAR section 2.4.1.3.3.

Since the applicant’s tornado site characteristics are based on those presented in RG 1.76 and the staff determined that the applicant accurately used RG 1.76 to determine the KU1 tornado site characteristics, the staff finds that the applicant has chosen acceptable tornado site characteristics.

2.4.1.1.7 Tropical Cyclones and Hurricanes

As described above in SE section 2.4.1.1.5, PSAR section 2.4.1.3.2 states that no hurricanes have been reported within 100-mi of the KU1 site based on data from NOAA. Similarly, PSAR section 2.4.1.3.4 states there have been no known hurricanes or tropical cyclones observed in the site region. The staff reviewed the NOAA data and determined that the applicant’s

⁵ <https://ascehazardtool.org/>

statements are accurate, and that hurricanes and tropical cyclones do not pose a threat to the KU1 site.

2.4.1.1.8 Winter Precipitation

Freezing Rain and Ice Storms

PSAR section 2.4.1.3.5 states there have been zero freezing rain or ice storm events recorded at the site, as reported by the NOAA Storm Events Database during the period of January 1996 through December 31, 2023. Although the staff prefers 30 years of continuous data⁶ for climatological hazards, since freezing rain and ice storms do not result in a site parameter, and there are none reported, the staff finds 27 years to be acceptable. Using the NOAA Storm Events Database for Lincoln County, WY where the KU1 site is located, the staff reviewed the data and concluded that the applicant accurately represented freezing rain or ice storm events at the KU1 site in the PSAR.

Snowfall, Snow Depth, and Snow Loads

DC/COL-ISG 07 clarifies the staff's position on identifying winter precipitation events as site characteristics and site parameters for determining normal and extreme winter precipitation loads on the roofs of Seismic Category I structures. DC/COL-ISG 07 is applicable to the staff's review of applications proposing to meet the regulations in General Design Criterion (GDC) 2 in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities." Analogous to GDC 2, the applicant's applicable design criterion is PDC 2, as defined in PSAR section 5.3 and reviewed by the staff in section 5.3 of this SE. Although DC/COL-ISG-07 was written for Part 52 applicants, the underlying safety concern, ensuring Seismic Category I structures can withstand combined snow and precipitation loads, is equally relevant to Part 50 licensing, since both frameworks require demonstrating structural integrity under site-specific extreme environmental conditions.

The ISG states that normal and extreme winter precipitation events should be identified as site characteristics for determining the normal and extreme winter precipitation loads on the roofs of Seismic Category I structures. The normal winter precipitation roof load is a function of the normal winter precipitation event; whereas, the extreme winter precipitation roof loads are based on the weight of the antecedent snowpack resulting from the normal winter precipitation event plus the larger resultant weight from either: (1) the extreme frozen winter precipitation event; or (2) the extreme liquid winter precipitation event. The extreme frozen winter precipitation event is assumed to accumulate on the roof on top of the antecedent normal winter precipitation event; whereas, the extreme liquid winter precipitation event may or may not accumulate on the roof, depending on the geometry of the roof and the type of drainage provided. The ISG further states:

⁶ In general, the staff prefers to have a minimum of 30-years of continuous observation when characterizing climatological hazards. The World Meteorological Organization considers that 30 years is long enough to eliminate year to year variations for the purpose of obtaining an accurate average value and accounting for variability.

- The normal winter precipitation event should be the highest ground level weight (in pounds per square foot (psf)) among: (1) the 100-year return period snowpack; (2) the historical maximum snowpack; (3) the 100-year return period two day snowfall event; or (4) the historical maximum two-day snowfall event in the site region.
- The extreme frozen winter precipitation event should be the higher ground level weight (in lb/ft²) between: (1) the 100-year return period two-day snowfall event; and (2) the historical maximum two-day snowfall event in the site region.
- The extreme liquid winter precipitation event is defined as the theoretically greatest depth of precipitation for a 48-hour period that is physically possible over a 25.9-square-kilometer (10-square-mile) area at a particular geographical location during those months with the historically highest snowpack.

PSAR section 2.4.1.3.5 states that snowfall is common in the site area, but due to high winds that generally accompany snowstorms, accurate measurements of the snowfall can be difficult to obtain. Using the data sources listed on DC/COL-ISG-07 page 5, along with observations from local National Weather Service stations, the staff reviewed the data and determined that the applicant's assessment of winter precipitation events at the KU1 site is consistent with DC/COL-ISG-07 and, therefore, that the extreme winter precipitation load of 70.72 psf listed in PSAR table 2.1-1 accurately represents the KU1 site. Data from the National Weather Service are the most accurate representation of snowfall events in the area. Because the applicant's assessment of winter precipitation events is consistent with DC/COL-ISG-07, and used data from a variety of observation stations, the staff finds this discussion and resulting value to be acceptable.

2.4.1.1.9 Droughts, Dust Storms, and Wildfires

PSAR section 2.4.1.3.6 states there have been 24 wildfires in Lincoln County, WY during the period from January of 1996 to December 31, 2023. The staff, using the NCEI Storm Events Database, was able to confirm the number of wildfire events by searching for a period of 1950 through the end of 2024. The oldest record of a wildfire in Lincoln County, WY was in August of 2000. During this same time period there were no dust storms reported. This section of the PSAR does not result in any design-basis hazards. The staff accepts this data as correct and for informational purposes.

2.4.1.1.10 Meteorological Data for Evaluating the Ultimate Heat Sink

PSAR section 2.4.1.4 describes the design dry- and wet-bulb temperatures used for the Ultimate Heat Sink (UHS) at the KU1 site. The PSAR states that because there are no mechanical draft cooling towers with water storage ponds that will be used as the UHS for KU1, the only design dry- and wet-bulb temperatures identified are the 100-percent and maximum two percent annual exceedance and mean coincident wet-bulb temperatures. Using the American Society of Heating, Refrigeration, and Air Conditioning Engineers (ASHRAE) Weather Data Viewer, the applicant determined the 100-year return period values of maximum and minimum dry-bulb temperatures. The staff obtained hourly meteorological data from the same nearby meteorological station and ASHRAE. The staff independently confirmed the design dry- and wet-bulb temperatures used for the KU1 site UHS and determined that the values provided by the applicant accurately represent the KU1 site.

2.4.1.1.11 Design-Basis Dry- and Wet-Bulb Temperatures

PSAR section 2.4.1.5 provides the design-basis dry- and wet-bulb temperatures, which are the ambient temperature and humidity statistics to establish heat loads (e.g., for cooling systems) and the applicant provided the associated parameters in PSAR table 2.4-8.

The PSAR states that the applicant used meteorological data from the NCDC for the Evanston, Wyoming station to determine the parameters (i.e., extreme values) in PSAR table 2.4-8. The National Weather Service station at Evanston, WY, is the nearest automated observation station to the Kemmerer site. Using this data, which the staff independently obtained, the staff was able to confirm the accuracy of the design-basis dry and wet bulb temperatures provided in PSAR table 2.1-1. The PSAR also states that the applicant obtained data for the years of 1987-1990, 1992-1999, and 2004-2021 and used this data to calculate the various exceedance temperatures. The staff obtained the hourly meteorological dataset from the same source and notes that the dataset is not continuous, as described in the previous sentence. This data set is the most complete the staff was able to obtain, and it matches the dates provided by the applicant. The results of the applicant's ambient design temperature analyses are presented in PSAR tables 2.4 7 and 2.4 8. PSAR section 2.4.2.1.5 states that, in addition to surface data from the Evanston station, the applicant used the Automated Surface Observing System station from the "[ASHRAE], Weather Data Viewer 2021."

The staff reviewed the information on dry- and wet-bulb temperatures in the region of the proposed KU1 site and, based on its review, including verification that the applicant obtained the information from sources listed as references in NUREG-0800 section 2.3.1 (i.e., the ASHRAE Weather Data Viewer, NOAA), and is reasonable for the KU1 site geographic area, the staff finds that the applicant's assessment sufficient for evaluation of expected conditions at the KU1 site to inform design bases for the facility. Therefore, the staff finds that the applicant's assessment of design basis dry- and wet-bulb temperatures is acceptable.

2.4.1.1.12 Restrictive Dispersion Conditions

PSAR section 2.4.1.6 discusses the topography surrounding the KU1 site and how it impacts the dispersion at the site for the potential of ground level releases. The PSAR states that the Kemmerer site is in a valley that is surrounded by hills and mountains and that dispersion at the site is considered poor. Using topographic maps of Wyoming, specifically the area around the KU1 site, the staff was able to confirm the applicant's description of the local topography. The staff also confirmed, using the atmospheric stability data in PSAR table 2.4-68, that in general there are significantly more hours of restrictive dispersion conditions (stability classes E, F, G) than there are unstable conditions (stability classes A, B, C). Therefore, the staff finds that TerraPower's assessment of restrictive dispersion conditions is acceptable.

2.4.1.1.13 Long-Term Climate Trends

To comply with NRC regulations 10 CFR 100.20 and 100.21 and meet PDC 2, the Sodium reactor must, among other things, be built with consideration, in part, of the most severe natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have

been accumulated.⁷ Long-term climate trends are a concern because of the potential for unforeseen changes in extreme conditions in the local and regional environment. PSAR section 2.4.1.7 provides a discussion of the climatology of the KU1 region with regard to the trends in normal daily mean, maximum, and minimum temperatures and normal maximum precipitation (rainfall).

In PSAR section 2.4.1.7, the applicant analyzed trends in temperature over a 30-year period from 1981 to 2010. These trends are presented in PSAR tables 2.4-74 through 2.4-76 for the KU1 site, Naughton Power Plant, and three other observation stations near the KU1 site.

To address long-term climate trends, the historical data used to characterize a site (and discussed in SE sections 2.4.1.3.1.3 through 2.4.1.3.5) should extend over a significant time interval to capture cyclical extremes. To perform its review, the staff obtained data sets considered to be of sufficient duration to determine the adequacy of the applicant's proposed site characteristics. For example, snow load was evaluated using a 100-year return period and ambient design temperatures were based on a minimum of 30 years of hourly data and an estimated 100-year return period value.

The U.S. Global Change Research Program (USGCRP) released the fifth National Climate Assessment (NCA5) report to the President and Members of Congress in November 2023. This report, produced by an advisory committee chartered under the Federal Advisory Committee Act, summarizes the science of long-term climate trends and the potential impacts of these trends on the U.S. NCA5 was thoroughly reviewed by Federal Government experts, external experts, and the public multiple times throughout the report development process. The expert external review was performed by an ad hoc committee of the National Academies of Sciences, Engineering, and Medicine.

The staff performed an independent review of historical data to characterize the KU1 site over a significant time interval to capture cyclical extremes. Based on its review, the staff concluded that the applicant's assessment of long-term climate trends in PSAR section 2.4.1.7 accurately represents the KU1 site.

2.4.1.2 Local Meteorology and Topography

2.4.1.2.1 Normal and Extreme Values of Meteorological Parameters

2.4.1.2.1.1 Comparative Study and Description of Meteorological Measurements Program

Two meteorological towers collected data near the KU1 site, the Naughton Power Plant meteorological tower and the KU1 meteorological tower. PSAR section 2.3.2.1.2 states that the Naughton Power Plant is located 3.1 mi northwest of the KU1 site and PSAR section 2.4.2.1.2 states that the Naughton Power Plant meteorological tower is in close proximity to the KU1 meteorological tower. The applicant conducted a study to compare meteorological data from the Naughton Power Plant and KU1 meteorological towers for the time period from April 9, 2022, to April 8, 2023. This was to assess whether the data from the

⁷ The staff reviewed the acceptability of PDC 2 and found it acceptable in ML24283A066 and the applicability of this determination is discussed in Chapter 5 of this SE.

Naughton Power Plant meteorological tower is representative of conditions at the KU1 site. The applicant also performed an evaluation of the data from the Naughton Power Plant meteorological tower during 2019–2021 and concluded that the data is representative of the region including the KU1 site. The applicant's study used scatter plots and correlation coefficients, considering 0.7 or higher as representative. Correlation coefficients exceeded 0.7 for all examined variables, with the lowest at 0.7006 for 10-meter wind direction (due to low wind speeds) and the highest at 0.9966 for 50- and 60-meter temperatures. Given the strong correlations, the applicant concluded that data from the Naughton Power Plant meteorological tower accurately represents conditions at KU1 and can supplement or replace its data. The applicant provided results from this study in different parts of PSAR section 2.4 as well as in audit material. Those results are discussed in the relevant sections of this SE. For the reasons stated in those below discussions as well as the proximity of the Naughton Power Plant meteorological tower to the KU1 site, the general consistency of the results, and the quality of the meteorological data, the staff finds the use of the Naughton Power Plant meteorological tower provides a reasonable representation of the meteorological conditions at the KU1 site and that use of the Naughton Power Plant meteorological data is acceptable.

The applicant stated, in an audit response, that the data collected and used for the atmospheric dispersion factors is part of the pre-application period. The applicant stated it intends to maintain the KU1 meteorological program at the site during the operational phase. The applicant also provided an audit response, and updated the PSAR to describe both the Naughton Power Plant and KU1 meteorological towers. PSAR tables 2.4-90-a, and 2.4-90-b describe the specifications and accuracies of each of the instruments on both towers.

2.4.1.2.1.2 Data Sources

Two meteorological towers collected data at the KU1 site, the Naughton Power Plant meteorological tower and the KU1 meteorological tower. As discussed above, the data from the KU1 meteorological tower was used by the applicant to show that the three-year dataset from the Naughton Power Plant meteorological tower, which is used in the atmospheric dispersion analysis, is representative of the area.

PSAR section 2.4.2.1.1 states that the Naughton Power Plant meteorological tower is a 50-meter tower that collects wind speed, wind direction, and temperature at the 10-meter and 50-meter levels. Data from the years 2019 – 2021 was used to characterize the onsite local meteorology and in the atmospheric dispersion analysis. RG 1.23 states that a minimum 12-month period should be provided for a CP. The submission of a consecutive 3-year period from the Naughton Power Plant is therefore acceptable to the staff.

2.4.1.2.1.3 Average Wind Direction

PSAR section 2.4.2.1.2 states that PSAR “[f]igure 2.4-8 and [f]igure 2.4-9 present 3-year composite wind rose plots of the Naughton Power Plant data for 2019, 2020, and 2021 using the wind speed and direction at the 33-foot (10-meter) level and the 164-foot (50-meter) level, respectively.” The staff reviewed the meteorological dataset and concluded that the PSAR accurately represents the annual wind speed, wind direction summaries, and wind roses.

To illustrate the representativeness of the Naughton Power Plant data and the KU1 data to the surrounding area, a study was completed which compared the meteorological data collected at the Naughton Power Plant and KU1 meteorological towers for the period from April 9, 2022, to

April 8, 2023. The applicant provided data from Big Piney Marbleton Airport in Big Piney, WY. Big Piney Airport is located 80 miles to the north-northeast of the site and is at a similar elevation to the Kemmerer site. PSAR table 2.4-17 compares wind speed by month for the KU1, Naughton Power Plant, and Big Piney, WY. Since the Kemmerer and Naughton towers are located in close proximity the values are comparable, as expected. Annual data from the KU1 meteorological tower, at the 10 m level, shows the most common wind direction from the north northwest, with secondary maximums from the west through northwest (clockwise). Data from the Naughton Power Plant tower for the same period from 2022 – 2023, which was used to compare data collected over the same time period, shows wind predominantly from the west north-west sector with a secondary maximum of wind from the north. The purpose of the comparison study was to evaluate whether the Naughton Power Plant data was representative of the conditions experienced at KU1. An evaluation of the 2019, 2020, and 2021 Naughton Power Plant meteorological data showed that the Naughton Power Plant data was representative of the region, including KU1.

The applicant stated in PSAR section 2.4.2.1.7 that the results of this study confirm that “the Naughton data provides a good representation for [the Naughton Power Plant and the KU1 site] and can be used to accurately represent the meteorological conditions at [KU1].”

2.4.1.2.1.4 Atmospheric Stability

PSAR section 2.4.2.1.4 states that the applicant classified atmospheric stability in accordance with the guidance provided in RG 1.23. Atmospheric stability is a critical parameter for estimating dispersion characteristics at the KU1 site. Dispersion of effluents is greatest for extremely unstable atmospheric conditions (i.e., Pasquill Stability Class A) and decreases progressively through extremely stable conditions (i.e., Pasquill Stability Class G). The applicant based its stability classification on temperature change with height (i.e., delta-temperature or $\Delta T/\Delta Z$) between the 50 m and 10 m height, as measured by the Naughton Power Plant meteorological tower during 2019, 2020, and 2021. The staff reviewed how the applicant calculated atmospheric stability and determined that the applicant calculated it consistent with the guidance in RG 1.23.

Frequency of occurrence for each stability class is one of the inputs to the dispersion models used in PSAR section 2.4.4. The applicant included these data in the form of a joint frequency distribution (JFD) of wind speed and direction data as a function of stability class. The staff developed a JFD from the hourly data submitted by the applicant. The staff compared the JFD it developed with the JFD developed by the applicant and the comparison showed reasonable agreement. Stability class comparisons between the staff and applicant were within 2 percent for F and G stability, which is more important for dispersion estimates. Since the applicant and staff results were close, the staff concludes that the applicant prepared the JFD correctly.

2.4.1.2.1.5 Wind Direction Persistence

The applicant presented wind persistence data from the KU1 meteorological monitoring program, as described in PSAR section 2.4.2.1.3, during 2022-2023. Wind persistence is an indicator of the duration of atmospheric transport from a specific sector to a corresponding downwind sector that is 180° opposite. The applicant provided detailed information on the wind persistence that was observed by the onsite meteorological measurements in PSAR tables 2.4-18 and 2.4-19. The staff, using internal tools to analyze site-specific data submitted to

the agency, independently reviewed wind persistence using data from the Naughton Power Plant monitoring system from 2019, 2020, and 2021, and determined that the data is reasonable for the location by achieving comparable results. Therefore, the staff determined that the applicant's data in PSAR tables 2.4-18 and 2.4-19, and the text in PSAR section 2.4.2.1.3 are acceptable.

2.4.1.3 Air Quality

In PSAR sections 2.4.3, 2.4.3.1, and 2.4.3.2, the applicant discussed the current regional air quality and well as the projected air quality as a result of the operation of the KU1 facility. The applicant stated that the proposed plant systems are not sources of criteria pollutants, and the supporting equipment is operated on an intermittent basis. Therefore, the emissions from the plant would not impact ambient air quality levels in the vicinity of the site. Air quality is not included as a safety related design parameter. Therefore, the staff accepts the applicants text in PSAR section 2.4.3 for informational purposes.

2.4.1.4 Atmospheric Dispersion

The short-term diffusion estimates are used to determine the amount of airborne radioactive materials expected to reach a specific location during an accident situation. The diffusion estimates should address the guidance in RG 1.249, "Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors," Rev. 0 (ML22024A241), for conservative atmospheric dispersion (relative concentration) factor (χ/Q value) estimates at the EAB, the outer boundary of the Low Population Zone (LPZ), and at the control room for postulated design basis accidental radioactive airborne releases. The review covers the following specific areas: (1) atmospheric dispersion models to calculate atmospheric dispersion factors for postulated accidental radioactive releases; (2) meteorological data and other assumptions used as input to atmospheric dispersion models; (3) derivation of diffusion parameters (e.g., sigma-Y [σ_y] and sigma-Z [σ_z]); (4) cumulative frequency distributions of χ/Q values; and (5) determination of conservative χ/Q values used to assess the consequences of postulated design basis atmospheric radioactive releases to the EAB, LPZ, and control room.

2.4.1.4.1 Dispersion Estimates

PSAR section 2.4.4.1.1 describes the methods used by the applicant to determine the offsite atmospheric dispersion factors using the ARCON computer code. To determine the offsite χ/Q values, the applicant followed the guidance provided in RG 1.249. The applicant used ARCON (NUREG/CR 6331, "Atmospheric Relative Concentrations in Building Wakes," Rev. 1 (ML17213A190)) to estimate χ/Q values at the control room for potential accidental releases of radioactive material. The ARCON model implements the methodology outlined in RG 1.194.

2.4.1.4.2 Short-Term Atmospheric Dispersion Estimates

ARCON assesses ground level, vent, and elevated releases. RG 1.249 outlines how ARCON is used to determine accident-related offsite atmospheric dispersion factors. This guidance provides procedures for applying the ARCON code to estimate χ/Q values at the EAB and the outer boundary of the LPZ up to 3,937 ft (1,200 m) from the nearest edge of a building within the NI. Given the distance limitations in RG 1.249, ground level releases are assumed. Appendix A

to RG 1.249 provides assumptions and methods that are acceptable to the staff for deriving the 99.5th-percentile χ/Q values from ARCON standard output files.

PSAR table 2.4-78 presents the applicant's 99.5th percentile χ/Q values at the average male breathing height for the onsite control room evaluation. PSAR table 2.4-79 provides ARCON's 99.5th-percentile χ/Q values for each of the 16 downwind sectors over the relevant time periods at the EAB distance of 400 m, covering averaging times from 1 to 720 hours. The staff, using the Naughton Power Plant meteorological data and the methods provided in RG 1.249, performed confirmatory calculations that resulted in comparable 99.5th percentile χ/Q values as those presented in the PSAR.

2.4.1.4.3 Long-Term Atmospheric Dispersion Estimates

The applicant used the NRC-sponsored computer code XOQDOQ (described in NUREG/CR.2919) to estimate χ/Q and D/Q values resulting from routine releases. The XOQDOQ model implements the constant mean wind direction methodology outlined in RG 1.111.

The XOQDOQ model is a straight-line Gaussian plume model based on the theoretical assumption that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. In predictions of χ/Q and D/Q values for long time periods (i.e., annual averages), horizontal distribution of the plume is assumed to be evenly distributed within the downwind direction sector (e.g., "sector averaging"). A straight-line trajectory is assumed between the release point and all receptors. Terrain affects the movement of air in and around the nuclear installation. XOQDOQ provides an option to correct for recirculation for open terrain which should be used unless site specific data is available for recirculation or data indicates that other factors are appropriate. The applicant stated that the KU1 site is located on a flat, open terrain, so this recirculation correction is appropriate for data points that are beyond the EAB. For receptors onsite, the presence of buildings between the release point and the receptor means the terrain is not open. For this reason, the applicant performed multiple XOQDOQ model runs to examine both scenarios.

PSAR 2.4.4.2 describes that the distance to the receptors of interest (i.e., milk cow, milk goat, garden, meat animal, and resident). Receptors of interest included in the calculation are:

- EAB
- Nearest dairy cow
- Nearest meat cattle
- Nearest vegetable garden
- Nearest residence
- Proposed Particulate/Iodine Sampling Locations in support of the Radiological Environmental Monitoring Program (REMP)

- Test and Fill Facility (TTF)

The applicant modeled one ground level release point beginning at the energy island (EI). Two release points were identified, the NI and the EI. For releases from the EI, specifically for the fuel handling building (FHB), an assumption of a minimum building vertical cross-sectional area of 20 × 20 m for a total cross section of 400 m² was used due to the location of a stack. A ground level release, as defined by RG 1.111, is a conservative assumption at a site such as KU1 resulting in higher χ/Q and D/Q values when compared to a mixed mode (e.g., parttime ground, parttime elevated) release or a 100 percent elevated release. A ground level release assumption is, therefore, acceptable to the staff. The applicant stated that some receptors are assumed to be in the same place. The nearest dairy animal and vegetable garden are assumed to be located at the nearest residence and the nearest meat animal is assumed to be at the EAB. The nearest residence is located 2.8 mi to the northeast but is modeled in all 16 directions to find the most conservative location. The use of the shortest distance results in higher (more conservative) χ/Q values for ground level releases. Therefore, the staff reviewed the assumptions presented by the applicant and, because the assumptions are conservative, concluded that they are acceptable.

PSAR table 2.4-81 lists the long-term atmospheric dispersion and deposition estimates for the EAB, and special receptors of interest that the applicant derived from their XOQDOQ modeling results. PSAR tables 2.4-82 through 2.4-87 also describe the applicant's long-term atmospheric dispersion and deposition estimates for 16 radial sectors from the site boundary to a distance of 50mi from the proposed facility.

The χ/Q values presented in PSAR tables 2.4-81 through 2.4-87 reflect several plume radioactive decay and deposition estimates for the EAB and special receptors of interest that the applicant derived from its XOQDOQ modeling results.

The χ/Q values presented in the PSAR reflect several plume radioactive decay and deposition scenarios. Section C.3 of RG 1.111 states that radioactive decay and dry deposition should be considered in radiological impact evaluations of potential annual radiation doses to the public, resulting from routine releases of radioactive materials in gaseous effluents. Section C.3.a of RG 1.111 states that an overall half-life of 2.26 days is acceptable for evaluating the radioactive decay of short-lived noble gases and an overall half-life of 8 days is acceptable for evaluating the radioactive decay for all iodines released to the atmosphere. Definitions for the χ/Q categories are as follows:

- Undepleted/No Decay χ/Q values are χ/Q values used to evaluate ground level concentrations of long lived noble gases, tritium, and carbon 14. The plume is assumed to travel downwind, without undergoing dry deposition of radioactive decay.
- Undepleted/2.26 Day Decay χ/Q values are χ/Q values used to evaluate ground level concentrations of short lived noble gases. The plume is assumed to travel downwind, without undergoing dry deposition, but is decayed, assuming a half life of 2.26 days, based on the half life of xenon 133.
- Depleted/8.00 Day Decay χ/Q values are χ/Q values used to evaluate ground level concentrations of radioiodine and particulates. The plume is assumed to travel

downwind, with dry deposition, and is decayed assuming a half life of 8.00 days, based on the half life of iodine 131.

Using the information provided by the applicant, the staff reviewed the applicant's χ/Q and D/Q values by running the XOQDOQ computer code and obtained comparable results. The JFDs used by the applicant for the long-term diffusion estimates consisted of 11 wind speed categories. Based on the results of the staff's analysis, the staff determined that the long term χ/Q and D/Q values presented by the applicant are acceptable.

2.4.1.4.4 Control Room Dispersion Estimates

The applicant used ARCON to estimate χ/Q values at the control room for potential accidental releases of radioactive material. The ARCON model implements the methodology outlined in RG 1.194.

The meteorological input to ARCON used by the applicant consisted of wind speed, wind direction, and atmospheric stability data based on hourly data from a 3-year period from 2019, 2020, and 2021 from the Naughton Power Plant. The wind data were obtained from the 10 m and 50 m levels of the onsite meteorological tower, and the stability data were derived from the vertical temperature difference (delta temperature) measurements taken between the 50 m and 10 m levels on the onsite meteorological tower.

The staff completed a detailed review related to the acceptability and representativeness of the hourly meteorological data. Based on its review, the staff considers the onsite meteorological database suitable for input to the ARCON model.

The diffusion coefficients used in ARCON and incorporated by the applicant have three components. The first component is the diffusion coefficient used in other NRC models such as PAVAN. The other two components are corrections to account for enhanced dispersion under low wind speed conditions and in building wakes. These components are based on analysis of diffusion data collected in various building wake diffusion experiments under a wide range of meteorological conditions. Because the diffusion occurs at short distances within the plant's building complex, the ARCON diffusion parameters are not affected by nearby topographic features such as bodies of water. Therefore, the staff finds the applicant's use of the ARCON diffusion parameter assumptions acceptable.

One receptor (i.e., air intake) point, the Nuclear Control Building, was modeled for the following three release points:

- Reactor Building (RXB)
- Reactor Auxiliary Building (RXB)
- FHB

The staff reviewed the applicant's atmospheric dispersion estimates by running the ARCON computer model and obtaining consistent results. Both the staff and applicant used a ground level release assumption for each of the release/receptor combinations as well as other

conservative assumptions. Based on its confirmatory analysis, the staff determined that the applicant's control room χ/Q values are acceptable.

Using site layout maps (PSAR figures 2.4-66 through 2.4-76) to determine the orientation of source and receptor pairs, as well as the distance between each pair, the staff reviewed the input values provided by the applicant. The applicant provided, through the audit, the distances and directions between the receptor and source as well as the building dimensions for input into ARCON. The staff concludes the applicant's input values are acceptable.

2.4.2 Conclusion

Based on its findings above, the staff concludes the information in PSAR section 2.4 satisfies the relevant regulatory requirements identified above in this SE and PDC 2 for the issuance of a CP in accordance with 10 CFR Part 50.

The staff evaluated the meteorology for the proposed KU1 site, including the description of the general climate of the region and local meteorological conditions relevant to the design and operation of the facility, as described in PSAR section 2.4. The staff reviewed how that information was used in assessing the general climate of the region and meteorological conditions relevant to the design and operation of the facility. In addition, the staff reviewed the applicant's atmospheric dispersion analysis for the site.

In summary, the staff finds that the site characteristics associated with meteorology, including general and local climatology, are acceptable and reasonably representative of the region of the proposed KU1 site. The staff finds that the data resources (e.g., meteorological data sources) the applicant used to prepare PSAR section 2.4 are appropriate. The staff finds, based on the review of the application, as well as the results of the staff's confirmatory analyses, that the applicant's analyses of meteorological hazards and atmospheric dispersion are sufficient and acceptable. The staff finds that the proposed KU1 site is not located where catastrophic meteorological events are likely, that the applicant considered credible meteorological events in developing the design basis parameters for the proposed facility, and that the applicant provided an adequate description of site characteristics needed to evaluate the potential uncontrolled release of radioactive materials.

Based on its review, the staff finds that the level of detail and analyses provided on KU1 site meteorology demonstrates an adequate design basis and satisfies the applicable regulations 10 CFR 50.34(a)(1)(ii), 10 CFR 100.20, and 10 CFR 100.21, allowing the staff to find that:

- The meteorological history and projections for the proposed site have been prepared in an acceptable form.
- These projections have been factored into the choice of facility location, as well as the design, sufficiently to provide assurance that no weather-related event is likely to cause damage to the reactor facility during its lifetime that could result in the uncontrolled release of radioactive material to the unrestricted area.
- The meteorological information is sufficient for analyses applicable to and commensurate with the risks of the dispersion of accidental airborne releases of radioactive material in the unrestricted environment at the proposed site.

2.5 Hydrological Description

This section of the SE describes the staff's review and evaluation of the PSAR section 2.5, which provides the site hydrological characteristics and summarize the design basis for the site characteristics and the values selected for the design of safety-significant SSCs and the analysis of the transport of radioactive material resulting from postulated spills or leaks of liquid waste. The staff reviewed the applicant's site-specific design basis hydrologic hazard levels, including bounding design basis flood level and local intense precipitation (LIP) flood as well as groundwater site characteristics, including maximum groundwater level, and travel time and pathways of radionuclide transport of accidental releases, as listed in PSAR tables 2.1-1 and 2.1-2, respectively.

The applicable regulatory requirements related to the hydrological description of the site are as follows:

- 10 CFR 20.1301, "Dose limits for individual members of the public,"
- 10 CFR 50.34(a)(1)(ii),
- 10 CFR 100.20(c), and
- 10 CFR 100.21(d).

The applicable guidance for the evaluation of the hydrological description of the site is:

- DANU-ISG-2022-02, "Chapter 2, 'Site Information,'" (ML23277A140),
- RG 1.59, "Design Basis Floods for Nuclear Power Plants," Rev. 2 (ML003740388),
- RG 1.102, "Flood Protection for Nuclear Power Plants," Rev. 1 (ML003740308),
- Draft Regulatory Guide (DG) DG-1290, "Design-Basis Floods for Nuclear Power Plants," Rev 3 (ML19289E561),

The following PDC, as defined in PSAR section 5.3, applies to the hydrological description of the site:

- PDC 2, "Design Bases for Protection Against Natural Phenomena"

2.5.1 Technical Evaluation

2.5.1.1 Floods

The staff reviewed the sufficiency and acceptability of the applicant's description of the KU1 site characteristics regarding hydrology to meet the requirements of 10 CFR 50.34(a)(1)(ii), 10 CFR 100.20(c), and 10 CFR 100.21(d), as presented in PSAR section 2.5. The staff relied on the guidance and acceptance criteria from DANU-ISG-2022-02 chapter 2.

2.5.1.1.1 Hydrologic Description

The staff conducted a review of the hydrologic description and the screening of potential flood hazards at the plant site, as presented in PSAR section 2.5 for the proposed KU1 site. The applicant provided an overview of the hydrologic setting and a preliminary evaluation of potential flooding hazards relevant to the proposed plant. PSAR section 2.5.1.5 specifies that the NI site grade elevation as 6,756 ft National Vertical Datum of 1988 (NAVD 88), corresponding to mean sea level (MSL). PSAR section 2.5.1.1.2 states that safety-significant structures are designed with entrance floor elevations of 6,758 ft MSL, ensuring adequate protection from potential flooding.

The staff reviewed the site-specific hydrologic descriptions, and the screening of flood hazards provided in PSAR section 2.5. The staff reviewed the applicant's descriptions and screening analyses of applicable flood mechanisms for the site and determined that they met the acceptance criteria described in DANU-ISG-2022-02 section 2.5.3.2. Therefore, the staff determined that the hydrologic descriptions and preliminary screening of flood hazards at the plant site are adequately addressed in the PSAR.

2.5.1.1.2 Flooding from Local Intense Precipitation

In PSAR section 2.5.1, the applicant evaluated potential onsite flooding from extreme rainstorms, called LIP events. The applicant used computer simulations to model how rainwater would flow and drain from the site during a postulated LIP event.

The staff performed confirmatory reviews of the applicant's LIP analysis and determined that the applicant's analysis was conservative in terms of the following:

- The assumed 6-hour storm duration is appropriate for this basin size.
- The applicant's Probable Maximum Precipitation (PMP) estimate is slightly higher than the NRC's independent calculation (10.7 inches vs. 9.85 inches), but this difference is acceptable.
- The adjustments for long-term trends are reasonable and provide extra safety margin.

The staff notes that the difference in PMP values between the applicant and the staff, primarily resulting from manual interpretation of sparse PMP contour lines in Hydrometeorological Report (HMR) HMR-47, falls within the uncertainty range of PMP estimates for the region and is therefore deemed acceptable. The staff also reviewed the Hydrologic Engineering Center's River Analysis System (HEC-RAS) LIP flood simulation and confirmed it accurately modeled water flow on the site. The resolution of topography, grid size, boundary conditions, model configurations, and facility layout in the model is adequate as the applicant has followed the guidelines provided by the HEC-RAS User Manuals (United States Army Corps of Engineers (USACE), 2024). Use of the assumptions, such as no infiltration and 100 percent runoff, were conservative and followed guidelines provided by NUREG/CR-7046, "Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America," (ML11321A195).

In summary, the applicant's analysis provides a detailed and accurate assessment of potential LIP flooding at the site. The staff reviewed and determined that water levels outside key buildings would remain shallow and short in duration. For this reason, the staff determined that LIP flooding does not need to be included in determining the site-specific design basis flood. The staff also determined that the simulated LIP flood levels remain below critical thresholds for SR structures and that no additional flood protection measures or barriers are needed to protect the plant from potential LIP flooding. The staff determines that the applicant's evaluation of potential onsite flooding from extreme rainstorms, LIP, is acceptable.

2.5.1.1.3 Flooding from Rivers and Streams

PSAR section 2.5.1.2 discusses the applicant's analysis of the Probable Maximum Flood (PMF) for the North Fork Little Muddy Creek near the plant site (see PSAR figure 2.5-1).

2.5.1.1.3.1 Rainfall Scenario

The staff independently performed confirmatory calculations on the applicant's PMP estimates using HMR 49 (National Oceanic and Atmospheric Administration (NOAA), 1984) guidelines. The staff calculated a 1-hour, 1-square-mile PMP and adjusted it for basin size, long-term climate trends, and elevation, resulting in a 7.77-inch PMP. This value is approximately 14 percent lower than the applicant's estimate of 9 inches, confirming that the applicant's PMP estimate is conservative. The staff also confirmed that the applicant's use of the center peaking PMP distribution in time is considered a conservative approach compared to other methods.

Furthermore, the staff reviewed the applicant's PMP analyses and estimates and determined that they adhere to the recommendations outlined in RG 1.59 and DG-1290. Consequently, the staff determined that the applicant's PMP scenario in the hydrologic modeling is conservative and acceptable for evaluating the PMF at the North Fork Little Muddy Creek.

2.5.1.1.3.2 Hydrologic and Hydraulic Modeling

The staff conducted a review of the applicant's Hydrologic Engineering Center's Hydrologic Modeling System (HEC-HMS) and HEC-RAS modeling efforts. Spot checks were performed on curve numbers, erodibility factors, and roughness coefficients, utilizing the cited references. Based on this review, the staff determined that these parameter values are adequate and conservative for the creek and its basin in estimating the creek PMF.

The staff reviewed the applicant's approach applying by auditing the outflow from the East Subbasin at its actual outlet within the HEC-RAS model. The staff questioned the modeling between the outflow point and actual outlet in HEC-RAS model. The applicant utilized four unequal subbasins in the HEC-HMS model (see PSAR table 2.5-2). The staff audited the adequacy of using this subbasin scheme and verified the effects of using a smaller number of unequal subbasins by conducting an independent modeling. That is, the staff conducted confirmatory modeling using a 2D HEC-RAS simulation to cross-check the applicant's approach of linking hydrologic and hydraulic models, as discussed later in this SE.

2.5.1.1.3.3 Long-Term Climate Effect

The applicant assessed the potential impacts of long-term climate on PMP depths. Several studies, including Kunkel et al. (2013), the U.S. Environmental Protection Agency (EPA) (2022), and the National Climate Assessment (USGCRP, 2018), indicate an increase in future PMP. These trends are supported by the physical relationship between temperature and humidity under the projected long-term climate. A warmer climate leads to increased evaporation, resulting in higher levels of atmospheric water vapor and consequently more frequent and intense precipitation events.

The IPCC Fifth Assessment Report (AR5) provides regional projections for Representative Concentration Pathway scenarios (IPCC, 2017). For RCP4.5 (intermediate emissions), the region may experience a 10-19 percent increase in precipitation from the heaviest 1 percent of events by the end of the 21st century. For RCP8.6 (high emissions), the projected increase in precipitation from the heaviest 1 percent of events for the period 2070-2099 (relative to 1986-2015) is estimated to be between 20-29 percent. Based on these projections, the applicant has adopted a 20 percent increase in the PMP value to account for long-term climate effects. The staff acknowledges that the impacts of long-term climate trends on extreme floods can vary regionally. The staff reviewed the applicant's assessment of the potential impacts of long-term climate and the staff determined that the applicant's consideration of a 20 percent PMP increase is conservative and acceptable.

2.5.1.1.3.4 Confirmatory Analysis

PSAR section 2.5.1.1.1 states there are no historical flood records for the North Fork Little Muddy Creek. As a result, the applicant could not calibrate or verify their hydrologic and hydraulic models. The staff's review focused on the setup of the hydrologic model, including: (i) using only four subbasins to represent the entire watershed, (ii) uneven subbasin areas, with the largest subbasin (East Subbasin) covering about 68 percent of the basin, and (iii) applying a single set of topographic parameters (such as length, width, slope, and channel characteristics) for each subbasin.

The staff conducted an initial review of detailed topography and elevation data downloaded from the USGS website (USGS, 2024) and found significant variations in the land elevation and landscape, including many small channels within each subbasin. Additionally, the 1D HEC-RAS utilized by the applicant employed two uniform velocities to represent flow within the channel and floodplain cross-sections. This simplified approach may introduce inaccuracies in the simulation of peak flow and its timing if not properly calibrated. To cope with these modeling concerns, staff performed an independent analysis using the 2D HEC-RAS model.

The 2D HEC-RAS is a more advanced, accurate hydraulic tool that simulates flow across complex terrains and provides more accurate water movement data compared to 1D HEC-RAS. It allows flow to move in any direction within a small grid, which is ideal for mapping floodplains and areas with variable flow patterns. The model is more accurate and detailed, but it requires more computing power and time.

For the creek PMF analysis, the staff set up a HEC-RAS using the 2D option, with a boundary based on the basin's land surface elevation for the basin area of 27.2 square miles. All upstream contributing basins were included in the model. The staff used a detailed digital

elevation map with 3-ft resolution downloaded from the USGS website (USGS, 2024) to create a computational grid with 30 × 30 ft square grid cells, resulting in about 350,000 grid cells. The staff assigned water inflows and outflows using boundary conditions and roughness coefficients (e.g., Manning's n-value) for different land covers to account for surface resistance in overland and channel flows. After setting up the model, the staff ran simulations to evaluate water depths, flow velocities, and flood extents.

The staff ran the 2D HEC-RAS model for a base condition that included PMP with a 40 percent antecedent condition, an n-value of 0.035, PMP back peaking in time, a 30 × 30 ft grid, and the effects of long-term climate trends. This base condition is more conservative than that of the applicant's scenario, as shown by resulting flood level estimates. Table 2.5-1 in this SE summarizes the maximum flood elevation at the location of the east creek point adjacent to the plant site (designated as east lower EL8 cross section (EL8) in the applicant's HEC-RAS model, as shown in PSAR figure 2.5-15) and the corresponding margin below plant grade. This table also includes those of dam failure flooding which will be discussed later. The staff identified that the maximum flood level estimate using HEC-RAS 2D is 1.9 ft higher than those of applicant, mainly due to using conservative PMP distribution and Manning's n-value.

Since historical stream gaging data to calibrate the model were not available to the creek, the staff conducted eight sensitivity runs using HEC-RAS 2D to check how different modeling factors and/or parameters could affect the flood level estimates. The scenarios tested included varying (i) PMP temporal distributions, (ii) PMP depths, (iii) roughness coefficients, and (iv) grid cell sizes (See SE table 2.5-2). Key observations from the staff's sensitivity analysis are:

- The staff's sensitivity analysis shows that the change in flood level estimates by changing many input and parameter values are approximately within a foot. The depth and distribution of PMP are sensitive, while n-value and grid resolution are relatively less sensitive.
- Of all sensitivity scenarios, change in PMP depth has the greatest effect on flood level estimates. A 20 percent increase in PMP due to long-term climate trends raised flood levels by nearly a foot at EL8, although the change was not linear.
- The PMP back peaking scenario results in more conservative flood estimates than other distribution scenarios, though the differences at the cross section EL8 were less than a foot.
- The roughness coefficient was less influential than PMP depth; lower n-values led to higher flood levels.
- The grid cell size had the least impact, with smaller cells causing a slight increase in flood levels. The grid cells less than 30 × 30 ft increase are not practical as these increase computing time drastically.
- The maximum flow velocity at the center of creek channel near the plant (EL8) was 10.4 ft/sec, which is slightly higher than the average main channel velocities of 8-9 ft/sec reported by the applicant. However, floodplain flow velocities were much lower (~5 ft/sec), approaching near zero at the floodplain rim near the plant. Therefore, the hydrodynamic effects at the embankments around the plant is negligible.

The staff also reviewed the applicant's estimate of wind wave effects on flood levels. The applicant has calculated maximum wave runup based on 2-year recurrence winds, using equations for significant wave height and spectral peak period. The staff reviewed the applicant's approach and determined that it follows RG 1.59, and the estimated wind wave effect of 1.1 ft is acceptable. After performing an independent analysis of the PMF for the creek, the staff determined that the applicant's PMF estimates in the PSAR are conservative and acceptable.

2.5.1.1.4 Flooding from Dam Failure

The applicant's dam failure analyses considered two failure modes separately: seismic and overtopping (hydrologic) failures. Each mode is analyzed in conjunction with various coincidental riverine flood conditions. The applicant has determined that sunny-day failure will not produce damaging flooding at the plant site, as it does not coincide with an extreme precipitation or flooding event; therefore, it was not included in the PSAR.

2.5.1.1.4.1 Seismic Dam Failure

The staff reviewed the applicant's modeling of dam failure floods. To estimate the 500-year precipitation value, the applicant used 500-year flood discharge data based on precipitation values from NOAA Atlas 14, volume 1 (NOAA, 2011). Since volume 1 does not cover WY, the applicant used data from a similar area in Utah, about 25 mi west of the plant site, to estimate the 500-year precipitation for the pond basin.

For the review, the staff used updated NOAA Atlas 14, volume 12 (NOAA, 2024), which includes storm data for WY. The staff compared the 500-year precipitation values from both Atlas volumes (See SE table 2.5-3). The staff found that the 5-minute precipitation depths are nearly identical. However, for durations of 15 minutes to 1 hour, the values from volume 1 are higher than those of volume 12, resulting in more conservative flood estimates. Based on this finding, the staff determined that the applicant's use of design rainfall data from Atlas 14 volume 1 provides more conservative (and safer) flood hazard estimates and is acceptable.

The staff conducted confirmatory modeling of hypothetical dam failure flooding scenarios using the HEC-RAS 2D option with a 30 × 30 ft grid. The staff simulated 500-year flooding events both with and without seismic dam failures for the three ponds. As a result, the staff found that the flood levels at the plant site for the seismic dam failure is approximately 2.7 ft lower than the applicant's seismic failure estimates. The staff's confirmatory modeling verified that the applicant's seismic dam failure flood estimate is conservative and bounded by the applicant's overtopping flooding scenario. The staff determined that the applicant's seismic dam failure flood estimate is acceptable.

2.5.1.1.4.2 Overtopping Failure

The staff reviewed the applicant's evaluation of potential flooding caused by overtopping failures of the embankments at each upstream pond. The applicant evaluated three ponds for overtopping failure and several smaller ponds near the Naughton Power Plant for potential overtopping risks. Specifically, the applicant assessed the North Ash Pond (NAP), located upstream of Lake Arambel, covering 123 acres. Closure plan reports for the NAP, prepared by

PacifiCorp (2021, 2023), detail its unlined surface impoundment with earthen embankments and a storage capacity of 2,580 ac-ft. Key closure activities include:

- Removing bottom sediment and remediating affected areas.
- Dewatering through decanting and enhanced evaporation, aiming for closure by 2029.
- Sourcing soil from embankments and nearby borrow areas.
- Diverting stormwater and regrading to prevent future impoundments.

Due to reduced water levels after these activities, the staff determined the risk of NAP overtopping impacting Lake Arambel and downstream areas is minimal. Smaller nearby ponds were excluded from applicant's analysis because of negligible storage capacities and planned closures. This aligns with the applicant's overtopping scenario. Through the audit, the staff confirmed that the post-flood, minor sediment deposition occurs over the wide floodplain but is dispersed and does not impact maximum flood levels.

The staff reviewed the applicant's breach parameter estimates in the PSAR and the staff performed independent calculations to determine they were adequate. The staff also reviewed the 1D HEC-RAS model cross-sections and found them consistent with USACE guidelines. Because the applicant lacked historical flow data and could not calibrate the model, the staff conducted independent 2D modeling for the creek's PMF analysis. The staff added conservatism in the 2D HEC-RAS model, including fine grid cell resolution (30 × 30 ft), climate-adjusted PMP, back peaking PMP distribution for maximum flood alignment, and conservative Manning's n-value of 0.4 for channel and overland plain.

The staff found that the postulated overtopping scenario resulted in bounding flood estimates, as summarized in SE table 2.5-1. However, the staff noted the net increase to the maximum flood level due to overtopping is small (only 0.5 ft water level increase) at the plant site due to peak outflow attenuation along the creek. In summary, the staff identified several conservative assumptions in the applicant's overtopping dam failure flood analysis, including the following:

- Use of 500-year precipitation values.
- A 25 percent reduction in lag times to increase peak discharge.
- Conservative initial pond water levels.
- Matching overtopping outflow and rainfall peaks.

Through the staff's confirmatory analysis, the staff validated the applicant's conservative assumptions and modeling approach, concluding that applicant's flood hazard estimates for overtopping failures are reliable and acceptable.

2.5.1.1.5 Other Flooding Mechanisms

The proposed plant site is not located adjacent to seacoasts or large bodies of open water such as lakes and reservoirs. The staff reviewed the applicant's assessment of other flooding mechanisms and determined that the application's statement that a detailed analysis of flooding due to surge, seiche, and tsunami is not required for this CP application. This SE section focuses on the staff's review of the effects of channel erosion, ice, and channel diversion impacting flooding.

2.5.1.1.5.1 Erosion and Sedimentation

The staff performed confirmatory calculations using HEC-RAS 2D flood analysis to validate the applicant's findings on channel erosion and sedimentation resulting from flooding. Flood velocity estimates, which approximately match the applicant's estimates, are low near the plant site (about 5 to 6 ft/sec in flood plain), the maximum flood level is below the plant grade, and flow directions are away from the plant site, minimizing impacts to the plant facilities. Sedimentation is expected to occur upstream, as the adjacent creek's steep slope promotes sediment deposition that is moved away from the bottom of the channel and floodplain. Therefore, the staff determined that erosion and sedimentation during severe flooding would have a minimal impact on the plant facilities, and that the applicant's analysis is reasonable and acceptable.

2.5.1.1.5.2 Ice Effects

The applicant assessed ice-related hazards, including potential effects of ice on flood hazards and flood protection measures for KU1 in PSAR section 2.5.1.6. Based on historical weather data and the USACE Ice Jam Database, the applicant described the potential for ice-effect in the PSAR.

The staff confirmed from the Fourth National Climate Assessment (USGCRP, 2018) that the average annual temperature in the contiguous United States would be increased as the applicant reported, and that the temperature changes will be more pronounced in extreme heat compared to extreme cold. The staff also found that, since the mid-1960s, heat waves have become more frequent and intense, while cold waves have decreased in both frequency and intensity. This trend is expected to continue, with fewer freezing days and more days exceeding 90°F. The warming effect is particularly significant in high-latitude and high-altitude regions, where accelerated warming can lead to increased ice melting and reduced ice formation.

The staff's review of the climate projection maps provided by EPA (2022) indicates that average annual temperatures in the U.S. could increase by 5.3–6.7°F by 2060, affecting all seasons, including winter. Extreme temperature variations, both hot and cold, are expected to rise faster than annual averages, suggesting a more significant impact on ice thickness than average temperature changes alone. Many long-term climate studies identified that climate warming could reduce ice thickness. These long-term trends indicate that the climate will likely lead to thinner ice layers and reduced ice formation over time, with potential implications for hydrological systems in regions like KU1. Therefore, the staff determined that ice formation during winter poses minimal risks to KU1.

In summary, the staff determined that the site has no safety-related SSCs that are subject to ice-induced forces or blockages from either sheet ice or frazil ice formation on open water

bodies. The staff found ice-related risks remain manageable, and the issue does not impact safe operation of the plant. Therefore, the staff determined that the applicant's assessment that the effects of long-term climate trends will not exacerbate ice and flood hazards for KU1 is acceptable.

2.5.1.1.5.3 Channel Diversions

PSAR section 2.5.1.7 discusses potential flood hazards from channel diversions near KU1. The applicant has assessed the potential of the Hams Fork River flow diverting into the site basin. This diversion scenario could generate a damaging flood at the plant site. Specifically, the applicant investigated the potential failure of the Viva Naughton Dam on Hams Fork River with landslides that would block the flow of water further downstream and result in the diversion of flow from Hams Fork River to North Fork Little Muddy Creek. The applicant reviewed available hydrologic, hydraulic, topographic, and geologic evidence as well as human-induced activities. As a result, the applicant found that the water surface behind the landslide will be well below the basin boundary of the two watersheds. Therefore, the applicant determined that channel diversions are unlikely to pose flooding hazards or affect plant safety functions.

Additionally, the applicant has found no significant course changes in streams near the site, including North Fork Little Muddy Creek, over the past 150 years. While landslides around the steep watershed area are possible, no incidents have been reported along North Fork Little Muddy Creek. The area's wide valley topography minimizes potential landslide and channel diversion impacts on nearby creeks.

Ice jams have been reported in Wyoming but are rare in Lincoln County. Observed backwater effects on the Hams Fork River caused no damage, and no ice jam-related diversions or damages have recorded on North Fork Little Muddy Creek. The applicant has found no credible evidence that channel diversions could cause flooding near the site. Sediment deposition could alter channels gradually and locally, but the impacts would be minor and short-lived. The site elevation, over 15 ft above North Fork Little Muddy Creek, protects SR structures from flood risks.

The plant's primary water source is Hams Fork River, but the plant does not rely on SR water systems, so water supply loss would not impact safety. Potential debris flow from Blazon Gap is not expected to cause backflow affecting the design basis flood level. The staff reviewed the applicant's assessment of channel diversion and determined that channel diversions are unlikely during the plant's operational life.

2.5.1.2 Flood Protection

The applicant has established the design basis flood elevation for KU1 at 6,755.4 ft MSL (i.e., NAVD 88). The applicant estimated this design parameter using a postulated scenario combining a PMP event, overtopping (hydrologic) failure of the Lake Arambel embankment, effects of long-term climate, and wind-induced wave action.

The staff reviewed the applicant's proposed flood protection measures, including site grading, drainage design, and embankment slope protection. The staff observed the following:

- The NI layout and grading effectively mitigate LIP events, ensuring positive drainage to nearby channels without impact on SSCs.
- The applicant proposes using riprap to protect the embankments. This method is considered sufficient because any potential erosion from floods will be minor and localized. If the riprap is damaged, it can be easily repaired immediately without causing major damage to the embankments.
- Predicted groundwater levels remain well below the site grade and do not threaten SSCs due to the limited lateral movement of water and the transient nature of elevated levels during extreme floods.
- Below-grade concrete structures are appropriately damp-proofed and waterproofed to prevent water ingress.

The staff determined that safety-significant SSCs remain unaffected by postulated flood scenarios, and that no above-grade flood protection measures are required within the NI, because the combination of site grading, layout, and embankment slope protection provides adequate mitigation against potential flood scenarios applicable to the site. The proposed flood protection measures are sufficient to ensure the continued safety and functionality of KU1 plant during any foreseeable flooding conditions.

2.5.1.3 Groundwater

2.5.1.3.1 Groundwater Characterization

2.5.1.3.1.1 Groundwater Data

PSAR section 2.5.3.1 addresses groundwater characteristics and groundwater use at and around the plant site. The staff focused its review on the characterization of SR aquifer and groundwater properties, such as maximum groundwater level, construction dewatering, groundwater use, and pathways and travel times for radionuclide releases to the ground and surface water.

The staff reviewed the applicant's groundwater data and determined that the data presented in the PSAR provides an adequate basis for understanding groundwater characteristics. The staff determined that the applicant's evaluation of hydrogeologic site characteristics, including the conceptualization of groundwater flow patterns and pathways and travel times for radiological releases, is reasonable and acceptable as part of the hydrology safety analysis.

2.5.1.3.1.2 Groundwater Modeling

The applicant conducted a 3D groundwater model using MODFLOW-2002 (Harbaugh, 2005) to evaluate post-construction groundwater conditions. The modeling aimed to determine the design basis maximum groundwater level and to assess dewatering rates during construction.

The applicant's simulations predicted a maximum groundwater level of 6,738 ft MSL, approximately 18 ft below plant grade. The applicant evaluated hydrostatic loading due to

groundwater elevation changes using advanced modeling techniques tailored to site-specific conditions. The model provided insights into uplift pressures, which ranged from 811 to 6,178 psf for the reactor shaft, the deepest structure at the site. The staff reviewed this assessment and found it to be robust, employing conservative parameter values to ensure safety.

The applicant determined that permanent dewatering systems would not be necessary for ongoing plant operations. However, temporary dewatering measures would be required during construction activities. The same groundwater model was used to estimate construction dewatering rates of 49 gal per minute, assuming all excavations at the plant site were conducted simultaneously. This value will be used to inform engineering decisions for excavation and foundation works.

The staff audited the applicant's groundwater modeling methodology, assumptions, and results. The staff found the modeling to be consistent with current industry best practices and regulatory guidance in RG 1.59, as described in the PSAR. The staff determined that:

- The model was appropriately calibrated using recorded groundwater levels, demonstrating good agreement with observed data.
- The selected boundary conditions (no-flow, drain, and general head boundaries) were appropriate for the site and structural layouts.
- The applicant incorporated variations in hydraulic conductivity and other uncertainties to ensure the robustness of the simulations.
- The inclusion of a 20 percent increase in recharge rates to account for the impacts of long-term climate trends was considered conservative and supported by referenced studies (Kunkel, 2013). This adjustment resulted in a modeled 1-ft increase in the maximum groundwater level.

In summary, the staff reviewed the applicant's groundwater modeling methodology and assumptions and the staff determined that they are technically sound and consistent with available regulatory guidelines. The staff determined that the simulated maximum groundwater level and construction dewatering rates, as determined using the three-dimensional groundwater model, are valid and conservative.

2.5.1.3.2 Accidental Releases of Liquid Effluents to Ground and Surface Water

PSAR section 2.5.3.2 presents a consequence analysis of a postulated conservative failure scenario involving the rupture of a Liquid Radwaste Processing System. This tank failure would result in the release of 3,200 gal of radioactive liquid. This scenario utilized maximum radionuclide activity concentrations in the Waste Holdup Tank to ensure conservative results. The applicant used the pathway and travel time defined in PSAR section 2.5.3.1.

The applicant has determined radionuclides present in the Waste Holdup Tank using effluent concentration limits provided in 10 CFR 20, "Standards for Protection Against Radiation," Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of

Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage.” To evaluate radionuclide transport, they assumed an instantaneous entry of radionuclides into shallow groundwater with horizontal migration toward the southern site boundary. The applicant derived site-specific dispersive coefficient (K_d) values from the Argonne National Laboratory testing. The applicant assessed a conservative exposure scenario to the user (receptor) of a hypothetical domestic well at the southern site boundary. This location lies within an aquitard with poor groundwater quality, making it an unlikely site for future use.

The applicant estimated the radiation dose to the maximally exposed individual using the LADTAP code tool. The resulting dose of 5.60×10^{-3} millirem per year is significantly below the regulatory limit of 100 millirem per year established by 10 CFR 20.1301.

2.5.1.3.2.1 Independent Assessment

The staff reviewed the applicant's methodology and results and performed an independent analysis to confirm the adequacy of the K_d values. The applicant stated that K_d values for 21 radionuclides were determined by Argonne National Laboratory using 24 onsite rock samples from ten borehole locations. The staff compared the applicant's K_d values with those provided in the EPA (1999) look-up tables for the following three species:

- Cesium (Cs-134, -135, -137): With a clay content of 7.4 percent, the measured K_d of 833 milliliters per gram (ml/g) fell within the EPA range of 30-9000 ml/g.
- Plutonium (Pu-241): With the same clay content and a soluble carbonate concentration of 1.5-4.5 mg/L, the measured K_d of 633 ml/g was slightly below the mean value from the EPA table but still within the range.
- Strontium (Sr-90): With a clay content of 7.4 percent and a pH of 6.84-8.62, the measured K_d of 3.09 ml/g was also within the EPA range of 15-200 ml/g.

As a result, the staff determined that the applicant's K_d values fell within the expected EPA ranges. The staff also found that the applicant's values are, by and large, within the range provided from previous reactor licensing applications during the past 15 years for similar soil/rock mediums. Therefore, the staff determined that the applicant's K_d values are adequate for evaluating the radiological effects in groundwater at the plant site.

2.5.1.3.2.2 Review of Dose Assessment

The applicant described in PSAR section 2.5.3.2.5 that it converted the radionuclide concentration estimates to an annual dose to an exposed individual using an assumed flow rate of one cubic foot per second. The applicant summed the ingestion doses and found that the maximum exposed adult individual would receive a total effective dose equivalent (TEDE) dose of 5.60×10^{-3} millirem per year. This dose is within the 10 CFR 20.1301 limit of 100 millirem. The staff determined that the applicant's estimation approach and results in PSAR section 2.5.3.2.5 meet the guidance and the acceptance criteria in DANU-ISG-2022-02.

The applicant used a hypothetical domestic well located at the southern site boundary for the exposure scenario, ensuring that even unlikely situations were addressed. The staff performed

confirmatory calculations using conservative assumptions for radionuclide pathways, groundwater transport, and exposure scenarios. The staff's confirmatory calculations supported the applicant's findings that estimated doses are within the 10 CFR 20.1301 requirements. The staff determined that the applicant's analysis meets the requirements of PDC 2, which accounts for the most severe natural phenomena historically reported at the site, and satisfied the requirements of 10 CFR 100, which addresses the identification and evaluation of hydrological features.

In summary, the staff reviewed the applicant's analysis of an accidental liquid effluent release and determined that it is reasonable and acceptable. The staff confirmed that the applicant's evaluations, assumptions, and methodologies are technically sound, appropriately conservative, and consistent with the guidance in RG 1.59 and the acceptance criteria in DANU-ISG-2022-02. The staff found that, even with a potential radiological accidental release from the liquid radwaste system, the site will effectively prevent detrimental radiological effects, ensuring compliance with regulatory requirements and protecting public health and safety.

2.5.1.4 Low Water Considerations

The applicant described in PSAR section 2.5.4 that the cooling of SR plant systems does not rely on surface water sources and is not affected by drought conditions. The cooling tower system, used only for normal cooling, is classified as a non-safety-related with no special treatment (NST) system. The applicant asserted that sufficient flow in the Hams Fork River exists to meet the water demands of the Naughton units and KU1, provided the future demands of the Naughton units remain unchanged. As a result, the applicant concluded that no SR future controls are necessary to mitigate low water conditions.

PSAR section 2.5.4.6 states that KU1 does not rely on surface water for SR functions and has no heat sink dependability requirements. The applicant noted that the Viva Naughton Reservoir will supply non-safety-related water for KU1, the Naughton Power Plant, and the City of Kemmerer. The staff notes that the total water demand for the three facilities is 82.6 ac-ft per day, while the Viva Naughton Reservoir's available storage volume at the estimated 100-year low water level is 6,800 ac-ft, sufficient for approximately 54 days. The raw water transfer pumps, rated at 5,270 gal per minute, are used exclusively for NST functions.

The staff reviewed the applicant's assessment of low water conditions and water resource availability and determined it to be acceptable.

2.5.2 Conclusion

The staff finds, based on the review of the application, that the applicant's analyses of hydrologic hazards are sufficient and acceptable as the applicant followed the applicable local, state, federal, and industry guidelines. The staff concludes that the KU1 site is not located where catastrophic hydrologic events are credible, that the applicant considered credible hydrologic events in developing the design basis hydrologic parameters for the proposed plant facility, and that the applicant provided adequate site hydrogeologic characteristics needed to evaluate an uncontrolled release of radioactive materials in the event of a credible accidental occurrence. Therefore, the PSAR section 2.5 meets the regulatory requirements in 10 CFR 50.34(a)(1)(ii), 10 CFR 100.20(c), 10 CFR 100.21(d), and 10 CFR 20.1301 for the hydrological description.

2.5.3 Tables

Table 2.5-1: Comparison of flood level estimates between applicant's analysis and staff's confirmatory evaluation.

Flooding Mechanism	Scenario	Maximum Flood Level at EL8 (ft MSL)		Difference (Applicant minus Staff) (ft)
		HEC-HMS & HEC-RAS 1D by the Applicant	HEC-RAS 2D by the Staff	
PMF	PMP	6,752.7	6,754.6	-1.9
Seismic Dam Failure	With 500-Year Flood	6,755.1	6,752.4	2.7
Overtopping Dam Failure	With PMP	6,755.4 (bounding)	6,754.9	0.5

Notes:

- The maximum flood levels are at the East Lower reach 8 (EL8) cross section specified in the applicant's HEC-RAS model.
- All flood level estimates include the effects of 20 percent long-term climate trends increase and wind wave of 1.1 ft as estimated by the applicant.

Table 2.5-2: Sensitivity Analysis performed using the staff's HEC-RAS 2D model with the creek PMF as a base scenario.

Run ID	Scenario	Increase of Max Flood Level at EL8 (ft)
1	Base with PMP Front Peaking	-0.9
2	Base with PMP Center Peaking	-0.6
3	Base with 0.8*PMP	-1.0
4	Base with 1.2*PMP	+0.9
5	Base with n=0.05	-0.4
6	Base with n=0.06	-0.7
7	Base with 50 × 50 ft Grid Resolution	-0.3
8	Base with 40 × 40 ft Grid Resolution	-0.2

Notes: The HEC-RAS 2D Base scenario includes PMP back peaking, n of 0.35, PMP with long-term climate trends , and 30 × 30 ft grid cells.

Table 2.5-3: Comparison of 500-year precipitation frequency estimates from the NOAA Atlas 12.

Duration	500-year Cumulative Precipitation Values (inches) from NOAA Atlas 14		
	Volume 1 (2011)	Volume 12 (2024)	Difference
5-min	0.75	0.68	0.07
15-min	1.41	1.1	0.31
1-hour	2.35	1.59	0.76
2-hour	2.68	1.88	0.80
3-hour	2.71	2.06	0.65
6-hour	2.75	2.41	0.34
12-hour	2.88	2.82	0.06
1-day	3.29	3.27	0.02

Notes: NOAA ATLAS 14 Volume 1 values without long-term climate trends are taken from PSAR table 2.5-9.

2.6 Geology, Seismology, and Geotechnical Engineering

PSAR section 2.6 provides an overview of the seismological and geological characteristics of the site and surrounding region to permit an analysis of the proposed site for load bearing capability and seismic activity. The PSAR includes the following subsections:

- 2.6.1, “Geologic Hazards”
- 2.6.2, “Vibratory Ground Motion”
- 2.6.3, “Surface Deformation”
- 2.6.4, “Stability of Subsurface Materials and Foundations”
- 2.6.5, “Stability of Slopes”

Section 2.6 of DANU-ISG-2022-02 provides the applicable guidance for the review. In accordance with 10 CFR 100.21(d) and 10 CFR 100.23, the applicant should provide sufficient information on the seismological and geological characteristics of the site and surrounding region to permit an analysis of the proposed site for load bearing capability and seismic activity. This analysis should include derivation of the site specific ground motion response spectrum (GMRS) and support analysis of the structures and seismic effects on SSCs at the proposed site. The GMRS is determined based on the geological, seismological, and engineering characteristics of the site and its environs. The size of the region to be investigated and the type of data pertinent to the investigations is described in RG 1.208, “A Performance Based Approach to Define the Site Specific Earthquake Ground Motion,” Rev. 0 (ML070310619), and should be determined based on an initial evaluation of the regional seismic hazards and their potential impact on the proposed facility. The applicant should summarize the relevant studies

describing the site, the investigations performed, and the investigation results and conclusions. Detailed geological information should be documented in a separate report that is available for the staff to audit.

Additional guidance can be found in the following documents:

- NUREG-2115, "Central and Eastern US Seismic Source Characterizations for Nuclear Facilities," (ML12048A776),
- NUREG-2213, "Updated Implementation Guidelines for SSHAC Hazard Studies" (ML18282A082),
- RG 1.132, "Site Investigations for Foundations of NPPs," Rev. 3 (ML21298A054),
- RG 1.138, "Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants," Rev. 3 (ML14289A600),
- RG 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at [Nuclear Power Plant] Sites," Rev. 0 (ML033280143),

2.6.1 Geologic Hazards

The staff reviewed PSAR section 2.6.1 which describes the local and regional geology as it pertains to geologic hazards, focusing on the geologic history, stratigraphy and lithology, geomorphology, tectonic evolution, and structural geology with increasing detail from the site region (320 km radius), site vicinity (40 km radius), site area (5 km radius) and site location (1 km radius). The staff's review of this section of the PSAR includes discussion of the seismic source characterization model for the KU1 site which was developed as part of a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 study.

2.6.1.1 Site Region Geology

PSAR section 2.6.1.1 describes the regional geologic and seismologic setting of the site. The site is located within the Middle Rocky Mountains physiographic province with the eastern margin of the site vicinity (40 km radius from the site) extending into the adjacent Wyoming Basin province. The staff reviewed the summary of geologic history within the site region which extends over 2 billion years and, within the site vicinity, includes geology that was formed by a series of accretionary events from the Antler, Sevier, and Laramide orogenies. Based on this review, the staff determined that the information provided in PSAR section 2.6.1.1 is an accurate summary of the regional geologic history. Additionally, the staff noted that although the region was subjected to numerous mountain building events, the most recent Laramide orogeny occurred in the Late Cretaceous to early Paleogene and the geologic mechanisms driving that event are no longer active at the proposed site.

2.6.1.2 Site Vicinity Geology

The staff reviewed PSAR section 2.6.1.2 which describes the geologic setting of the site vicinity within a radius of 40 km from the site, which includes Cretaceous sedimentary rocks deposited

contemporaneously with the Sevier orogeny. The staff noted that local geologic features include west dipping thrust faults that were grouped into five major systems that have been overprinted to varying degrees by more geologically recent extensional deformation and normal faulting. The staff further observed the applicant's interpretation that some of these older fault zones may be spatially associated with more recent, Quaternary fault activity, and in isolated cases may show evidence of reactivation. The staff determined that the thrust faults in the site vicinity are not considered active faults for the purposes of the seismic hazards analysis.

2.6.1.3 Area Geology

PSAR section 2.6.1.3 describes the local site area geology within a radius of 5 km from the KU1 site. The staff noted that the bedrock at the KU1 site is sedimentary Hilliard Shale, which is also the foundation unit. The staff noted that the Frontier Formation, Adaville Formation and Hams Fork Conglomerate are also mapped at the surface in the site area.

The staff reviewed the geologic and excavation profiles for the site. The staff noted that the geologic profiles in PSAR figures 2.6-104 and 2.6-105 include the location of two clay seams in the uppermost portion of the Hilliard Shale near the surface, while the excavation profiles in PSAR figures 2.6-108 and 2.6-109 do not. The applicant clarified that profile C-C' in figure 2.56-104 and A in figure 2.6-108 are the same section, as are H-H in figure 2.6-105 and C in figure 2.6-109. The applicant further stated that while not shown in the excavation profiles, clay seam 1 will be encountered in the east and west excavation walls while clay seam 2 will possibly be encountered in the uppermost layers, or perhaps not at all, and may be difficult to distinguish from the weathered rock. The staff determined that the clarification on the location of the clay seams within the near-surface geologic and excavation profiles is sufficient to characterize the site geology. The effect of the clay seams on excavation stability is discussed below in SE section 2.6.4.

The staff reviewed the stratigraphic information provided in the PSAR and supporting audit documents and noted that the thicknesses provided in the subsurface profile vary from the best estimates used in the shear wave velocity profile. The applicant clarified that the difference in thickness is attributable to the difference between the apparent vertical thickness reported in the SSHAC report and the stratigraphic thicknesses provided in the PSAR. The applicant further stated that the difference in thickness does not alter the site response and is therefore insignificant. The staff determined that the explanation of the difference in thickness between the SSHAC report and the PSAR is sufficient and the stratigraphic thicknesses in the PSAR are representative of the unit thicknesses in the subsurface of the site. Further discussion of the effect of differing thicknesses on the site response is provided below in SE section 2.6.2.

The staff reviewed the rock core from the site investigations focusing on the foundation level of the reactor building and noted the presence of features that were not described in the boring logs. The staff examined the core in person during a site visit in November 2024 and determined that the features observed are mechanical fractures associated with fossils in the strata or weakness along the bedding plane. The observed features are not associated with active fracturing of the rock. Therefore, the staff concluded that the geologic characterization of the foundation-bearing rock unit is adequate. Further discussion of the foundation interface is provided below in SE section 2.6.4.

2.6.1.4 Site Location Geologic Mapping

PSAR section 2.6.1.4 describes the site location geologic mapping, which the applicant updated based on recent field reconnaissance and LiDAR analyses to support the development of the application. The staff noted that the applicant stated that the new data can better distinguish the geomorphic and topographic characteristics at the site to align with the field observations of gravel, colluvium, alluvial materials, fan material, bedrock, channel deposits and anthropogenic features mapped as artificial fill. The updated geologic mapping of the surface is provided in PSAR figure 2.6-8. The staff reviewed the explanation for the updated site location geologic mapping and the updated map provided in the PSAR. The staff also visited the site and observed the site location, including areas where the geologic mapping was updated to reflect the geologic characteristics present at the site. The staff determined that the updated mapping reflects the current state of the site, including the geomorphic and topographic characteristics at the site location.

2.6.1.5 Seismic Source Characterization of Kemmerer Unit 1

PSAR section 2.6.1.5 describes the seismic source characterization of KU1 that considers the geologic and tectonic information provided in the preceding sections. The seismic source characterization was conducted as part of a SSHAC Level 3 study performed for the region that encompasses a radius of 320 km (200 mi) around the KU1 site. The KU1 SSHAC report was provided to the staff via letter dated June 17, 2025 (ML25171A021) and summarized in PSAR section 2.6.1.5.

The staff reviewed the seismic source characterization, including the seismicity, seismic source zones and fault sources within the KU1 site region. PSAR figure 2.6-24 illustrates the seismic source zones and fault sources. While reviewing the information provided in the PSAR and in the supporting SSHAC report, the staff identified several areas of clarification related to the treatment of geologic information in the seismic source characterization model. The following sections summarize the staff's review of the seismicity, seismic source zones and fault sources.

2.6.1.5.1 Seismicity

The seismicity of the site is discussed in PSAR section 2.6.1.5.1, which describes the development of the regional earthquake catalog; the applicant compiled this catalog for a region that extended at least 320 km around the KU1 site region. The applicant utilized this regional catalog to characterize the seismotectonic setting of the site region as well as to develop recurrence parameters for certain seismic sources, as described in PSAR section 2.6.1.5.2. The applicant developed the regional catalog from six previous catalog compilations applicable to the region, which primarily included the Updated Central and Eastern United States (CEUS) (Lettis Consultants International (LCI), 2022), the Idaho National Laboratory (INL) (INL, 2022), and the Working Group on Utah Earthquake Probabilities (WGUEP) (Arabasz et al., 2016) earthquake catalogs. Chapter 6 of the SSHAC report provides additional details regarding the development of the resulting compiled regional earthquake catalog. [[

]], which

is shown in PSAR figure 2.6-1 (as well as PSAR section 2.6.1.5.2). The applicant's final catalog includes 22,973 earthquakes ranging in magnitude from E[M] 2.3 to 7.2, which occur from 1850 through December 31, 2020. The applicant then used this catalog to determine the recurrence parameters for the seismic source zones developed for the KU1 site region.

The staff reviewed chapter 6 of the SSHAC report and determined that the applicant followed an acceptable approach to compile the regional earthquake catalog. Specifically, the applicant used the magnitude conversion methods employed by the CEUS-seismic source characterization study catalog development (NUREG-2115). As part of its confirmatory analysis, the staff developed a supplementary earthquake catalog covering the KU1 site region from January 1, 2021, through January 31, 2025. The staff used the USGS National Earthquake Information Center earthquake catalog for this analysis. As a result, the staff determined that the applicant's regional earthquake catalog adequately characterizes the seismicity of the 320 km region surrounding the KU1 site.

2.6.1.5.2 Seismic Source Zones

PSAR section 2.6.1.5.2 discusses the seismic source zones for the KU1 site region. PSAR figures 2.6-24 and 2.6-26 show the eleven seismic source zones for the site region (depicted by the blue lines). PSAR section 2.6.1.5.2 states that the following source zones were included as part of the seismic source characterization model: Intermountain Seismic Belt (ISB), Colorado Plateau-Wyoming Craton (CP-WC), Trona Triggered Seismicity Zone (Trona), Yellowstone Geoid Anomaly (YGA), Eastern Snake River Plain (ESRP), Basin and Range (BR), Northern Rio Grande Rift (NRGR), Yellowstone Caldera (YC), Centennial Tectonic Belt (CTB), Idaho Batholith (IB), and Great Plains (GP). PSAR table 2.6-1 provides the seismic source zone characteristics of each of these source zones; these characteristics include seismogenic thickness, style of faulting, rupture orientation (i.e., strike, dip, and dip direction), maximum magnitude, and recurrence. The applicant used these parameters to characterize the above seismic source zones in the KU1 site probabilistic seismic hazard assessment (PSHA) (summarized in SE section 2.6.2.2). The PSAR notes that of the seismic sources included in the seismic source characterization model, only the ISB and CP-WC source zones contribute significantly to the hazard at the site; the remaining sources each contribute less than 1 percent to the total hazard; this is also shown in tables 11-2 and 11-3 of the SSHAC report. As such, the staff mainly focused its review on these two seismic sources zones.

As noted in PSAR section 2.6.3.1.5.1, the applicant used the regional earthquake catalog compiled for the site region to perform recurrence calculations (i.e., to determine b-values and rates) for these eleven seismic source zones; this is detailed in section 9.2.2 of the SSHAC report. In summary, the applicant performed earthquake declustering to remove dependent events from the catalog. The applicant then developed three subsets of the earthquake catalog by the [[

]] declustering methods. After defining completeness intervals for earthquakes of different magnitudes, the applicant used the penalized-likelihood approach (PLA) to compute spatially smoothed seismicity rates for each of the eleven seismic source zones; this approach was also used in CEUS-seismic source characterization model. The staff determined that the applicant's recurrence calculations are adequate as the selection of three different declustering methods is conservative, and they follow the CEUS-seismic source characterization approach for computing spatially smoothed seismicity rates, which has been previously endorsed by NRC.

PSAR section 2.6.1.5.2 states that the boundary of many of these sources are generally adopted (with some revisions) from the seismic source characterization model for the INL Site (SSHAC Level 3 study). A comparison of the boundaries of the INL (2022) study is shown in PSAR figure 2.6-26; the INL (2022) source zones are depicted in pink while the source zones for the KU1 site are shown in blue. The applicant defined new source zones for areas of Wyoming, Montana, Colorado, and the Dakotas that minimally overlap or lie entirely to the east of the INL study region. These new source zones are based on the seismic source characterization technical integration (TI) Team's evaluation of province-scale geodynamic relationships such as variations in lithospheric structure, heat flow, and state of stress. More detail regarding these evaluations is provided in sections 9.2 and 9.3 of the SSHAC report.

The staff reviewed SSHAC report sections 9.2 and 9.3, which clarified that the seismogenic thicknesses in PSAR table 2.6-1 were based on the expert judgment of the seismic source characterization TI team, which considered the available data on observed seismicity, models of seismogenic crustal thickness, heat flow determinations from previous studies and other data to determine the seismogenic thickness. The staff noted that the applicant considered the contours from Zeng et al. (2022) for certain sources east of the ISB, while the applicant adopted thicknesses for other sources from previous SSHAC studies from INL (2022) and the Next Generation Attenuation for Central and Eastern North America (NGA-East) (Goulet et al., 2018). The staff reviewed the referenced studies as well as the source-specific justifications provided in the SSHAC report and determined that the applicant's assessment of seismogenic thickness is adequate.

The staff also reviewed the applicant's assessment of the other seismic zone characteristics SSHAC report sections 9.2 and 9.3 with a particular focus on the Mmax distributions for the ISB and CP-WC seismic source zones. Section 9.2.1.4 of the SSHAC report states that the Mmax distribution was []. Furthermore, the TI Team's determination of the upper bound in []

[]. The staff reviewed the referenced study, as well as the SSHAC report documentation, and determined that the applicant's assessment of Mmax is reasonable.

In summary, based on the above review, the staff determined that the applicant's seismic source zone characterization is adequate. Furthermore, the applicant's seismic source characterization was performed using SSHAC Level 3 study, which provides assurance that the seismic source characterization for the site represents the center, body, and range of technically defensible interpretations.

2.6.1.5.3 Fault Sources

PSAR section 2.6.1.5.2 also discussed the fault sources. The staff reviewed PSAR figures 2.6-33 and 2.6-35, which shows the specific fault sources as grouped in the seismic source characterization model for the KU1 site, and the three clustered fault sources of which the Rock Creek fault is nearest to the KU1 site and the primary fault source that contributes to the seismic hazard, respectively.

PSAR section 2.6.1.5.2 describes three clustered fault sources, the Rock Creek, Bear River and Greys River faults. The staff noted that the Sublette Flat and Bear Valley faults are also located

in the site vicinity and region and show similar geographic characteristics (e.g., strike, dip and dip direction) and could be interpreted to be spatially associated with the clustered fault sources. However, the staff noted that applicant grouped the Sublette Flat and Bear Valley faults into the Wyoming Salient and not as part of the clustered fault sources. The staff reviewed the applicant's clarification that the Rock Creek, Bear River and Greys River faults are classified as individual fault sources because the faults show temporally clustered behavior with similar frequency of seismic activity and geomorphic expression and are included as separate fault sources within the seismic source characterization model. The staff also considered the applicant's conclusion that these similarities in frequency of seismic activity and geomorphic expression are not shared by the Sublette Flat and Bear Valley faults, which lack paleoseismic data and have geomorphic expressions indicating an absence of recent, clustered ruptures, and were therefore excluded from the clustered faults despite their proximity to the clustered sources. The staff concluded that the information provided on the evidence to support clustering of the Rock Creek, Bear River and Greys River faults based on similar temporally clustered behavior, frequency of seismic activity or geomorphic expression is not seen for the Sublette Flat and Bear Valley faults. Accordingly, the staff determined that the inclusion of the Rock Creek, Bear River and Greys River faults as the only clustered fault sources is supported by the available geologic evidence.

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Figure 2.6-1: Comparison of Mean Hazard Curves Calculated for the Rock Creek Fault.⁸

In summary, the staff performed a detailed review of the applicant's seismic source characterization model discussed in PSAR section 2.6.1.5.2 as well as the more detailed descriptions provided in the SSHAC report. Specifically, the staff's review included the following chapters and appendices of the SSHAC report: chapter 9, appendix P, appendix T, and appendix U. As a result of this review, the staff determined that the applicant adequately characterized the seismic sources that have the potential to impact seismic hazard at the KU1 site. Furthermore, the applicant's seismic source characterization was performed using SSHAC

⁸ The applicant and staff's curves are similar at annual frequencies of exceedance ranging from 10^{-4} and 10^{-5} . The flattening of the NRC of curves at low spectral accelerations is because the staff's calculation only considered the Mmax portion of the logic tree; the Mchar branch was not included, which considers smaller magnitude earthquakes for the Rock Creek fault.

Level 3 study, which provides assurance that the seismic source characterization for the site represents the center, body, and range of technically defensible interpretations.

2.6.2 Vibratory Ground Motion

PSAR section 2.6.2 describes the development of the ground motion characterization (GMC) model, PSHA, and site response analysis performed for the KU1 site, which were developed as part of a SSHAC Level 3 study. This PSAR section also describes the development of the GMRS and safe shutdown earthquake (SSE).

2.6.2.1 Ground Motion Characterization Model

The applicant's GMC model is summarized in PSAR section 2.6.2.1. The development of this GMC model followed the SSHAC Level 3 process in accordance with the guidance in NUREG-2213. PSAR section 2.6.2.1 states that the GMC model consists of three separate models. Each of these models predicts motions at the top of the same reference rock profile (i.e., V_{s30} 2500 feet/second (ft/s) (762 meters/second (m/s))) to account for the different source and path characteristics of the three kinds of earthquakes that could impact the KU1 site. The first model considers ground motion from crustal earthquakes in the faults and area sources within 320 km (198.8 miles) of the site (as shown in PSAR figure 2.6-24). The second and third models consider motions from the more distant Cascadia subduction zone and the New Madrid region, which are shown in PSAR figure 2.6-25.

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The staff reviewed the applicant's adjustments to the NGA-East (2018) and INL (2022) median ground motion model, as summarized above, and determined that the GMC TI Team adequately accounted for the ground motion from potential earthquakes from New Madrid and

the Cascadia Interface by making the appropriate distance and site-specific adjustments as summarized above. The staff notes that these sources are quite distant from the KU1 site, and neither source contributes significantly to the total site hazard. Furthermore, the INL (2022) and NGA-East (2018) GMMs were both developed using SSHAC Level 3 studies, which provides an added level of regulatory assurance that these models adequately capture epistemic uncertainty in median ground motions. In addition, the NRC evaluated and endorsed the NGA East model in Research Information Letter (RIL) 2020-11 (ML20255A115).

The staff focused its review on the GMC model used for the crustal earthquakes since the seismic sources applicable to this model are the most significant contributors to the seismic hazard at the site. PSAR section 2.6.2.1 states that the GMC model for crustal earthquakes in developed using a “backbone” approach, which considers the adjustments of a ground motion prediction model (GMPM) from a “host” region to a “target” region, and the epistemic uncertainty in those adjustments. The adjustments are developed by comparing regional ground-motion data to the predictions of the backbone model. The GMC TI Team selected the Chiou and Youngs (2014) backbone model. Figure Q-3 of the SSHAC report shows the logic tree for the median predictions. Figures Q-4 depicts the logic tree for the partially non-ergodic single-station sigma aleatory variability. Sections 10.3 and Q.3 provide details regarding the series of host-to-target adjustments made by the TI team. As shown in the logic tree in figure Q-3, [[

]] to the SSHAC report.

The GMC TI Team judged the [[

]]. The WUS-CEUS boundary is described in sections 7.4, 9.7, and 10.3.5 of the SSHAC report; the staff reviewed these sections, along with Appendix O, and determined that the GMC TI Team’s selection of the 2023 USGS and Kottke boundaries are representative of the center, body, and range of possible interpretations for the boundary between the WUS and CEUS.

The staff also performed a confirmatory analysis to support its review of the GMC model developed for the crustal earthquakes. This involved comparing USO’s crustal model with the center, body, and range (see SE figures 2.6-3, -4, and -5, respectively) of several GMC models developed from SSHAC studies. SE figure 2.6-2 shows a comparison between the Kemmerer GMC model with the SSHAC Level 1 generic Western United States (GWUS), and other SSHAC Level 3 studies, which included the Southwestern US (SWUS) for Diablo Canyon (SWUS-DCPP) and Palo Verde (SWUS-PV), the INL Site (INL3), and the Hanford Site (Han). The comparison demonstrates that the KU1 GMC model captures a sufficiently wide body and range compared to previously developed SSHAC GMMs for the WUS.

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Figure 2.6-2: Comparison of the center (median) of ground motions from the Kemmerer study, the SSHAC Level 1 GWUS, and previous SSHAC Level 3 studies for 10 Hz magnitude 7 event for distances ranging from 1 to 80 km.

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Figure 2.6-3: Comparison of the body (16th and 84th percentiles) of ground motions from the Kemmerer study, the SSHAC Level 1 GWUS, and previous SSHAC Level 3 studies for 10 Hz magnitude 7 event for distances ranging from 1 to 80 km.

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Figure 2.6-4: Comparison of the range (5th and 95th percentiles) of ground motions from the Kemmerer study, the SSHAC Level 1 GWUS, and previous SSHAC Level 3 studies for 10 Hz magnitude 7 event for distances ranging from 1 to 80 km.

2.6.2.2 PSHA Calculations

PSAR section 2.6.2.2 describes the PSHA calculations performed for the KU1 site. The results of these calculations represent the top of a generic rock profile, which the applicant referred to as the “reference rock profile” and is defined as having a time-averaged shear-wave velocity in the upper 30 m (V_{s30}) of 2,500 ft/s. As such, this SE section focuses on the staff’s review of the PSHA calculations for the reference rock condition. The applicant performed additional analyses to adjust these reference rock results to the site conditions at the KU1 site. Specifically, PSAR section 2.6.2.3.1 discusses the development of hazard curves at the PSHA control point location, which is located at a depth of 130 ft (40 m) below the ground surface. The applicant made additional modifications to the PSHA control point hazard to obtain the GMRS, located at a depth of 15 ft (5 m) below the ground surface; these calculations are summarized in PSAR section 2.6.2.4. The staff’s review of these additional calculations is provided in section 2.6.3.2.3 of this SE.

PSAR section 2.6.2.2.1 summarizes the seismic source characterization model inputs for the reference-rock PSHA calculations, which include the seismic source zones and faults sources in the study region as shown in PSAR figure 2.6-1, as well as the more distant Cascadia Interface and New Madrid fault sources. PSAR section 2.6.2.2.2 summarizes the GMC inputs for the PSHA calculations. The seismic source characterization and GMC PSHA inputs are also described in sections 11.1.2 and 11.1.3 of the SSHAC report. The staff reviewed the applicant's seismic source characterization model and GMC inputs as documented in SE sections 2.6.3.1.5 and 2.6.3.2.3, respectively.

PSAR section 2.6.2.2.3 and SSHAC report section 11.1.4 summarize the results of the reference rock PSHA. Notably, SSHAC report figures 11-2 and 11-3 show that the [[

]]; the percent contribution values are provided in SSHAC report tables 11-2 and 11-3 for 10 Hz and 1 Hz, respectively.

The staff reviewed the applicant's PSHA for reference rock in the PSAR and SSHAC report sections noted above. Based on this review, the staff determined that the applicant's PSHA calculation is consistent with the guidance in DANU-ISG-2022-02, which references RG 1.208. The staff also performed a simplified PSHA calculation to confirm the applicant's PSHA calculation. SE figure 2.6-1 shows the resulting for comparison of mean hazard curves calculated for the Rock Creek fault. The results of the staff's confirmatory calculation provide confidence that the applicant's PSHA calculations are acceptable.

2.6.2.3 Site Adjustment Factors

The PSHA calculations summarized above in SE section 2.6.2.2, calculated the rock hazard curves for a reference profile. Site response calculations are necessary to adjust these hazard curves for the GMRS horizon at the KU1 site.

PSAR section 2.6.2.3 summarizes the site response calculations that were performed for the PSHA control point location at a depth of 130 ft; these results were convolved with the reference rock hazard (summarized in SE section 2.6.2.2) to obtain the hazard at the control point (Approach 3 from NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines" (ML013100012)). Specifically, the applicant stated that it used the one-step approach in NRC RIL 2021-15. The applicant's calculations also employed random vibration theory and the equivalent-linear approach. The resulting mean horizontal uniform hazard response spectra (UHRS) at the PSHA control point are shown in PSAR figure 2.6-75 as well as PSAR table 2.6-13.

The applicant performed additional site response calculations to account for the material between 130 ft (39.6 m) and the GMRS horizon at 15 ft (4.6 m) depth (below the ground surface elevation of 6756 ft NAVD 88), which are described in PSAR section 2.6.2.4. These calculations also utilize an equivalent-linear random vibration theory approach as well as Approach 3 from NUREG/CR-6728 to develop 10^{-4} and 10^{-5} UHRS at the GMRS horizon. The resulting 10^{-4} and 10^{-5} UHRS are used to develop the GMRS.

The SSHAC report provides additional details regarding the applicant's site response inputs, methodology and results. SSHAC report chapter 8 describes the input to the site response analyses; section 10.10 of chapter 10 provides details on the site amplification approach; section 11.2 of chapter 11 provides further discussions of the site response analyses including the results and sensitivity analyses (section 11.2.8); Appendix R provides the guidelines for conducting the site response analyses at the KU1 site as well the input data files used in the calculations. In addition, the GMRS report provided to the staff via letter dated June 17, 2025 (ML25171A021) provides details regarding the additional site response calculations performed for the GMRS and FIRS locations, which were not included in the SSHAC report.

The staff reviewed the SSHAC report sections and supplemental technical report referenced above and determined that the applicant's site response methodology is consistent with the guidance in DANU-ISG-2022-02 (which references RG 1.208) as well as more recent staff guidance on site response as documented in NRC RIL 2021-15. However, the staff determined that the applicant deviated from the one-step approach by defining the top of the target profile at a depth of 130 ft rather than at the GMRS horizon. Specifically, the applicant performed site response calculations for the PSHA control point location at a depth of 130 ft using the one-step approach and then performed an additional set of site response calculations to develop site-specific hazard curves at the GMRS horizon. The staff performed confirmatory site response calculations to assess the significance of the applicant's modified one-step approach. The staff performed a single set of site response calculations by employing the one-step approach to calculate site specific hazard curves at the GMRS horizon. A comparison of the resulting GMRS is shown in SE figure 2.6-5. The applicant and staff's GMRS are similar at frequencies below ~4 Hz. The applicant's GMRS is more conservative than the staff's GMRS at frequencies above ~4 Hz. The staff determined that some of this conservatism is due to the applicant's use of a broader range of shear modulus reduction and damping (MRD) curves, which is described in more detail below. In summary, the staff determined that the applicant's modified site response approach did not have a significant impact on the site GMRS at the KU1 site.

In addition to reviewing the applicant's site response approach, the staff reviewed the applicant's site response logic tree, shown in PSAR figure 2.6-73a (as well as figure 8-96 and figure R-1 of Appendix R of the SSHAC report), which provides the main site response inputs (and epistemic uncertainty). Section 8.6 of the SSHAC report provides additional details about the construction of the site response logic tree.

The first three nodes of the logic tree correspond to the construction of the Vs profiles and result in 20 Vs profiles. Section 8.6.2 of the SSHAC report provides details on their development and weighting. The []

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The applicant also collected [[

]]. The staff audited the results of these calculations in the electronic reading room and confirmed the applicant's analysis. The staff reviewed the applicant's Vs measurements and the resulting site response Vs profiles detailed in sections 8.1 through 8.4 of the SSHAC report. Based on this review, the staff determined that the applicant adequately characterized the Vs profile of the KU1 site by [[

]] as

shown in figure 8-96 of the SSHAC report.

The [[

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The application of these curves in terms of the logic tree development is discussed in sections 8.6.4 and R.3 (appendix R) of the SSHAC report. The [[

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The approach used to develop the [[

]]. The staff reviewed sections 8.5 and 8.6.4 of the SSHAC report and determined that the applicant's approach to develop [[

]] is acceptable; the staff concludes that the applicant's MRD logic tree branches adequately captures the epistemic uncertainty in the absence of MRD curves based on actual testing.

The final node of the logic tree corresponds to the kappa. Kappa (κ_0) is discussed in PSAR section 2.6.2.3 and in more detail in section 8.6.5 and R.4 (Appendix R) of the SSHAC report. To estimate κ_0 , the applicant used an approach that involved the use of ground motion recordings across the Natrium ground motion network and the inversion models developed as part of the project (in section 7.5); this information included a κ_0 estimate for each station. Their approach involved plotting these κ_0 values versus the estimated V_{s30} for each station; using a linear fit to the data the applicant interpolated for the value of κ_0 that corresponds to the V_{s30} for the KU1 site (559 m/s). This resulted in a κ_0 value of 0.0422 sec. The applicant obtained the additional values of κ_0 shown in the site response logic tree in figure R-1 (appendix R) of the SSHAC report by assuming a lognormal distribution with a standard deviation of ($\sigma \ln \kappa$) of 0.4, which they sampled using [[]]; the resulting in values of κ_0 from 0.0204 sec to 0.0872 sec. The staff reviewed the applicant's methodology to determine κ_0 logic tree branch values and concludes that the applicant's utilization of κ_0 estimates from inversions for each station in the Natrium ground motion network is appropriate in lieu of ground motion recordings at the KU1 site.

2.6.2.4 Site GMRS

PSAR section 2.6.2.4 describes the development of the site-specific GMRS, which is located at a depth of 15 ft below the ground surface elevation of 6756 ft NAVD 88. According to the applicant, this depth corresponds to the uppermost in-situ competent material (V_s greater than 1000 ft/s (5 m/s)). To develop the GMRS, the applicant used the site-specific 10^{-4} and 10^{-5} UHRS, representative of the GMRS horizon (as summarized by the staff in SE section 2.6.2.3).

As stated in PSAR section 2.6.2.4.1, the applicant used the calculation method in ASCE/SEI43-19 for Seismic Design Category 5. The applicant's resulting GMRS values are provided in PSAR figure 2.6-81 and PSAR table 2.6-15. The applicant noted that the horizontal GMRS at PGA is equal to 0.304 g. In PSAR section 2.6.2.4.2, the applicant stated that since the calculated horizontal GMRS is greater than 0.1 g, and has sufficiently smooth and broad spectral shape, the GMRS is selected to be the site SSE, consistent with 10 CFR 50, appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," and RG 1.208. Regarding the vertical SSE, the applicant indicated that it used the Gülerce and Abrahamson (2011) approach, using the appropriate V_{s30} and PGA_{1100} for the GMRS horizon, to obtain vertical-to-horizontal (V/H) ratios, which are shown in PSAR figure 2.6-80; tabulated values are provided in table 16 of the GMRS report. The applicant stated that it applied these V/H ratios to the horizontal spectrum to calculate the vertical spectrum (refer to PSAR figure 2.6-81 and table 2.6-15).

As described in SE section 2.6.2.3, the staff performed a confirmatory site response analysis. Using these results, along with the methodology of ASCE/SEI 43-19, the staff calculated a horizontal GMRS and compared it with the applicant's horizontal GMRS (see SE figure 2.6-5). Despite the differences in site response methodology with respect to the implementation of the

one-step approach in RIL 2021-15, the results are similar. Note that the staff's GMRS is lower at frequencies above ~3 Hz, which is largely due to the applicant's selection of a wider range of MRD curves. As such, the staff determined that the applicant's horizontal GMRS is adequate.

The staff also reviewed USO's V/H ratios and resulting vertical GMRS. The staff concluded that the use of Gülerce and Abrahamson (2011) is appropriate to develop V/H ratios for the KU1 site and significant seismic sources the site based on the 10^{-4} and 10^{-5} hazard deaggregation at 10 Hz (i.e., appropriate for shallow crustal earthquakes in active tectonic regions). The also staff performed a confirmatory calculation and found that the mean results are comparable to USO's mean results provided in table 15 of the GMRS report. SE figure 2.6-6 provides a comparison of these results; the differences are less than 9 percent difference across the entire frequency range.

The staff determined that the applicant's use of ASCE/SEI 43-19 to develop the GMRS for Seismic Design Category 5 is acceptable as it is consistent with the guidance in DANU-ISG-2022-02. According to appendix A to DANU-ISG-2022-02, SSCs designed to this design category would generally meet the requirement in 10 CFR Part 50 appendix S. Furthermore, the staff determined that the GMRS is an appropriate SSE as it satisfies the SSE criteria set forth in 10 CFR 50, appendix S, paragraph IV.a.1.i.

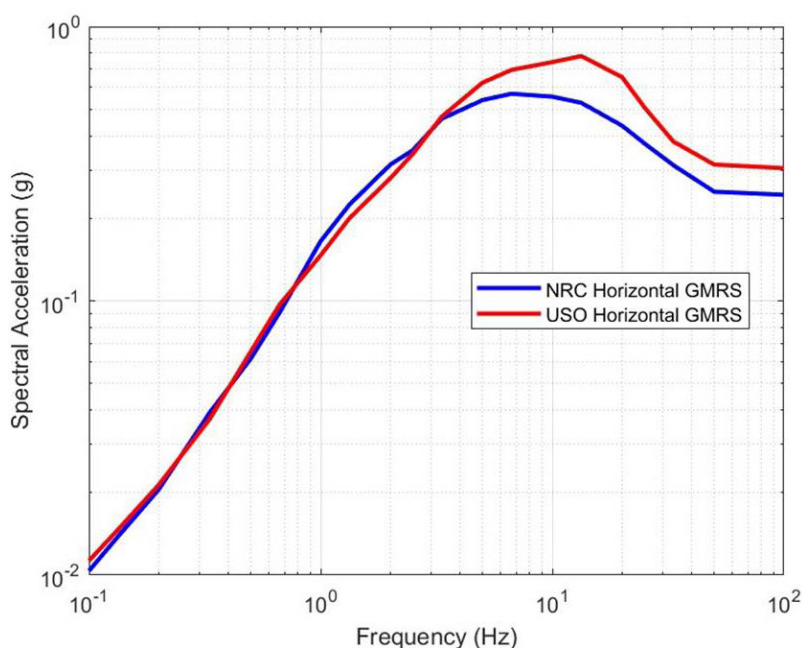


Figure 2.6-5: Comparison of the applicant's horizontal GMRS (red curve) to the staff's horizontal GMRS (blue curve).

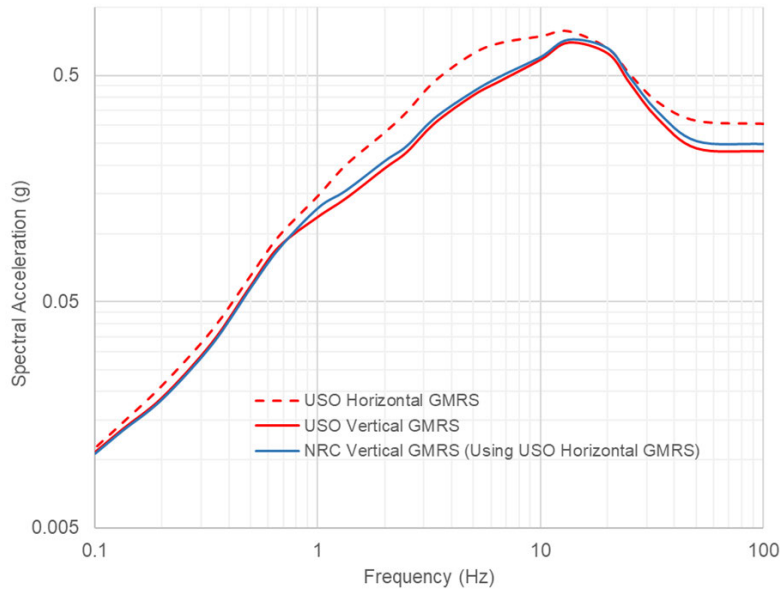


Figure 2.6-6: Comparison of the applicant's vertical GMRS (red solid curve) to the staff's vertical GMRS (blue solid curve). The applicant's horizontal GMRS is depicted by the red dashed curve.

2.6.3 Surface Deformation

The staff reviewed PSAR section 2.6.3 which describes the sources of surface deformation at the site, including landslides, mining or mine collapse, karst and liquefaction as potential non-seismic geologic hazards. The staff noted that the Rock Creek Fault is the only recognized active fault in the site vicinity and the applicant did not observe any evidence of active faulting in the site area. The staff noted that the applicant considered geologic field reconnaissance, published geologic mapping information and previous investigations in the site area, geologic mapping performed to support the CP application, and LiDAR data to characterize the potential for surface deformation at the KU1 site.

The staff reviewed the information provided in the PSAR related to potential sources of surface deformation and observed the site location in the field. With respect to landslides, the applicant determined that because there are no significant landslides in the site area and the site is in a low-lying valley with the nearest ridge is composed of a more resistant sandstone the potential for surface deformation due to landslides is very low. The staff directly observed the proximity of the proposed site to the nearest ridge and determined that deformation due to landslides will not affect the site location. The staff also reviewed the conclusion in the PSAR that there is no evidence for surface deformation due to anthropogenic activities, such as mining or mine collapse, in the site area that present a hazard to the site. The staff observed that the distance to the nearest active mine is nearly 5 mi across the valley and confirmed the applicant's conclusion that there is no evidence that anthropogenic activities would result in surface deformation at the proposed site. The staff also considered the applicant's statement that karst features were not observed during the geologic field reconnaissance or on LiDAR to suggest that karst will be a concern for surface deformation in the site vicinity. The staff did not observe any karst features in the field, nor did the staff identify notable karst features in the core from the

site investigation program. The staff further noted that numerous fractures observed in the core are associated with fossils in the core rather than structural fractures. Finally, the staff confirmed that applicant's conclusion that there is no evidence of diapirism, oil and gas extraction, and fracking and wastewater wells as potential sources of surface deformation in the site vicinity to indicate a potential hazard from these activities.

The staff reviewed PSAR section 2.6.4.5.1 which states that the applicant will perform geologic mapping of the excavation prior to the "placement of shotcrete, backfill materials, dental concrete, or foundation concrete" and this mapping "will include photographic documentation of the exposed surface and documentation of geologic features." The staff is proposing a permit condition to perform detailed geologic mapping in section 2.6.4.2.4.5 of this SE.

2.6.4 Stability of Subsurface Materials and Foundations

USO described the subsurface investigation program at the proposed site in PSAR section 2.6.4.2.1 and the laboratory tests conducted on the samples collected from this site are described in PSAR section 2.6.4.2.2. Properties of the subsurface materials measured using the laboratory tests are described in PSAR sections 2.6.4.2.3 through 2.6.4.2.10. The staff has reviewed these PSAR sections. In addition, the staff has audited the following USO internal documentation:

- NAT-7078 "Subsurface Investigation Data Report," Rev. 2;
- NAT-7015 "Lateral Earth Pressure Calculation," Rev. 0;
- NAT-5037 "Slope Stability Calculation," Rev. 0;
- NAT-4843 "Bearing Capacity and Settlement Calculation," Rev. 1;
- NAT-5040 "Static and Dynamic Geotechnical Engineering Properties," Rev. 0; and
- NAT-5038 "Excavation Support Analysis for Nuclear Island," Rev. 1.

Additionally, the staff has reviewed the response to the staff's audit questions by USO and the documents referred to in the PSAR sections. The staff also reviewed pertinent literature, as appropriate.

2.6.4.1 Subsurface Characterization

The staff reviewed information USO presented on the properties of subsurface materials at the proposed site, given in PSAR section 2.6.3.1 and 2.6.4 and in the documents mentioned above. At the proposed site, USO drilled 111 primary borings and 49 offset borings, a total of 160 borings, as stated in PSAR section 2.6.3.1. Using information from the boring logs and associated laboratory tests, USO concluded that the surficial soil at the proposed site consists of alluvium/colluvium soils and residual soils. The alluvium/colluvium soils are derived from weathering of the underlying residual soils and the bedrock. They were locally transported and deposited by streamflow, gravity, wind, or similar processes. In situ weathering and decomposition of the weathering of the underlying bedrock generated the residual soils at the

proposed site. At the proposed site, Hilliard shale is the bedrock, as stated in PSAR section 2.6.3. Cores obtained from the borings show that the Hilliard shale primarily consists of siltstone, interbedded sandstone, and bentonitic intervals. All borings terminated in the Hilliard shale except boring B-122, which was advanced through the Hilliard shale into the underlying Frontier Formation. Cores from the Frontier Formation include sandstone, siltstone, shale, carboniferous shale, and a coal bed, approximately 28 ft thick at this location, as described in PSAR section 2.6.4.1.1. The geotechnical logs of these borings were audited by the staff and are given in Appendix B.1 of NDP-1162.3. The Hilliard shale and the Frontier Formation show that the stratigraphic units tend to dip between approximately 10° and 25° which matches the observations in regional structure mapping of approximately 20°.

In addition to these borings, USO excavated test pits at the proposed site to collect bulk samples of the subsurface materials and to conduct field thermal resistivity testing. These pits were approximately 4 ft deep. Bulk samples were collected following the American Society for Testing and Materials (ASTM) standard D2488-17e1 (ASTM, 2018). Additionally, borehole geophysical tests; for example, natural gamma, dual induction (conductivity), long- and short-normal resistivities, magnetic resonance, three-arm caliper, spontaneous potential, acoustic televiewer, downhole service logging, and P S suspension logging, were conducted in selected boreholes. Eight uncased (B-117, B-122, B-127, B-134, B-148, B-150, B-160, and B-168) and three PVC-cased (B-117 CB, B-122 CB, and B-127 CB) borings were used for the geophysical testing. The staff notes that only four uncased (B-117, B-122, B-127, and B-134) and all three PVC-cased borings are within the NI. Additionally, USO conducted surface and near-surface geophysical investigation at the proposed site. These investigations include Spectral Analysis of Surface Waves (SASW), field electrical resistivity, and thermal resistivity tests. The staff further notes that USO used the information from the boring logs. USO only used the discontinuities identified from the acoustic viewer logs to perform quantitative evaluation of the subsurface parameters.

The staff notes that USO stated during an audit conversation a total of **[]** discontinuities in the acoustic viewer logs from 12 borings and were used to extract quantitative information to be used in geotechnical design and analyses. The staff audited the boring logs and photographs of the core show thin laminations of the sedimentary rock and saw that these laminations are parallel to the overall dip of the strata. However, these laminations are not detectable in acoustic viewer logs and USO did not consider them in developing engineering design of the subsurface construction. Only bedding planes and joints were used in the geotechnical assessment. The staff determined that these laminations are small-scale features and drilling-induced vibration generally do not open them up easily. Consequently, the staff concludes that these laminations would not have major impact on the geotechnical design of the excavations as they seem to be quite strong and slippage was not observed along the lamination plane. Therefore, the staff determined it acceptable to consider only the bedding planes and joints in the geotechnical analyses to design the subsurface facilities and assess their stability at the proposed site.

The staff determined that most of the test procedures follow the relevant ASTM standards. The procedures used by USO to characterize the discontinuities in the rock mass follow the Suggested Methods by the International Society for Rock Mechanics and Rock Engineering (ISRM). As these are internationally recognized standards and methods to identify and measure different parameters to characterize soil and rock at the subsurface, the staff concludes that USO has used appropriate methods to determine the properties of subsurface materials at the proposed site.

PSAR section 2.6.3.1.3 states that no sinkholes, caves, sinking streams, or other karst features were observed in the field and LiDAR survey of the proposed site. USO confirmed in the PSAR section 2.6.3.1.3 that the nearest limestone exposure is a small outcrop approximately 6 to 9 mi away. The staff reviewed PSAR section 2.6.3.1.3 and determined that no karst-related hazards exist at the proposed site.

The generalized subsurface profile shows an overburden soil (mostly silty/sandy clay) at the top, as stated in PSAR section 2.6.4.2.3. This soil layer is approximately 20 ft (6 m) thick and underlain by weathered siltstone (weathering of the rock varies from completely weathered to slightly weathered). It is approximately 30 ft (9.1 m) thick and underlain by fresh siltstone. Foundation for the RXB will be at approximately 118 ft (36 m) below the finished grade at an elevation of 6,639 ft (2023.5 m) on the fresh rock unit. The finished plant grade would be at elevation 6,757 ft (2059.5 m), as stated in PSAR section 2.6.4.1.2.

Brief description of the engineering properties of the overburden soil, fill materials, clay seams, and both weathered and fresh rock units is provided below.

2.6.4.1.1 Overburden Soils

USO measured the engineering properties of the overburden soils using the Standard Penetration Test (SPT) blow counts. Results are summarized in PSAR table 2.6-21. Strength of the overburden soils was measured using the results of 36 unconsolidated undrained (UU) triaxial tests and 7 consolidated undrained (CU) tests. The static engineering properties of the soil are summarized in PSAR table 2.6-20. Results of the consolidation testing of the overburden soils from locations outside the NI area are presented in table 12-1 of NDP-1162.3.

The staff determined that USO followed the guidance in RG 1.138 and used international standards, such as ASTM standards, to measure different engineering properties of the soil samples collected at the proposed site. Index properties in addition to grain size, natural moisture content, specific gravity, plasticity, shear strength, compaction, and settlement characteristics were measured, as described in PSAR section 2.6.4.2.2. Based on above discussion, the staff determined that USO has used appropriate and industry-recognized standards to measure the engineering properties of the overburden soils at the proposed site.

2.6.4.1.2 Fill Materials

USO proposed to use engineered fill materials in constructing the SR structures at the proposed facility. For example, engineered fills will be used to fill the sides of the SR structures. Additionally, controlled low strength material (CLSM) is planned to fill the annulus between the foundation wall of the RXB shaft and the surrounding fresh rock unit. PSAR section 2.6.4.2.5 specifies the properties of the backfill to be used at the proposed facility. USO has committed to confirm these properties at the OL stage once the source of the borrow material is identified in PSAR section 2.6.4.2.5. The backfill material is assumed to behave as a Mohr-Coulomb material; that is, the material will respond in linear-elastic perfectly-plastic manner when load is applied.

The staff determined that USO proposed to use the compacted granular backfill during construction of the structure. In addition, lean concrete or controlled low strength material is

planned to be used to fill the annular gap between the fresh rock and the foundation wall of the RXB shaft. The staff determined that USO has listed the desired properties of the backfill, which will be conformed later, and currently listed properties seem reasonable.

2.6.4.1.3 Clay Seams

Several parallel clay seams were observed in the borings in the NI area. These clay seams have a dip direction of 277° with a dip angle of 20°. One of the clay seams intersects the foundation of the proposed Reactor Building and was incorporated into the model to assess stability of the Reactor Building. The properties of the clay seam were not measured; however, zero cohesion and a friction angle of 12° were assumed. The staff determined this assumption acceptable as USO used a lower end (conservative) friction angle value of the clay reported in table 1 of Hoek (2007) "Practical Rock Engineering."

2.6.4.1.4 Rock

Weathering of the rock increases closer to the surface. Mean uniaxial compressive strength decreases as the weathering increases, as shown in PSAR table 2.6-25. USO divided the weathered rock into two units: (1) moderately weathered (MW) to slightly weathered (SW) rock and (2) completely weathered (CW) to highly weathered rock (HW) rock, the weathered rock unit closer to the surface. The fresh rock unit lies below the weathered rock units. Mean uniaxial compressive strength of fresh rock is approximately 7,000 psi, as given in PSAR table 2.6-25. It decreases to 3,000 psi in MW to SW rock unit. It is the lowest (1,500 psi) in CW to HW rock unit.

The staff determined that USO has measured the strength and compression properties of fresh rock unit using appropriate methods. The staff determined that USO has used standard methods to measure the properties, and therefore, acceptable. Susceptibility of fresh rock to slaking was also measured. USO has used these measured properties to estimate the rock mass properties discussed below.

2.6.4.1.5 Rock Mass

As discussed before, the natural fractures of the rock mass at the proposed site were measured from the acoustic televiewer logs. Acoustic televiewer logs indicate that only one discontinuity set is present in the rock mass along with many randomly oriented joints present in the subsurface, as stated in PSAR section 2.6.4.2.7. The discontinuity set has a dip of 20° and a dip direction of 285° and corresponds to the bedding planes at the site. The basic friction angle, measured on saw-cut siltstone surfaces, varies from 27° to 33°, with a median of 29°.

USO measured several different parameters to characterize the rock mass at the subsurface at the proposed site. PSAR figure 2.6-91 shows that the Rock Quality Designation (RQD) with depth measured on 1,965 core runs at the proposed site. In the fresh rock unit, PSAR figure 2.6-91 shows that the mean RQD is high, greater than 90 percent. However, RQD values decrease near the surface. This observation indicates that the discontinuities are widely spaced at depth, as indicated by the large fracture spacing of 15 ft or more and the spacing decreases near the surface. [[

]]. Significant variation of the spacing values was observed. PSAR section 2.6.4.2.7 states that fracture spacing as well as the RQD decreases near the surface where the effects of weathering is more pronounced.

Weathering also increases the degree of fracturing in the rock; discontinuity spacing drastically decreases to 3 ft in SW/MW rock unit and less than a foot (0.7 ft) in HW/CW rock unit.

USO measured the estimated Joint Alteration Number J_a of the Norwegian Geotechnical Institute's (NGI's) as a part of the core logging process, as stated in PSAR section 2.6.4.2.7. PSAR figure 2.6-94 shows the J_a values measured in the cores directly following the weathering profile observed at the proposed site. J_a is maximum, approximately 3, in the HW/CW rock unit beneath the overburden soils. It decreases to approximately 2 in SW/MW rock unit, indicating a decrease in degree of weathering as the depth from the surface increases. It is approximately 1.5 at the top of the fresh rock unit and decreases to less than 1.0 at depth.

USO used the Barton-Bandis joint model to characterize the shear strength of the discontinuities. The measured Joint Roughness J_r and the estimated Joint Roughness Coefficient (JRC) values indicate relatively smooth discontinuity surfaces. A value of JRC equal to 2 was used by USO in stability analyses. As strength of the discontinuity walls was not measured, USO assumed the wall strength would be approximately half of the intact rock strength. The staff determined this reduction of strength is appropriate as it is commonly used in practice.

USO also conducted slake durability test on 11 specimens of fresh rock to determine its susceptibility to slaking (cyclic wetting and drying), as stated in PSAR section 2.6.4.2.6. A median value of 97 percent has been obtained for the fresh rock unit indicating that the fresh rock would not disintegrate easily from cyclic wetting and drying.

USO measured the orientation of the in situ discontinuities in the rock mass in acoustic viewer logs from only eight boreholes (PSAR section 2.6.4.2.7) out of 160 boreholes (PSAR section 2.6.3.1) drilled at the proposed site. Some of the discontinuity characteristics, described in PSAR section 2.6.4.2.7, are based on measurements from these eight boreholes and rest are based on observation in all boreholes. The staff determined that USO has measured the discontinuity characteristics following the suggested methods of the International Society of Rock Mechanics or other suggested methods. As these methods are followed worldwide, the staff determined that USO measured the discontinuity properties of the rock mass using appropriate methods and, therefore, are acceptable.

2.6.4.1.6 Rock Mass Properties

As stated in PSAR section 2.6.4.2.8, USO used the Geological Strength Index (GSI) to characterize the rock mass underneath the NI area of the proposed site and used Hoek-Brown failure criterion to estimate its strength. However, USO did not measure the GSI values directly from the fracture network present in the rock mass or converted from the Rock Mass Rating (RMR) index of Bieniawski (1989), which uses the direct observations of different features of the rock mass. Instead of fracture mapping at the proposed site, USO estimated the GSI indirectly using the methodology of Hoek, Carter, and Diederichs (2013). This methodology uses the values of J_r and J_a combined with RQD to estimate the GSI of a rock mass. Hoek, Carter, and Diederichs (2013) commented that this empirical approach would be an acceptable approximation for engineering applications.

PSAR section 2.6.4.2.8 lists the estimated GSI values of the weathered and fresh rock units at the proposed site. For HW/CW rock unit, the estimated GSI is 25. It is estimated to be 60 in the

SW/MW rock unit. Fresh rock unit at the proposed site is estimated to have GSI of 80. USO used 0.7 as the disturbance factor for the first 10 ft of rock beyond the surface of the excavation. The staff determined that 0.7 as value of the disturbance factor is appropriate based on the recommendation of Hoek (2007).

USO estimated the rock mass modulus for each of the weathered and fresh rock units using seven empirical equations available in the literature, as stated in PSAR section 2.6.4.2.8 and is listed along with its variabilities for each rock unit at the proposed site: HW/CW, SW/MW, and fresh rock, in PSAR table 2.6-34. The staff notes that six of these equations use the RMR value of the rock mass as one of the parameters. As the RMR value representing the quality of the rock mass was not measured at the proposed site, USO used the empirical equation correlating GSI with RMR to estimate the rock mass modulus at the proposed site. The staff notes that the variability of the estimated rock mass modulus value also includes variability introduced by the empirical equations correlating RMR and GSI values and GSI value estimated from J_r , J_a , and RQD values, which is not captured in PSAR table 2.6-34.

In addition, the staff notes that the Hoek-Brown material model accounts for the fractures to reduce the compressive strength of the intact rock. As stated in PSAR section 2.6.4.2.8, based on numerous publications by Hoek and his colleagues, both GSI classification and Hoek-Brown material model assume “that the rock mass contains several sets of discontinuities that are closely spaced relative to the proposed structure, such that it behaves as a homogeneous and isotropic mass and that a predetermined failure plane does not exist. In other words, while the behavior of the rock mass is controlled by the movement and rotation of the rock blocks separated by intersecting discontinuities, there are no preferred failure directions. The size of the excavation is expected to be much larger than the rock blocks that make up the rock mass at the site.” However, as stated in PSAR section 2.6.4.2.7, measurements from the acoustic viewer surveys performed in eight borings show that there is only one discontinuity set at the proposed site, which is the bedding plane, many randomly oriented joints. Number of these random fractures decreases as depth from the surface increases as rock quality improves. The staff also observed in borehole logs and core photos that intensity of these randomly oriented joints decreases with depth.

The staff determined application of the Hoek-Brown material model to the rock units at the proposed site does not satisfy several inherent assumptions of the Hoek-Brown model formulation. One of the assumptions is that the rock mass behaves isotopically without any preferential direction of failure. Several sets of discontinuities need to be present in the rock mass so that the fractured rock mass behaves isotopically, as given in, for example, Hoek and Brown (1997) and repeated in PSAR section 2.6.4.2.8. These discontinuity sets should have varying orientations but similar properties. Hoek and Brown (1980) showed that four or more discontinuity sets are needed to approximate the rock mass response to the applied stresses as an isotropic material, similar behavior independent of the orientation, which is one of the basic requirements of the Hoek-Brwon model, as stated in PSAR section 2.6.4.2.8. In addition, as stated in Hoek and Brown (1997) and also in PSAR section 2.6.4.2.8, the subsurface excavation should be large enough compared to the size of the in situ blocks so that the rock mass can be treated as a Hoek-Brown material and any scale effects can be ignored. Given the discontinuity spacing measured in the acoustic viewer logs, as discussed in PSAR section 2.6.4.2.7, the size of the excavation (e.g., reactor shaft) is expected to be large compared to the rock blocks in the weather rock units HW/CW and SW/MW; however, the rock blocks will be of comparable size with the diameter of the reactor shaft, based on PSAR figures 2.6-108 and 2.6-109.

Consequently, requirements of small size rock blocks may also be not satisfied in the fresh rock unit.

USO acknowledged the possible limitations of the subsurface characterization. As stated in PSAR section 2.6.4.5.1, USO has committed to conduct geological mapping of the subsurface excavations in conformance with RG 1.132. USO will conduct the mapping of the geological features of the subsurface before placing shotcrete, backfill materials, dental concrete, or foundation concrete at the excavations. This geological mapping program will include photographic documentation of the exposed surface and documentation of the geologic features. As stated in PSAR section 2.6.4.2.8, USO estimated the GSI of the weathered and fresh rock masses using empirical correlations with information observed in rock cores on roughness and alternation of the fractures. These empirical correlations introduce uncertainties on the estimated GSI values. At the time when USO maps the excavated surfaces (see proposed permit condition below), the staff will review to confirm if the mapped information can be directly used to estimate the GSI of the rock masses.

The staff notes that the current knowledge of the subsurface at the proposed site comes from borings only as no nearby surface exposures were available. Measurements from borings gives one-dimensional information of the subsurface geologic features using a tiny fraction of the entire subsurface volume. Surface geophysical techniques currently do not have sufficient resolution to identify each discontinuity including their spatial extents. Hoek and Brown (1997) also discussed the implications of using just borehole cores to estimate the GSI value for the rock mass. For reasonable quality rock mass ($GSI > 25$), they recommended to determine the RMR value from the cores recovered and then use the correlation equation(s) between the GSI and RMR values to estimate the GSI value. However, for poor quality rock masses ($GSI < 25$), such as, the HW/CW rock unit, Hoek and Brown (1997) recommended to use the fracture mapping to estimate the GSI value, which USO has committed to conduct as the excavation continues, which is described by USO in PSAR section 2.6.4.5.1.

2.6.4.2 Bearing Capacity and Settlement

USO estimated the bearing capacity and settlement of the foundation of the safety-significant structures in the NI area of the proposed facility in PSAR sections 2.6.4.10.1 and 2.6.4.10.2. The staff has reviewed the discussion in PSAR and audited NAT-4843. Additionally, the staff has reviewed selected documents referred to in PSAR sections 2.6.4.10.1 and 2.6.4.10.2.

2.6.4.2.1 Methodologies Used for Overburden Soils

Overburden soils at the proposed site are silt or silty/sandy clay and are considered cohesive soils. USO used the Skempton's equation, as given in Terzaghi, Peck, and Mesri (1996), to estimate the ultimate bearing capacity, q_{ult} , of a foundation supported on the overburden soils. As this equation is widely used, the staff determined that it is appropriate to use this equation at the proposed site to estimate q_{ult} of the overburden soil as the overburden soil is classified as clay (cohesive soil). USO calculated the elastic settlement of the overburden soil using the linear elastic strain equation. The stress distribution below a rectangular flexible foundation is computed using the Boussinesq solution. This load will induce settlement of the underlying soil.

As clay will undergo consolidation with time, the consolidation settlement was estimated using the widely used method given in the literature, for example, the method given in Holtz and

Kovacs (1981). This method uses the laboratory measured consolidation properties of the overburden soil, given in tables 1 and 2 of NAT-4843 to estimate the long-term settlement from consolidation. Both the weight of the structural fill and the foundation loads, estimated using the Boussinesq solution for flexible foundation, will induce the consolidation settlement. Settlement from secondary compression (creep) of the soil is neglected as the soil is inorganic lean clay, which undergoes negligible secondary (creep) compression.

The staff determined that the methods used by USO to estimate the bearing capacity and consolidation settlement of foundations constructed on overburden soils are widely used in the industry and described in the textbooks. Consequently, the staff determined that the selected methods are acceptable for estimating the bearing capacity and consolidation settlement of the overburden soils.

2.6.4.2.2 Methodologies Used for Structural Fill

For estimating the bearing capacity of a foundation over the granular structural fills, USO used the method of Vesic (1975). This staff determined this method is an extension of the Terzaghi Bearing Capacity method and is appropriate for use with structural fill. This method uses a slightly more accurate definition of the shear failure surface and, therefore, provide a slightly more accurate estimate of the bearing capacity, and is acceptable to the staff.

The elastic settlement below a rectangular flexible foundation is computed assuming linear elastic materials layer(s) below the foundation. The load estimated by the Boussinesq solution will induce the elastic strain in each layer beneath the foundation. The elastic settlement will be the summation of the elastic settlement each layer is undergoing.

As compacted fill can undergo long-term creep settlement from self-weight, USO used information given in Naval Facilities Engineering Command (1986) to estimate the magnitude of this settlement as a fraction of the fill thickness. The staff determined the method in Naval Facilities Engineering Command is appropriate to estimate the creep induced settlement as this method is widely used in designing many foundations and earth structures.

2.6.4.2.3 Methodologies Used for Fractured Rock Mass

USO assumed that the fresh rock unit at the proposed site behaves like a Hoek-Brown material and used the analytical method proposed by Serrano et al. (2000, 2001) to estimate the bearing capacity of a rock mass. Method proposed by Serrano et al. (2000, 2001) is specially formulated for a fractured rock mass that can be represented by the Hoek-Brown model. As discussed in PSAR section 2.6.4.2.8, Hoek-Brown material model inherently assumes that the rock mass is well fractured by at least four closely spaced relative to the proposed structure discontinuity (e.g., bedding planes, joint/fracture sets) sets so that the fractured rock mass behaves isotopically (Hoek and Brown, 1997).

The staff determined that the methods proposed by Serrano et al. (2000, 2001) are widely used to estimate the bearing capacity of fractured rock mass if the rock mass can be represented by Hoek-Brown material model and, therefore, is acceptable for those rock masses only.

2.6.4.2.4 Estimation of Bearing Capacity and Settlements for Safety-Related Structures

USO used the Boussinesq solution to estimate the load imposed on the rock layer by the structure and overlapping subsurface materials assuming isotropic subsurface behavior. PSAR figures 2.6-108 and 2.6-109 show the cross-sections of the subsurface through the excavations and the safety-significant structures. The staff has reviewed the estimation of the bearing capacity and settlements of foundation of each of the four safety-significant structures on the NI at the proposed facility; namely, RXB, FHB, Reactor Auxiliary Building (RAB), and NI Control Building (NCB), as described in PSAR section 2.5.1.1.5, and is presented below.

2.6.4.2.4.1 Reactor Building

The foundation of the RXB will be at two levels. The size of the foundation of the Head Access Area (HAA) slab of the RXB is [[

]].

As the reactor cavity will be placed on fresh rock unit, USO used the methodology proposed by Serrano and Olalla (1996) and Serrano et al. (2000, 2001). USO estimated the bearing capacity of the foundation of the reactor cavity to be 76 kilo-pound per square foot (ksf) (3.6 megapascal (MPa)) with a safety factor of 27, which is adequate, as stated in PSAR section 2.6.4.10.1. The staff determined that USO used a methodology appropriate for estimating the bearing capacity of a fractured rock mass that can be represented as a Hoek-Brown material only. As discussed before, appropriateness of the rock mass represented as a Hoek-Brown material is not established yet. USO committed to conduct additional geologic mapping of the exposed surfaces during excavation of the proposed facility in PSAR section 2.6.4.5.1 and in response to audit question 2-15. The estimated settlement (elastic) is less than 0.02 in. in the fresh rock unit, as stated in PSAR section 2.6.4.10.2. Using the same methodology, USO also estimated the bearing capacity of the foundation of the HAA slab to be 73 ksf (3.5 MPa), as given in PSAR section 2.6.4.10.1. PSAR section 2.6.4.10.2 states that the settlement of the HAA slab is estimated to be 0.07 in. Therefore, the staff determined that the foundation of the RXB will have substantial bearing capacity and will be able to tolerate the settlements.

2.6.4.2.4.2 Fuel Handling Building

The below-grade structure of the FHB will have seven foundations at different depths from the finished grade, as shown in PSAR figures 2.6-108 and 2.6-109. In situ soil will be excavated and replaced by granular structural fill. The FHB will have mat foundations over the compacted granular structural fill as the existing elevation at the FHB location is lower than that at the RXB location, as shown in PSAR figure 2.6-108. The basemats of the foundations will be at three different levels supported either on the weathered or the fresh rock unit. The shallow basemat will be on the weather rock unit at El. 6,725 ft. This elevation is near the boundary between CW/HW and MW/SW rock units. As this basemat would be resting on the MW/SW rock unit, USO considered the properties of the MW/SW rock unit to estimate the bearing capacity and settlement of the foundation. The deeper basemats would be on the fresh rock unit at El. 6,710 ft and 6,704 ft.

USO estimated the ultimate bearing capacity of the mat foundations on the structural fill to be 33 ksf (1.6 MPa) and recommended an allowable bearing capacity of 6 ksf (0.29 MPa) with a factor of safety of more than 5, as stated in PSAR section 2.6.4.10.1. The staff determined the recommended allowable bearing capacity of the mat foundations on structural fill is acceptable because it has a high factor of safety.

USO estimated the bearing capacity of the shallow basement on weathered rock at El. 6,725 ft using the methodology of Serrano et al. (2000). As stated in PSAR section 2.6.4.10.1, the estimated bearing capacity is 18.5 ksf (0.89 MPa). The bearing capacity of the deeper basements on fresh rock is 73 ksf (3.5 MPa). USO estimated the settlement of the FHB using a 2D finite element model using the computer program Plaxis 2D as relatively flat stratification was observed near the FHB footprint from the borehole logs. PSAR section 2.6.4.10.2 states that the average settlement of the shallow mat foundation varies approximately from 0.1 to 0.2 in. The FHB basements undergo settlements ranging from 0.02 to 0.03 in. Therefore, the staff determined that FHB basemats are expected to have sufficient bearing capacities and will undergo extremely small settlements.

2.6.4.2.4.3 Reactor Auxiliary Building

RAB is a rectangular building with dimensions approximately [[

]], whereas east portion will be at elevation El. 6,693 ft resting on the weathered rock unit as shown in PSAR figure 2.6-108.

A basemat of approximately [[]] will be placed on the in situ clay at the west side. USO estimated the allowable bearing capacity of [[

]].

As the clay layer is very stiff to hard clay with the SPT blow counts over 60 in most cases, any potential for consolidation settlement of the soil was ignored. Assuming the soil beneath the foundation acts as a layered medium and each layer acting as a linear elastic material, NAT-4843 estimated the settlement of the soil beneath the foundation. [[

]].

USO ignored any settlement from secondary compression of the soil as the in situ soil is mostly lean clay. Although the in situ soil can undergo settlement from secondary compression of the soil particles with time, its magnitude is expected to be negligible as the soil is mostly lean clay at the proposed site. The staff determined that none of the soil samples are classified as organic clays, which generally exhibits appreciable secondary compression. Therefore, the staff determined that the settlement due to secondary compression of the soil below the RAB would

be negligible and USO has used appropriate methods to estimate the bearing capacity and settlement of the RAB.

2.6.4.2.4.4 NI Control Building

The foundation of the NCB will be [[

]]. Consequently, the larger dimensions are used in estimating the bearing capacity and settlement. The foundation of the NCB will be at 33 ft below the final grade (El. 6,724 ft), as shown in PSAR figure 2.6-109. Both MW/SW and fresh rock units would bear the foundation load as the MW/SW rock unit is about 1 ft below the foundation level, USO assumed that the subsurface of the NCB consists of the MW/SW rock unit. The staff determined this a conservative assumption as the fresh rock unit is significantly stronger. Using the methodology of Serrano and Olalla (1996) and Serrano et al. (2000), USO estimated the allowable bearing capacity to be 19 ksf (0.9 MPa), as given in PSAR section 2.6.4.10.1. PSAR section 2.6.4.10.2 gives the estimated settlement of the MW/SW rock below the NCB foundation to be 0.04 in.

2.6.4.2.4.5 Summary of Bearing Capacity and Settlement Assessment

To summarize, the staff determined that USO has used different methods to estimate the bearing capacity and settlements of the foundations/basemats of the safety-significant structures in the NI area located on different strata. The staff determined that the selected methods appropriately account for the particular subsurface characteristics assumed and are commonly used in the construction industry. The staff determined that the estimated bearing capacity of different safety-significant structures located on overburden soil or structural fill show significant factor of safety against foundation/basemat failure and the foundation/basemat of the structures are expected to undergo small amount settlements as the loads are applied. The staff found that these expected settlements can be tolerated by the structures. However, as discussed and also summarized below, the rock mass characteristics assumed for both weathered and fresh rock units may not be appropriate for what is known at this time. As discussed above, USO committed to conduct geologic fracture mapping of the excavated surfaces to determine the rock mass properties appropriate for both weathered and fresh rock units. As a result, the staff added the permit condition below so that the estimated bearing capacity and associated settlements would be representative of what to be experienced by the foundations of the structures located on rock units.

As stated in PSAR section 2.6.4.2.8, USO assumed that both the weathered and the fresh rock units beneath the NI structures at the proposed facility would behave as a Hoek-Brown material so that the GSI) rock mass classification scheme can be used to predict the response of both rock units. However, currently available subsurface information on both rock units indicates that the Hoek-Brown model may not be suitable to model the response of the rock units at the proposed site as the conditions necessary to use the Hoek-Brown model are not satisfied, as summarized in PSAR section 2.6.4.2.8. For example, the rock mass must have several sets of discontinuities in different orientations (at least four distinct sets of discontinuities, as per Hoek and Brown, 1980) so that the rock mass behavior can be approximated by isotropic and homogeneous behavior assumed by the Hoek-Brown model. The staff noted that acoustic viewer logs recorded in eight boreholes at the proposed site show only one primary discontinuity set with dip direction/dip of 285°/20°, corresponding to the bedding planes, as discussed in PSAR section 2.6.4.2.7. In addition, although that staff observed many randomly oriented joints

(discontinuities) in the acoustic viewer logs, collectively they do not form a set or multiple sets with well-defined orientation features. In addition, their spatial extent is currently unknown from only observations at widely spaced eight borings at the proposed site. The random joints and bedding planes may plausibly form closely spaced intersecting fractures with spacing varying from 0.7 to 3 ft (0.2 m to 0.9 m) in the weathered rock unit, these random fractures are sparse in the fresh rock unit as can be seen in the rock cores recovered from the borings. This results in large discontinuity spacing of approximately 15 ft (4.6 m) of the fresh rock unit, as reported in PSAR section 2.6.4.2.7, which is of the same order of magnitude as the shaft (shaft diameter is 54 ft (16.4 m)) of the RXB. This large discontinuity spacing in the fresh rock unit violates another necessary condition of the Hoek-Brown model of close fracture spacing so that the response of the rock mass is controlled by the movement and rotation of the rock blocks separated by these intersecting discontinuities without any preferred failure direction. Presence of these large-spaced fractures forces preferred direction of rock mass failure.

The staff understands that the subsurface information currently available is from boreholes only as the Hilliard Shale does not have any outcrop at the site. As stated in PSAR section 2.6.4.5.1, USO wants to include the information of the subsurface fractures to develop a better understanding of the rock mass response by mapping the fractures visible on the excavation surfaces and stated that “[g]eologic mapping of the excavation, in conformance with RG 1.132, will be performed before placement of shotcrete, backfill materials, dental concrete, or foundation concrete. The geologic mapping program will include photographic documentation of the exposed surface and documentation of geologic features.”

To confirm that the exposed bedrock does not show fracturing or other geologic conditions that could affect the assumed rock mass classification value of the subsurface using the GSI scheme and applicability of the Hoek-Brown model to simulate the behavior when the excavations are complete and before the foundation is prepared, and to provide reasonable assurance that regulatory requirements and license commitments are adequately addressed during the construction of the Natrium facility, the staff recommends that the CP include the following permit condition:

USO shall perform detailed geologic mapping of the excavations for the safety-related engineered structures; examine and evaluate the geologic features discovered in those excavations; and, once geologic mapping information from the excavations for safety-related structures is available for examination by the NRC, notify the Director of the Office of Nuclear Reactor Regulation, or the Director’s designee, as specified in 10 CFR 50.4.

2.6.4.3 Lateral Earth Pressure

In PSAR section 2.6.4.10.3, USO discussed the estimated lateral earth pressure to be experienced by the permanent walls of the RXB, FHB, and NCB structures to be constructed below the grade. The staff has reviewed PSAR section 2.6.4.10.3 and audited NAT-7015. In addition, the staff has reviewed selected references available in the literature.

Current grade at the RXB and FHB locations are very close to the planned grade. However, [[

]].

As stated in PSAR section 2.6.4.10.3, USO estimated the lateral earth pressure to be experienced by the below-grade permanent walls of the RXB, FHB, and NCB structures as the summation of the static earth pressure and dynamic (seismic) earth pressure. The static earth pressure on a non-yielding wall is estimated using the at-rest earth pressure coefficient K_0 . The staff determined it acceptable as the soil exerting static lateral earth pressure on a non-yielding wall will be in the at-rest condition. USO used the Jaky's simplified formula to estimate K_0 . The staff notes that the Jaky's formula is appropriate for normally consolidated soils. Although samples may not indicate normally consolidated soils (the Over Consolidation Ratio for the two samples from depths comparable to the excavations of the SS structures within the NI area is 1.5 and 2.1), the staff determined that choice of lower friction angle makes the Jaky's simplified formula numerically not significantly different from the full formula; hence, the results are acceptable.

USO considered the hydrostatic pressure exerted by the groundwater against the buried wall. USO also considered the lateral earth pressure from the surcharge applied at the ground surface behind the top of the walls from construction of the facility. USO assumed a [[

]]. The staff determined these assumptions acceptable at this preliminary design stage.

USO estimated the [[

]], is similar to section 3.5.3.2(2) of ASCE 4-98. As the method used to estimate the lateral earth pressure follows the ASCE design standard and also the recommendations of section 3.8.4 of NUREG-0800, the staff determined that USO has used an acceptable method to estimate the dynamic lateral earth pressure on the below-grade walls of the RXB, NCB, and FHB of the proposed facility.

In the NI area, [[

]].

USO assumed that the buried permanent concrete walls of the buildings would be poured directly. These walls would experience lateral earth pressure from the weathered rock units. In the fresh rock unit, USO also assumed that lean concrete would be poured to backfill any gap between the excavated rock face and the concrete wall. The lateral pressure exerted on the wall

after each lift of the lean concrete would be negligible. The staff determined these assumptions are reasonable. The weathered rock unit is expected to behave more like a soil inducing lateral earth pressure. However, the fresh rock unit is a solid material and will exert only negligible force on the wall. The lean concrete when solidifies is also expected to behave like a solid material and exert only negligible force on the wall.

USO assumed the peak horizontal ground acceleration (PGA) (zero-period acceleration) to be 0.333 g at the proposed site, based on a seismic site response analysis for the RXB using generic soil profiles. Using actual soil profiles encountered at the proposed site, the horizontal PGA was estimated to be 0.304 g. Therefore, the staff determined that use of 0.50 g PGA is conservative.

2.6.4.4 Liquefaction Assessment

In PSAR section 2.6.4.8, USO discussed the potential for liquefaction of the subsurface materials adjacent to and beneath the safety-significant structures at the proposed facility. As stated in PSAR section 2.6.3.1.4, no evidence of paleoliquefaction was observed at the proposed site and its vicinity. Potential for liquefaction was assessed using the guidance in RG 1.198. Additionally, USO used the criteria proposed by Polito (1999) and Seed et al. (2003) to assess the liquefaction potential at the proposed site. The staff reviewed the assessment of potential liquefaction of the overburden soil at the proposed site and underneath the foundations of the safety-significant structures, described in PSAR section 2.6.4.8.

The SS structures at the proposed facility will be at various depths below the ground surface and will generally be constructed on weathered rock and fresh rock zones, except the grade slabs of the FHB, which would be constructed on compacted granular backfill. As stated in PSAR section 2.6.4.5.2, CLSM is planned to fill the annulus between the fresh rock zone and the foundation wall of the RXB shaft. In addition, compacted granular backfill is planned to be used as side fill of the RXB, FHB, and NCB. An alternative plan for construction of the RXB calls for no annulus between the excavation and the reactor structure and, consequently, no backfill would be needed to fill the gap. As stated in PSAR section 2.6.4.8, the backfill would be compacted to at least 95 percent of the modified Proctor maximum dry density following the industry-standard ASTM D1557 (ASTM, 2021). The HAA slab of the RXB would be constructed on the fresh rock zone 55 ft (17 m) below the finished grade. Additionally, the slab for the reactor cavity will be approximately 118 ft (36 m) below the finished grade on the fresh rock unit. The NCB will be constructed on the weathered rock zone at a depth of 33 ft (10 m) below the finished grade. As liquefaction cannot occur in rock or in concrete, USO concluded in PSAR section 2.6.4.8 that there will not be any potential for liquefaction below the safety-significant structures.

USO also assessed the potential for liquefaction of the overburden soil at the proposed site in PSAR section 2.6.4.8. As discussed in PSAR section 2.6.4.2.4, fine-grained and cohesive soils have been encountered at the proposed site. Using the criterion proposed by Polito (1999), as shown in PSAR figure 2.6-107, USO concluded in PSAR section 2.6.4.8 that the overburden soil at the proposed site will also not be liquefy.

The staff reviewed the assessment of liquefaction potential of the proposed site presented in PSAR section 2.6.4.8. In addition, the staff conducted confirmatory analysis with the criteria presented in RG 1.198, Seed et al. (2003), and Bray and Sancio (2006). Although the

staff's review of the potential for liquefaction is generally limited to the NI area where the safety-significant structures would be constructed, the staff also reviewed the liquefaction potential outside the NI area as these areas are adjacent. Any liquefaction outside the NI area may impose additional lateral stresses on the SR structures in the NI area and access to the SR structures may become difficult after a liquefaction event outside the NI area.

The staff determined that all the safety-significant structures at the proposed facility would be constructed at various depths below the finished grade. These structures would generally be constructed on weathered rock zones, and fresh rock units, except the grade slabs of the FHB. The grade slab would be constructed on compacted granular backfill. As the backfill would be compacted to at least 95 percent of the modified Proctor maximum dry density following an industry-standard ASTM code, as stated in PSAR section 2.6.4.5.2, the staff determined that the backfill would not be liquefiable. Therefore, the staff determined that the subsurface (both weathered and fresh rock units including granular backfill) below the foundation of the safety-significant structures of the proposed facility would not liquefy; however, the potential for liquefaction of the soil is evaluated below.

Residual soils at the site are derived from in situ weathering and decomposition of the underlying bedrock. The overburden soils are primarily residual soils with some alluvium/colluvium. Alluvium/colluvium soils are reworked sediments derived from weathering of the underlying residual soils and bedrock Hilliard shale. They were locally transported and deposited at the proposed site through streamflow, slope wash, gravity, and wind or similar processes.

The Hillard shale is of the Upper Cretaceous (145 to 60 million years ago) age, which is significantly older than the Holocene (last 11,700 years) age. PSAR section 2.5.3.1.1 states that the alluvium/colluvium at the proposed site is of Holocene to Pleistocene age. Recent Holocene soil deposits are generally susceptible to liquefaction, as per RG 1.198, although a few observed liquefaction cases has been observed in Pleistocene and even in Pre-Pleistocene deposits. Therefore, potential for liquefaction cannot be eliminated based on age of the deposits alone.

RG 1.198 cautions against very loose types of these soils and extremely loose collapsible soils as they are prone to liquefaction. USO stated that uncompacted fills and eolian sands or silts were not identified at the proposed site. In addition, the staff determined that USO did not observe any evidence of paleoliquefaction in the vicinity of the site, as stated in PSAR section 2.6.3.1.4.

The staff determined that the proposed site has approximately 20 ft (6 m) of overburden soil (silty or silty/sand clay) overlying approximately 30 ft (9 m) of weathered rock. This weathered rock overlies the fresh siltstone of the Hilliard shale formation, which is at a depth of approximately 50 below the surface. The staff notes that the groundwater elevation at the north end of the proposed site in the area of the NI is approximately 6,727 ft (2,050 m) (30 ft (9 m) below the finished grade) and 6,722 ft (2,049 m) (35 ft below the finished grade) at the south end after construction. Therefore, the overburden soil will be unsaturated after construction, which deters liquefaction, as per RG 1.198.

The staff also determined that USO tested total 100 samples obtained at various locations within the proposed site, as illustrated in PSAR figure 2.5-107. The staff notes that the average

corrected SPT blow count $(N_1)_{60}$ value of 82, which is high for liquefaction to occur; however, there are a few samples with very low blow counts, $(N_1)_{60} < 20$. These low values indicate that a further assessment of the liquefaction potential is needed.

The staff also determined that majority of the samples from the proposed site are classified as clay with high fines content, greater than 30 percent. Most of the samples are classified as Lean Clay, inorganic clay with low to medium plasticity, as per the Unified Soil Classification System (USCS) classification. Only a few samples are classified as inorganic clay with high plasticity or fat clay. Outside the NI area, two samples are classified as inorganic silts and very fine sands (ML), one sample each as clayey gravel (GC), silty sand (SM), and clayey sand (SC), following the USCS classification.

The staff notes that no sands with dual USCS classification have been observed at the proposed site. Clay content has been reported on 18 soil samples, out of 100 samples, used in assessing the liquefaction potential. All of these 18 samples show clay content greater than 15 percent and the Liquid Limit (LL) greater than 35 percent with natural water content less than 90 percent. Therefore, based on RG 1.198, these 18 samples are not liquefiable.

Properties of the soil at the proposed site show that the samples contain fines greater than 30 percent and the soil in these samples is classified as clay, mostly Lean Clay. However, Plasticity Index (PI) of these samples is not always greater than 30 percent. Therefore, based on the qualitative criteria given in RG 1.198, these samples cannot be screened out for not liquefiable. In addition, five samples contain gravelly soils and need further assessment for liquefaction potential. All these samples were further evaluated for liquefaction potential using other measured properties.

As the qualitative criteria given in RG 1.198 could not definitely determine whether the proposed site has a potential for liquefaction, the staff used two quantitative criteria for the assessment: criteria proposed by Seed et al. (2003) and Bray and Sancio (2006). The staff did not use the criterion proposed by Polito (1999) as the criterion proposed by Seed et al. (2003) is improvement of the Polito criterion.

The staff determined that majority of the 100 samples fall within “Zone B” of Seed et al. (2003) in the PI versus LL plot, only a few samples fall within “Zone A” and rest of the samples fall outside of “Zone B.” Samples falling outside “Zone B” are not liquefiable, as per Seed et al. (2003). As the moisture content w_c of all samples within “Zone A” is less than 80 percent of LL , they also are not liquefiable. The staff also determined that all samples in “Zone B” are also not liquefiable as their moisture content is less than 85 percent of LL following Seed et al. (2003). Therefore, the staff concludes, based on Seed et al. (2003), that all 100 samples are not liquefiable. Using also the Bray and Sancio (2006) criterion, the staff determined that all 100 samples are not liquefiable as they fall in the “Not Susceptible” zone (i.e., w_c/LL is less than 80 percent) in the PI versus w_c/LL plot. Consequently, the staff determined that the overburden soil in the NI area and its surrounding at the proposed site is not liquefiable.

Therefore, based on the preceding discussion of the staff’s review, the staff determined that the safety-significant structures within the NI area will not experience liquefaction as they would be constructed on rock or engineered granular fill or concrete or non-liquefiable soils. In addition, the staff determined that the overburden soils surrounding the NI area are also not susceptible to liquefaction.

2.6.5 Stability of Slopes

As stated in PSAR section 2.6.5.1, no permanent slopes are planned at the proposed facility in proximity to the safety-significant structures in the NI area. This area will be graded level with a gentle sloping away from the safety-significant structures for drainage. As shown in PSAR figure 2.6-113, although temporary slopes are planned for the excavations in the NI area during construction of the foundations of the safety-significant structures. These excavations will be in the overburden soil and weathered and fresh rock units. The temporary vertical cut surfaces of the excavations will be adequately supported to prevent failure. Vertical cuts will be supported with tied-back sheet piles or soldier piles and lagging walls. USO stated that these slopes will have two (horizontal) to one (vertical) inclination and the excavations will adhere to the regulations from the Occupational Safety and Health Administration (OSHA), in accordance with 29 CFR 1926, "Safety and Health Regulations for Construction."

The staff determined that the permanent slopes would be for stormwater outside the NI area and the temporary slopes would be part of the excavation sequences. Consequently, the staff did not review the slope stability aspects during construction at the proposed site as none of the slopes are safety-significant.

2.6.6 Conclusion

Based on its findings above, the staff concludes that the information on geology, seismology, and geotechnical characteristics in PSAR section 2.6, as supplemented, is sufficient and meets the applicable guidance and regulatory requirements identified in this SE section for the issuance of a CP in accordance with 10 CFR 50.35 and 50.40.

2.7 Volcanic Hazards

Section 2.7 of the of the PSAR incorporates by reference the volcanic hazard assessment documented in topical report NAT-3226-A, "An Analysis of Potential Volcanic Hazards at the Proposed Natrium Site near Kemmerer, Wyoming," (ML24303A409). The assessment was performed because Quaternary volcanic sources are present within 320 km of the KU1 site.

Section 2.7.1 of DANU-ISG-2022-02 provides the applicable guidance for the review. The need to consider volcanic hazards is determined by information gathered during the site characterization process required under 10 CFR 100.23(c). An additional assessment of potential volcanic hazards is indicated by either (1) a Quaternary volcano within 320 km (200 mi) of the proposed site, or (2) a volcanic deposit within 40 km (25 mi) of the proposed site, from a Quaternary volcano located more than 320 km (200 mi) away. The NRC developed RG 4.26, "Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites," Rev. 1, (ML23167A078) to provide an acceptable risk informed framework for the consideration of volcanic hazards in licensing new reactors.

The staff's evaluation of the volcanic hazards at the KU1 site, specifically the maximum thickness of tephra that may reach the site, is contained in the SE for topical report NAT-3226-A. In that SE, the staff identified two limitations and conditions to be addressed in the review of any licensing submittal that references the TR.

Limitation 1

“The conclusions reached in this SE do not address the content provided in section 10 of the TR. Thus, any licensee or applicant referencing this TR must evaluate specific design, mitigation or monitoring actions required to mitigate the effects of volcanic hazards at the site, including any monitoring requirements for notification of impending volcanic events.”

Limitation 2

“The conclusions reached in this SE do not address the impacts of the calculated probabilities of volcanic hazard events on the cumulative plant risk. Thus, any licensee or applicant referencing this TR should evaluate the effect of volcanic hazards on the overall plant risk.”

In PSAR section 3.10, the applicant states that mitigation of beyond design basis events will be addressed at the OL stage. Accordingly, the staff will review any necessary mitigation actions to be taken in advance of a tephra fall at the site during the OL review because tephra fall was identified by the volcanic hazard assessment as a credible beyond design basis event.

2.7.1 Conclusion

The applicant incorporated by reference the volcanic hazards assessment provided in topical report NAT-3226-A. Accordingly, the staff incorporates by reference the conclusions in the SE for the subject topical report. The staff further concludes that the potential mitigation of volcanic hazards can be deferred to the OL stage when final detailed design information is available to determine whether mitigation is necessary.

2.8 Summary and Conclusions on Site Characteristics

The staff evaluated the descriptions and discussions of the proposed KU1 site characteristics, as described in chapter 2 of the PSAR and determined that the information on KU1 site characteristics: (1) provides reasonable assurance that the final design will conform to the design basis, (2) meets all applicable regulatory requirements, and (3) meets the applicable acceptance criteria in DANU-ISG-2022-02. Based on these determination and subject to the conditions referenced above, the staff makes the following conclusions regarding the issuance of a CP in accordance with 10 CFR 50.35 and 50.40:

- USO described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.
- Such further technical or design information as may be required to complete the safety analysis of the site characteristics, and which can reasonably be left for later consideration, will be supplied in the FSAR.
- There is reasonable assurance that, taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

- The issuance of a permit for the construction of the facility would not be inimical to the common defense and security or to the health and safety of the public.

2.9 References

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3 LICENSING BASIS EVENTS

3.1 Probabilistic Risk Assessment

3.1.1 Introduction

Section 3.1 of the Kemmerer Power Station Unit 1 (KU1) preliminary safety analysis report (PSAR) describes the KU1 probabilistic risk assessment (PRA) and the use of PRA as the primary tool for implementing the risk-informed, performance-based methodology established under the Licensing Modernization Project (LMP). This methodology is outlined in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision (Rev.) 1 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19241A472), and is endorsed with clarifications by the U.S. Nuclear Regulatory Commission (NRC) in Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Rev. 0 (ML20091L698).

The KU1 PRA serves as a key input to several elements of the LMP framework, including:

- Identification of licensing basis events (LBEs);
- Classification of structures, systems, and components (SSCs), along with the determination of their associated special treatment requirements;
- Derivation of design basis accidents (DBAs); and
- Evaluation of the adequacy of defense-in-depth (DID) measures.

The KU1 PRA is also used to evaluate overall plant risk against the following risk evaluation criteria described in section 3.2.2, step 7b of NEI 18-04 and addressed in PSAR section 4.1:

- The total frequency of events resulting in a site boundary dose exceeding 100 mrem should not exceed 1/plant-year;
- The average individual risk of early fatality within 1 mile of the exclusion area boundary (EAB) should not exceed 5×10^{-7} /plant-year; and
- The average individual risk of latent cancer fatality within 10 miles of the EAB should not exceed 2×10^{-6} /plant-year.

As part of its safety review, the staff evaluated how USO developed the construction permit (CP) stage PRA, used it to develop the preliminary KU1 licensing basis following the LMP approach, and applied specific risk insights throughout the PSAR. The staff followed the guidance in Appendix A of RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Rev. 0 (ML23269A222) to review the KU1 CP stage PRA within the context of the LMP framework. Appendix A provides guidance on the expected scope, level of detail, and quality of a PRA suitable for the CP stage,

recognizing that the plant design is still evolving and that the PRA model is inherently limited in maturity. The staff's review of PRA information in the PSAR and supporting PRA documentation through regulatory audit (ML25302A450) were therefore conducted with a focus on assessing whether the PRA provides a technically sound foundation to support LMP-related activities, specifically, the identification and evaluation of LBEs, classification of SSCs according to safety significance, and establishment of adequate DID. At the CP stage, the PRA's role is to preliminarily demonstrate compliance with RG 1.233 and Appendix A of RG 1.253, thereby providing a provisional basis for assessing risk acceptance criteria (e.g., quantitative health objectives (QHOs), frequency-based thresholds for LBEs), and to inform early-stage design and licensing decisions through systematic risk insights, given that uncertainties related to parameter, model, and completeness have not yet been fully characterized or assessed.

Because the KU1 PRA is still preliminary and currently addresses only internal initiating events (IEs), the staff applied applicable positions in RG 1.247, "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities," for Trial Use (ML21235A008), which endorses, with clarifications, the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) RA-S-1.4-2021 non-light-water reactor (non-LWR) PRA standard, to evaluate the technical adequacy and completeness of each PRA element performed. This detailed element-by-element review allowed the staff to determine whether key modeling aspects (e.g., IE identification, system success criteria, failure data, human reliability, quantification methods) were developed with sufficient technical justification and documentation to support their intended use. The staff examined the assumptions, limitations, and uncertainties in each element to ensure traceability and to verify that the PRA, while limited in scope and development, can credibly support risk-informed decision-making consistent with the LMP framework. The staff focused on whether the CP stage PRA, though preliminary, provides a reasonable technical basis for ongoing design development and establishes a foundation for future updates and expanded scope at the operating license (OL) stage.

Section 3.1 of the PSAR summarizes the KU1 CP stage PRA and serves as a foundation for how the PRA results and insights are linked to safety analysis and design evaluation throughout the PSAR. Specifically, this section presents the PRA overview, key modeling assumptions, and a summary of PRA results essential to the licensing basis under the LMP framework. The information in section 3.1 supports the risk-informed identification of LBEs and provides the quantitative basis for determining safety-significant functions, establishing DID, and defining reliability and capability targets for important SSCs. Section 3.1 of the PSAR summarizes the KU1 CP stage PRA and serves as a foundation for how the PRA results and insights are linked to safety analysis and design evaluation throughout the PSAR. Specifically, this section presents the PRA overview, key modeling assumptions, and a summary of PRA results essential to the licensing basis under the LMP framework. The information in section 3.1 supports the risk-informed identification of LBEs and provides the quantitative basis for determining safety-significant functions, establishing DID, and defining reliability and capability targets for important SSCs.

The PRA insights and results described in PSAR section 3.1 are interwoven throughout the PSAR and provide inputs across several PSAR chapters and sections that apply risk information to specific aspects of the KU1 design and safety assessment. For example, section 4.1 incorporates PRA results into the overall assessment of plant-level risk, while section 4.2 evaluates key uncertainties and assumptions as part of the integrated decision-making process and assesses the adequacy of DID. Section 5.1 of the PSAR applies PRA insights to the classification of SSCs. Similarly, section 5.2 of the PSAR identifies the

safety-significant PRA safety functions (PSFs), including safety-related (SR), non-safety-related SSCs with special treatment (NSRST), and non-safety-related with no special treatment (NST) functions derived from PRA results. Section 6.2 uses the PRA data to establish reliability and capability targets. Section 7.1 of the PSAR applies PRA-derived safety functions to define functional requirements and design criteria, and PSAR chapter 11 uses PRA insights to support analyses related to main control room operations and human factors engineering. Collectively, these interconnections demonstrate that the PRA described in section 3.1 is central in supporting development of the KU1 LMP licensing basis, ensuring that the design, classification, and operational evaluations are systematically informed by quantitative risk insights and aligned with the overall risk-informed licensing approach.

3.1.2 Regulatory Evaluation

The applicable regulatory requirements at the CP application stage related to the use of PRA for evaluating accident probabilities, plant operational safety, and risk to public health are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, “Contents of applications; technical information,” paragraph (a), “Preliminary safety analysis report,”; and
- 10 CFR 50.35, “Issuance of construction permits”

The relevant guidance documents related to the use of PRA for this CP application include:

- RG 1.233 which endorses NEI 18-04;
- RG 1.247 for Trial Use, which endorses the non-LWR PRA standard ASME/ANS RA-S-1.4-2021; and
- RG 1.253, which endorses NEI 21-07, “Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology,” Rev. 1 (ML22060A190).

3.1.3 Technical Evaluation

The staff considered the applicable regulations and guidance identified in safety evaluation (SE) section 3.1.2 above to determine whether the KU1 PRA was developed using generally accepted methods and reflects an appropriate level of technical rigor to support its intended use at the CP stage.

Specifically, the staff evaluated PSAR section 3.1 against:

- RG 1.233;
- RG 1.253; and
- RG 1.247.

These guidance documents outline acceptable practices for PRA development and application within the framework of the LMP and provide expectations for PRA scope and level of detail in support of a CP application. In particular:

- Section A.3 of appendix A to RG 1.253 defines the expected PRA scope at the CP stage; and
- Section A.5 of appendix A to RG 1.253 specifies the level of detail and content to be included in the PSAR.

RG 1.253 indicates that the PRA description submitted with a CP application be a high level overview or summary that addresses PRA quality, scope, intended applications, and overall acceptability. The applicant is expected to demonstrate that the PRA was developed in a manner commensurate with the maturity of the design, and that the PSAR includes sufficient information to support the CP-level safety determinations. The PSAR should also include any necessary commitments to upgrade and maintain the PRA so that its completion status at the OL is consistent with its intended uses.

The staff assessed the acceptability of the KU1 PRA by evaluating the following four key areas in an integrated manner:

1. PRA scope – this includes:
 - The risk metrics used;
 - The radiological source terms considered;
 - The internal event causes (hazard groups) analyzed;⁹ and
 - The plant operating states (POSS) evaluated.
2. Level of detail – the extent of modeling necessary to capture plant behavior, system interactions, and interdependencies with sufficient resolution to support intended applications.
3. PRA elements – the technical analyses that support the development and quantification of the PRA model, consistent with its intended use at the CP stage.
4. Plant representation and configuration control – the degree to which the PRA accurately reflects the current design and the mechanisms in place to manage configuration changes as the design evolves.

In accordance with RG 1.253 the KU1 PRA is described throughout several chapters of the PSAR, reflecting its intended use during the design and construction phases:

⁹ As described in Appendix A of RG 1.253, an applicant for a CP may disposition certain hazards by crediting design basis hazard levels (DBHLs) in lieu of explicitly modeling these hazards in the PRA or accounting for them through a risk-informed supplementary evaluation.

- Chapter 3 provides a high level overview of the PRA model and presents the set of LBEs, which are derived from and informed by PRA event sequence families.
- Chapter 4 presents the integrated risk results and compares them to the three cumulative plant performance criteria outlined in NEI 18-04.
- Chapter 5 identifies the preliminary set of PSFs, including those fulfilled by both SR and NSRST SSCs. The chapter also documents risk-significant SSCs credited for meeting frequency-consequence (F-C) targets and cumulative risk criteria.
- Chapter 6 discusses the reliability and capability targets for SR and NSRST SSCs and describes how PRA results and insights informed the development of these targets and their associated special treatments.

As part of its review, the staff conducted both a virtual audit and a three-day in-person audit focused on the PRA and its supporting documentation. These audits provided the staff with access to PRA materials and facilitated direct discussions with the applicant to clarify technical issues. The staff's audit included the KU1 PRA model that the applicant developed using the Computer Aided Fault Tree Analysis System (CAFTA) software. In this audit area, the staff focused on the model structure, development methodology, and quantification. The staff also conducted an audit of selected sensitivity analyses to assess the robustness of the plant design at the CP stage and to better understand the uncertainties inherent in the PRA modeling.

As discussed below, based on its review of the PSAR, as confirmed in the audit, the staff determined that the KU1 PRA was developed in a manner consistent with applicable guidance and in reasonable conformance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as endorsed by RG 1.247. The KU1 PRA demonstrates that the risk associated with the KU1 site and preliminary Sodium design is acceptably low and provides meaningful risk insights in support of implementing the LMP methodology described in NEI 18-04.

The staff's evaluations and conclusions regarding the PRA-related content presented in other sections of the PSAR are documented in the corresponding sections of the staff's SE. This section focuses specifically on the scope, level of detail, and quality of the PRA information provided in chapter 3 of the PSAR.

Overall, the staff determined that USO followed the guidance provided in RG 1.253 and addressed the following technical elements¹⁰ of the PRA at a level sufficient for the CP stage:

- Element 1: Plant Operating Status (POS) Analysis;
- Element 2: Initiating Event Analysis;
- Element 3: Event Sequence Analysis;
- Element 4: Success Criteria Analysis;
- Element 5: Systems Analysis;

¹⁰ RG 1.253, appendix A, section A.4 indicates a minimally acceptable PRA for a non-LWR CP application based on the LMP methodology addresses PRA elements 2-12.

- Element 6: Human Reliability Analysis;
- Element 7: Data Analysis;
- Element 8: Hazard Screening Analysis;
- Element 9: Event Sequence Quantification;
- Element 10: Mechanistic Source Term Analysis;
- Element 11: Radiological Consequence Analysis; and
- Element 12: Risk Integration.

USO performed a self-assessment to ensure the KU1 PRA conformed to these twelve elements in accordance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021. USO stated in PSAR section 3.1.1.2 that its self-assessment was performed following the guidance in NEI 20-09, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard," Rev. 1 (ML20302A115), which is also endorsed in RG 1.247. A summary of the staff's review of USO's self-assessment is provided later in this section. Unlike the surrogate risk metrics commonly used in LWR PRAs, such as core damage frequency and large release frequency, the KU1 CP application employs risk metrics focused on offsite radiological consequences from event sequences involving the release of radioactive material. These consequence-based metrics are aligned with those used in LWR level 3 PRAs but exclude consideration of radiological or other impacts on on-site personnel.- At this stage of design development, the applicant evaluated the following radionuclide source terms in the KU1 PRA:

- Reactor enclosure system, including spent fuel in in-vessel storage
- Primary sodium processing system and sodium cover gas system
- Intermediate sodium processing system
- Gaseous radiological waste system
- Ex-vessel handling machine
- Ex-vessel storage tank
- Spent fuel processing systems, including:
 - Bottom loading transfer cask
 - Pin removal cell
 - Pool immersion cell
 - Dry cask loading

- Spent fuel pool.

ASME/ANS RA-S-1.4-2021, as endorsed, with exceptions, by RG 1.247, provides guidance for PRA radionuclide source terms for non-LWRs. The staff reviewed the PSAR and audited supplemental materials to confirm and understand the information in the PSAR. Based on its review, the staff determined these source terms to be reasonable and sufficient for the CP stage PRA because they were developed with approved methods and are consistent with the ASME/ANS RA-S-1.4-2021, as endorsed, with exceptions, by RG 1.247. More details of the staff's review follow.

PRA Element 1 – POS Analysis

As described in RG 1.247, the objective of the POS analysis is to identify plant operating conditions such as full-power operation and low-power and shutdown (LPSD) conditions that are important to risk, and to segment them into distinct states in which plant parameters are assumed to remain relatively constant.

PSAR section 3.1.1.6 summarizes the scope of USO's POS analysis. To further understand the description in the PSAR, the staff audited NAT-7109, "Natrium PRA Plant Operating States (POS) Analysis," Rev. A. The audit focused on the approach used to perform the POS analysis, ensuring that the analysis is generally consistent with accepted industry practices, the modeling assumptions and inputs are reasonable, and the POS analysis is consistent, to the extent possible, with the preliminary plant design.

Specifically, the audit concentrated on the methods used to segment full-power and low-power and shutdown conditions into distinct, relatively stable states that support comprehensive risk evaluation. The staff examined how plant evolutions were identified, how POSs were defined for each evolution, and how plant configurations affecting radionuclide release barriers and key safety functions, such as reactivity control, heat removal, and confinement, were characterized. The staff also reviewed the basis for consolidating operating states, the rationale for the POSs defined for the KU1 PRA, and the underlying assumptions and uncertainties. The staff also gave particular attention to the use of design stage information, expected operational practices, and outage schedules in establishing representative plant conditions. The review confirmed that the analysis followed a structured approach consistent with the current design information and non-LWR PRA standard, with sufficient documentation of assumptions, sources of uncertainty, and key plant attributes relevant to each POS.

Based on its review, the staff determined the POS analysis to be reasonable because the information in PSAR section 3.1.1.6 describing the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance applicable to the CP stage, including the key items listed below:

- Identified POSs represent distinct and relatively stable plant conditions and were reasonably defined.
- LPSD evolutions were segmented into separate POSs based on differences in plant response to IEs.
- Each POS was characterized using available design information.

- Sources of model uncertainty associated with the POSs were identified and described.
- Key plant condition attributes relevant to each POS were documented.
- The POSs analysis was documented with sufficient clarity to ensure traceability of the technical basis.

As KU1 progresses from design to operation, the POS analysis and associated assumptions will be periodically revised and refined using updated and more detailed plant information. The staff will review the final POS analysis at the OL stage to ensure it is acceptable and properly performed to support the LMP application, should the applicant apply for an OL.

PRA Element 2 – Initiating Event Analysis

As described in RG 1.247, the objective of the IE analysis is to identify and characterize events that could challenge safe plant operation during any POS and require effective mitigation by plant equipment and personnel to prevent or limit a release of radioactive material. This includes both events that may occur at the plant and those with a credible probability of occurrence.

PSAR table 3.1-2 and table 3.1-3 summarize the hazards that were either screened out or into the PRA, respectively, as well as those addressed through alternative approaches such as DBHL framework. While the OL stage PRA should encompass all plant modes, hazards, POSs, and radionuclide sources, some PRA elements, such as other hazards, were not completed at the CP stage. PSAR section 3.1.1.7 describes the internal IEs that the applicant considered for all sources of radionuclides and POSs that are identified in PSAR section 3.1.1.3 and section 3.1.1.6, respectively. The KU1 IE analysis focused on events that could impact the reactor core and core components within the reactor enclosure system (RES), including in-vessel storage (IVS), core assemblies transferred to the ex-vessel storage tank (EVST) during refueling outages, and radionuclide sources associated with systems directly connected to the RES during reactor operation, such as the sodium processing system (SPS) and sodium cover gas (SCG) system. Ex-vessel fuel handling events were also identified and included in the PSAR.

To further understand the description in the PSAR, the staff conducted an audit that included NAT-7127, “Natrium PRA Initiating Events (IE) Analysis,” Rev. A. The audit focused on confirming the characteristics and attributes of the IE analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247. The audit examined the approach and method applied in the IE analysis, ensuring they are reasonable and consistent with accepted industry practices; the staff also confirmed during the audit that the assumptions are reasonable, and the analysis aligns with the preliminary plant design to the extent possible.

Specifically, the audit focused on the systematic approach used to develop a comprehensive list of potential internal IEs across all POSs, including those affecting the reactor core, in-vessel and ex-vessel fuel storage, and connected systems such as the sodium processing and cover gas systems. The staff examined how USO used multiple data sources, such as NUREG/CR-6928 (ML070650650), and relevant non-LWR and sodium fast reactor PRAs, to ensure extensiveness, and how system-level design reviews were applied to identify plant-specific initiators.

The audit also included the methods used to group similar IEs and estimate their frequencies using generic data supplemented with design-specific information. The staff also gave particular attention to the assumptions, data sources, and treatment of parametric and modeling uncertainties. The staff's audit identified that the IE identification, grouping, and frequency estimation processes were conducted in a structured and traceable manner consistent with non-LWR PRA standard. The staff's audit verified that the analysis provided reasonable coverage of potential initiators, appropriately reflected current design information, and identified key sources of uncertainty for follow-up and refinement at the OL stage.

Based on its review, the staff determined the IE analysis to be reasonable because information in PSAR tables 3.1-2 and 3.1-3 and section 3.1.1.7 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance applicable to the CP stage, including the following:

- The analysis provides detailed identification and characterization of IEs suitable for the design stage.
- IEs were logically grouped based on similarity in mitigation requirements.
- Many of the IE frequencies were reasonably estimated.
- Sources of model uncertainty related to the IE analysis were identified and qualitatively characterized.
- The IE analysis documentation offers traceability of the methodology employed and most of the results obtained.

As KU1 advances from design to operation, as stated in the PSAR, the IE analysis and its underlying assumptions will be updated and refined using increasingly detailed plant information. The staff will review the final IE analysis at the OL stage to ensure it is acceptable and has been properly performed to support the LMP application, should the applicant apply for an OL.

PRA Element 3 - Event Sequence Analysis

As described in RG 1.247, the objective of the event sequence (ES) analysis is to model, in chronological order to the extent practical, the potential progressions of events following the IE. These event sequences lead either to successful prevention or to the release of radioactive material. At the OL stage, the analysis will account for the availability of plant systems and operator actions based on defined success criteria and established plant operating procedures, including emergency and abnormal operating procedures.

The KU1 ES analysis, which encompasses both at-power and LPSD conditions, is described in PSAR sections 3.1.1.8 and 3.1.1.14. PSAR section 3.1.1.14 describes that the USO performed the ES modeling and quantification using the Electric Power Research Institute Phoenix Architect 2.0, which includes CAFTA, PRAQuant, UNCERT, and FRANX, and FTREX Version 1.8. This analysis integrates POSS, IEs, safety functions, and the success or failure of SSCs, culminating in end states that may involve potential release of radioactive material. The primary output of the ES analysis is a set of event trees, which delineate possible event

progressions. Each event tree represents a time-independent, system-level response to a specific IE. The results provide essential inputs for plant risk quantification.

To further understand the description in the PSAR, the staff conducted an audit that included NAT-7154, "Natrium PRA Event Sequence Analysis (ES)," Rev. A. The audit focused on confirming the characteristics and attributes of the ES analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247. The audit examined the approaches and methodologies used in the ES analysis, ensuring they are reasonable, technically defensible, and consistent with established industry practices; the staff also confirmed in audit that assumptions are reasonable and the analysis reflects a level of detail suitable for the preliminary plant design.

Specifically, the audit concentrated on the modeling of event progressions following the initiation of an IE for both at-power and LPSD conditions. The audit focused on the development of event trees (ETs) representing plant responses to various IEs, the identification of key radionuclide release barriers, and the integration of safety functions supporting reactivity control, core flow, primary sodium heat removal, and confinement. The staff examined how the applicant defined fuel damage, the associated end states, and the mission times applied to active and passive heat removal sequences. The staff's audit also examined the treatment of short-term reactivity and long-term decay heat removal sequences and the role of passive systems, such as the reactor air cooling (RAC) system, in maintaining safe plant conditions.

The audit further included the ES analysis for LPSD conditions, including ETs addressing fuel handling and sodium leak scenarios within the reactor vessel, ex-vessel handling machine, and EVST. The staff assessed the use of design stage assumptions, treatment of modeling and completeness uncertainties, and the documentation of success criteria for key safety functions. The staff's audit confirmed that the ES analysis was structured and traceable, reflected the functional dependencies influencing event progression, and followed the non-LWR PRA standard for defining barriers, success criteria, and mission times.

Based on its review, the staff determined the ES analysis to be reasonable because information in PSAR sections 3.1.1.8 and 3.1.1.14 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance applicable at the CP stage, including the following:

- The barriers to radionuclide release and corresponding safety functions required to protect each barrier were reasonably defined and described for each source of radioactive material and POS.
- The ES analysis reflected functional dependencies influencing event progression.
- Individual function success criteria and associated mission times were considered.
- Relevant sources of modeling uncertainty related to the ES analysis were identified and characterized.
- The ES analysis was sufficiently documented to ensure traceability of the underlying technical basis.

As KU1 advances from design to operation, the ES analysis and its supporting assumptions will be updated based on the final design. The staff will review the final ES analysis at the OL stage to ensure it remains appropriate and has been properly performed to support the LMP application, should the applicant apply for an OL.

PRA Element 4 - Success Criteria Analysis

As described in RG 1.247, the objective of the success criteria (SC) analysis is to establish the minimum functional requirements for each key safety function and the systems that fulfill these functions to prevent or mitigate radioactive release following an IE.

PSAR section 3.1.1.9 describes the CP application's SC analysis. To further understand the description in the PSAR, the staff conducted an audit that included NAT-7580, "Natrium PRA Success Criteria Analysis (SC)," Rev. A. The audit focused on confirming the characteristics and attributes of the SC analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247. The audit assessed the SC analysis approach, confirming general reasonableness and that the assumptions are appropriate, and the analysis aligns with the preliminary plant design.

Specifically, the audit concentrated on the minimum functional requirements for SSCs necessary to maintain critical safety functions during full-power and LPSD conditions. The audit focused on the methods and assumptions used to establish performance requirements for key safety functions, reactivity control, core flow, primary sodium heat removal, and confinement, and the corresponding SSCs that maintain radionuclide barriers such as the fuel matrix, cladding, sodium coolant, reactor vessel, and confinement structures. The staff assessed how both active and passive systems were treated in the analysis, including the determination of SC for active systems and the engineering basis for passive system performance.

The audit also included the assumptions and uncertainties to confirm they are reasonable for the CP stage, given the preliminary nature of the design. The audit emphasized the treatment of modeling and completeness uncertainties and the traceability of the analytical bases supporting the SC. Overall, the staff's audit focused on ensuring that the SC analysis appropriately reflects the KU1 design's capability to fulfill essential safety functions and that the approach aligns with the non-LWR PRA standard for the current stage of design maturity.

Based on its review, the staff determined the SC analysis to be reasonable because information in PSAR section 3.1.1.9 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance that is applicable at the CP stage, including the following.

- The SC analysis addressed fundamental safety functions, supporting SSCs, and operator actions.
- Relevant sources of model uncertainty were identified.
- The SC analysis was documented with clarity to provide traceability of the technical basis at a level sufficient for the CP stage.

With the applicant conforming to the non-LWR PRA standard, as mentioned in PSAR section 3.1.1.1, additional documentation of the SC for each safety-significant PSF reflecting the

final design, including quantification and uncertainty characterization will be established at the OL stage. The staff will review the final SC analysis results at the OL stage to verify its acceptability and that it was properly performed to support the LMP application, should the applicant apply for an OL.

PRA Element 5 - Systems Analysis

The objective of the systems analysis (SY) is to identify and model the various combinations of failures that could prevent a system from fulfilling its functions as defined by the SC. The system model incorporates hardware, instrumentation, and their associated failure modes, as well as human failure events (HFEs) that could impair system performance. As described in RG 1.247, basic events representing equipment and HFEs should be developed in sufficient detail to capture dependencies among systems and to highlight specific equipment or human actions that significantly impact the system's ability to perform its intended functions.

PSAR section 3.1.1.10 describes the CP application's SY analysis. To further understand the description in the PSAR, the staff conducted an audit that included the following PRA system notebooks:

- NAT-7137, "Natrium PRA System Analysis – Containment," Rev. A,
- NAT-7138, "Natrium PRA System Analysis – Control (RPS/NIC)," Rev. A,
- NAT-7139, "Natrium PRA System Analysis – Electrical," Rev. A,
- NAT-7140, "Natrium PRA System Analysis – Ex-Vessel Fuel Handling Machine (EVHM)," Rev. A,
- NAT-7141, "Natrium PRA System Analysis – Ex-Vessel Storage Tank (EVST) Cooling," Rev. A,
- NAT-7142, "Natrium PRA System Analysis – Gaseous Radwaste Processing (RWG)," Rev. A,
- NAT-7143, "Natrium PRA System Analysis – Inherent Reactivity Feedback Feature (IRF)," Rev. A,
- NAT-7144, "Natrium PRA System Analysis – Intermediate Air Cooling (IAC)," Rev. A,
- NAT-7145, "Natrium PRA System Analysis – Intermediate Heat Transport System (IHT)," Rev. A,
- NAT-7146, "Natrium PRA System Analysis – Miscellaneous," Rev. A,
- NAT-7147, "Natrium PRA System Analysis – Molten Salt and Steam Generation System (MSS/SGS)," Rev. A,
- NAT-7148, "Natrium PRA System Analysis – Nuclear Island (NI) Air and Inert Gas Distribution System (NGS)," Rev. A,

- NAT-7149, “Natrium PRA System Analysis – Primary Heat Transport (PHT),” Rev. A,
- NAT-7150, “Natrium PRA System Analysis – Reactor Control (CRD),” Rev. A,
- NAT-7151, “Natrium PRA System Analysis – Reactor Air Cooling (RAC),” Rev. A,
- NAT-7152, “Natrium PRA System Analysis – Sodium Cover Gas (SCG),” Rev. A,
- NAT-7153, “Natrium PRA System Analysis – Sodium Processing System (SPS),” Rev. A.

The audit focused on confirming the characteristics and attributes of the SY analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247. The audit examined the approach and methods applied in the SY analysis, ensuring they are reasonable and consistent with accepted industry practices; the staff also confirmed in audit that the assumptions are reasonable and the analysis aligns with the preliminary plant design to the extent possible.

Specifically, the audit examined the modeling of system functions, dependencies, failure modes, and HFEs for key plant systems. The audit focused on how each system was characterized to support the SC for essential safety functions, including reactivity control, core flow, primary sodium heat removal, and confinement. The audit assessed the inclusion of active and passive components, support systems, dependencies, mission times, component actuations, and operator actions, as well as the treatment of common cause failures and human errors. The audit also confirmed the documentation of assumptions, sources of uncertainty, and traceability of the analysis within each system notebook. The staff’s audit confirmed that the models, supporting data, and documentation were developed in accordance with the current design information and high level and supporting requirements of the non-LWR PRA standard. Based on its review, the staff determined the SY analysis to be reasonable because information in PSAR section 3.1.1.10 is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance that is applicable at the CP stage, including the following:

- Systems analysis models were developed with reasonable detail, consistent with the currently available design information.¹¹
- Fault tree models reflected the plant as-designed, as-to-be-constructed, and as-to-be-operated within the limits of the preliminary design information available at this time.
- The fault tree models addressed the SC necessary to mitigate event sequences.
- Support systems and their dependencies were represented within the models. However, at the CP stage, assumptions were made around the availability and capability of these systems that will need verification at the OL stage.

¹¹ Due to the preliminary nature of the design and unavailability of design information, some SSCs are not modeled within the CP stage PRA. The incorporation of additional SSCs as the design progresses may increase the frequency of some events because the additional SSCs may introduce new modes of failure.

- Both active and passive components, along with relevant failure modes affecting system functions, were included.
- Common cause failures and human errors were addressed at a preliminary level within the systems models. The identification of human errors was limited at this stage due to incomplete information on the operation and maintenance for systems.
- Mission times, component actuations, functionality, and limited associated HFEs were incorporated where applicable.
- Sources of model uncertainty related to the systems analysis have been identified and documented.
- The systems analysis documentation provides traceability of the analyses performed.

The staff will review the final SY analysis results at the OL stage, including uncertainty assessments, operational information, and assumptions underpinning the SY analysis to ensure it is performed properly and acceptable, should the applicant apply for an OL.

PRA Element 6 - Human Reliability Analysis

As described in RG 1.247, the objective of the human reliability analysis (HRA) is to identify and evaluate HFEs that could adversely affect normal or emergency plant operations, and to quantify their associated probabilities. HFEs associated with normal operations involve actions that render systems unavailable. In contrast, HFEs related to emergency operations encompass human actions that, if omitted or performed incorrectly, could prevent critical systems from performing their intended safety functions. PSAR section 3.1.1.11 describes the CP application's HRA. To further understand the description in the PSAR, the staff conducted an audit that included documents describing the HRA process implemented by USO. The audit focused on confirming the characteristics and attributes of the HRA analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247.

Specifically, the audit concentrated on the identification and evaluation of HFEs that could affect normal or emergency plant operations. The audit focused on the methodology used to characterize and quantify HFEs for pre-IE, IE, and post-IE actions, as well as the treatment of dependencies. The audit examined the use of the HRA methodology, referenced guidance, and calculation tools to ensure the analysis aligned with non-LWR PRA standards and systematically captured the potential impacts of human actions on system performance and plant safety. The audit also included the underlying assumptions and sources of modeling and completeness uncertainty. Generally, these assumptions and uncertainties were reasonable for the current design stage.

Based on its review, the staff determined the HRA analysis to be reasonable because information in PSAR section 3.1.1.11 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance that is applicable at the CP stage, including the following:

- The pre- and post-initiator HFEs affecting mitigation, were identified and analyzed.
- The HRA was performed on a POS-by-POS basis.

- No credit was taken for recovery actions.
- Human error probabilities were estimated with treatment of dependencies.
- Sources of model uncertainty related to the HRA were identified and evaluated for the limited number of human actions evaluated.
- The HRA documentation provides detail to ensure traceability of the technical basis.

The staff will review the final HRA and its results at the OL stage, including the HRA assumptions and uncertainties, to ensure it is performed correctly and meets PRA acceptability requirements, should the applicant apply for an OL.

PRA Element 7 - Data Analysis

As described in RG 1.247, the objective of the data analysis (DA) is to quantify the failure probabilities and unavailability's of the modeled SSCs by:

- Defining parameter boundaries,
- Grouping components appropriately,
- Ensuring consistency between parameter data and their definitions,
- Incorporating relevant generic industry data, design-specific information, and plant-specific evidence in parameter estimation, and
- Addressing the associated parameter uncertainties.

PSAR section 3.1.1.12 describes the CP application's DA. To further understand the description in the PSAR, the staff conducted an audit that included NAT-7878, "Natrium PRA Data Analysis (DA)," Rev. A. The audit focused on confirming the characteristics and attributes of the DA analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247.

Specifically, the audit focused on the failure probabilities and unavailabilities of the SSCs modeled in the PRA. The audit concentrated on the methods used to estimate demand-based and time-based failure rates, including the use of industry data from NUREG/CR-6928 and other references, incorporation of design-specific information, and treatment of parametric uncertainty using beta and gamma distributions. The audit also examined how component boundaries were defined, components were grouped, and dependencies, including plant-level, intersystem, intra-system, and human action dependencies, were represented, as well as the approach to common cause failure modeling.

The audit also emphasized the traceability of parameter definitions to the PRA logic model and the identification and characterization of uncertainty. The staff's audit determined that the assumptions and sources of modeling and completeness uncertainties to be reasonable for the current design stage.

Based on its review, the staff determined the DA analysis to be reasonable because information in PSAR section 3.1.1.12 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance that is applicable at the CP stage, including the following:

- Parameter estimation for basic event failure probabilities and unavailabilities incorporated generic and design-specific data. The staff notes that additional information on the bases of the failure probabilities selected is expected at the OL stage.
- Each parameter was defined with respect to the PRA logic model and probability evaluation framework.
- Component boundaries were established and respected during parameter estimation.
- Uncertainty in the parameter estimates was identified and characterized.
- The DA was sufficiently documented for the CP stage. Additional information should be provided by the applicant to conform with the PRA standard and RG 1.247, ensuring traceability to the underlying technical basis at the OL stage.

The staff will review the final DA analysis, including assumptions and uncertainties, at the OL stage to ensure the database is properly developed, comprehensive, and acceptable, should the applicant apply for an OL.

PRA Element 8 - Hazard Screening

As described in RG 1.247, the objective of the hazards screening (HS) analysis is to systematically identify all natural and human-induced hazards and, where applicable, provide a technically justified basis for the exclusion of specific hazards or hazard groups from further PRA consideration. Screening methods are typically used to demonstrate that the potential contribution of a hazard to key risk metrics, such as radiological dose or public health effects, is negligible.

PSAR section 3.1.1.5 describes the CP application's HS analysis. To further understand the description in the PSAR, the staff conducted an audit that included NAT-8294, "Natrium Demonstration - PRA Screening of External Hazards," Rev. B. The audit focused on confirming the characteristics and attributes of the HS analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021. USO identified potential hazards and evaluated each for potential inclusion in the PRA. The hazards were assessed using the qualitative or quantitative screening criteria outlined in the non-LWR PRA standard ASME/ANS RA-S-1.4-2021.

PSAR table 3.1-2 identifies internal and external hazards and indicates whether they were screened using qualitative or quantitative criteria. PSAR table 3.1-3 lists the hazards retained within the PRA scope. NAT-8294 table 6-1 contains detailed information for each hazard, including a description, screening method, and the technical basis for the screening decision. Hazards that could not be screened out were retained for further PRA development.

The staff audit of NAT-8294 focused on the identification and evaluation of the potential natural and human-induced hazards for inclusion in the PRA. The audit concentrated on assessing the

methods used for qualitative and quantitative screening to ensure consistency with the non-LWR PRA standard. The audit confirmed that hazards were screened out if bounding evaluations demonstrated a negligible impact on plant risk, while those retained were subject to further qualitative analysis, quantitative analysis, or included within the PRA scope. The staff also examined the documentation supporting screening decisions, including the technical bases for inclusion or exclusion of hazards, as summarized in tables 3.1-2 and 3.1-3 of the PSAR, as part of the audit. The staff's audit confirmed the assumptions underlying the HS analysis, as well as the identified sources of modeling and completeness uncertainties, are reasonable for the current design stage.

Based on its review, the staff determined the HS analysis to be reasonable because information in PSAR section 3.1.1.5 and tables 3.1-2 and 3.1-3 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance applicable at the CP stage, including the following:

- Natural and human-induced hazards that could affect the design, plant, or site, including site- and design-specific hazards, were systematically identified.
- Initial screening was conducted using reasonably defined qualitative criteria.
- Quantitative screening applied bounding assessments using predefined criteria.
- Quantitative screening decisions were supported by evaluation of available data.
- Uncertainties associated with the screening process were identified and characterized.
- The HS analysis was documented in detail to ensure traceability of the assessment.

The staff expects USO to reassess the HS assumptions, sources of modeling, and completeness uncertainty at the OL stage, based on final design information, and either confirm their continued validity or update the HS analysis as necessary. The staff will review the final HS analysis at the OL stage to ensure it is properly performed and acceptable, should the applicant apply for an OL.

PRA Element 9 - Event Sequence Quantification Analysis

As described in RG 1.247, the objective of the event sequence quantification (ESQ) analysis is to estimate the frequencies of event sequences and event sequence families across all phases of the plant lifecycle. This ensures that all risk-significant contributors are identified, characterized, and well understood. The ESQ analysis addresses key dependencies and provides a robust evaluation of the uncertainties and assumptions inherent in the PRA, including their impact on overall risk insights.

PSAR sections 3.1.1.13 and 3.1.1.14 describe the CP application's ESQ analysis. To further understand the description in the PSAR, the staff conducted an audit that included NAT-7364, "Natrium Event Sequence Quantification," Rev. B. The audit focused on confirming the characteristics and attributes of the ESQ analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021. The ESQ analysis integrates event sequences, system models, HRA, and data to quantify the frequencies of each modeled event sequence and event sequence family. These quantified results serve as inputs to risk

integration tasks, supporting evaluations against QHO risk metrics. PRA quantification was conducted using a single top gate approach for each event sequence and release category. Individual sequence results were aggregated for reporting and served as input for the LMP risk evaluation. Quantification results are expressed in terms of minimal cutsets representing the smallest combinations of failures or conditions that lead to radionuclide release.

The audit concentrated on the integration of event sequences, system models, HRA, and data to quantify the frequencies of ESs and event sequence families across the plant lifecycle. The audit focused on the methodology for generating minimal cutsets, the use of CAFTA, PRAQuant, UNCERT, FRANX, and FTREX for modeling and quantification, and the verification of cutsets to ensure proper integration of event trees, fault trees, system models, and human actions. The audit also assessed the application of truncation limits, the aggregation of sequence results for release categories, and the derivation of equipment and operator importance measures.

The staff's audit confirmed that the KU1 ESQ analysis reasonably incorporates dependencies, functional interactions, and uncertainties, while identifying and characterizing risk-significant contributors. The staff determined the assumptions underlying the ESQ analysis are reasonable for the current design stage.

Based on its review, the staff determined the ESQ analysis to be reasonable because information in PSAR sections 3.1.1.13 and 3.1.1.14 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance applicable at the CP stage, including the following:

- The KU1 ESQ analysis integrated all relevant modeling elements.
- Event sequences were quantified using suitable models and software, with sufficiently low truncation limits to ensure convergence of risk results.
- Risk-significant contributors to each sequence and sequence family were identified and characterized.
- Uncertainties in the quantification process were reasonably addressed and quantified.
- The ESQ analysis was documented in sufficient detail to ensure traceability of the assessment performed.

Due to the preliminary stage of the KU1 design in the CP application, the staff expects USO to reassess the ESQ assumptions at the OL stage, based on finalized design information, and either confirm their continued validity or make appropriate updates to the ESQ analysis. Additionally, since ESQ analysis involves several aspects related to event frequency, its acceptability is coupled to the acceptability of the other elements (e.g., IE, HR, SYS, ES). As those elements are to be fully addressed at the OL stage, the applicant is expected to address the limitations in the ESQ as well. The staff will review the final ESQ analysis and its results at the OL stage to ensure that it has been properly performed and is acceptable, should the applicant apply for an OL.

PRA Element 10 - Mechanistic Source Term Analysis

As described in RG 1.247, the objective of the mechanistic source term (MST) analysis PRA element (MS) is to characterize potential radiological releases to the environment resulting from event sequences that could lead to radionuclide release. This characterization includes identifying risk-significant isotopes for consequence assessment and providing key input parameters such as release locations, the physical and chemical forms of radionuclides, time-dependent isotopic release rates to the atmosphere, the heat content of the carrier fluid, and parameters necessary to estimate plume buoyancy. The MS analysis is expected to produce MSTs suitable for use in radiological consequence evaluations.

PSAR section 3.1.1.13 describes the CP application's PRA model integration and MST analysis. The MST analysis evaluates radionuclide release pathways from in-vessel reactor core events and from fuel assemblies during transport and storage in the spent fuel pool. It also addresses radionuclide transport within the plant and potential environmental release pathways. Additional evaluations that consider potential releases from supporting systems, including the SPS, SCG system, and gaseous radwaste system are discussed in section 3.2 of this SE.

Multiple computational tools were employed to develop the source term estimates, including **[]**, and RADTRAD. Bounding source terms were calculated to account for uncertainties inherent in the source term estimation process. Section 3.2 of this SE provides a detailed review of the MST analysis and summarizes the staff's determinations.

To further understand the description in PSAR section 3.1.1.13, the staff conducted an audit that included NAT-6161, "PRA Source Term and Radiological Consequences Analyses," Rev. A, including the assumptions supporting the KU1 MS analysis and their associated sources of modeling and completeness uncertainty. The audit assessed the potential radiological releases from event sequences, including in-vessel reactor core events and spent fuel handling in the EVHM and EVST. The audit verified the radionuclide transport within the plant and potential environmental release pathways, including contributions from supporting systems such as the SPS, SCG system, and gaseous radwaste system.

The staff's audit confirmed that the MS analysis reasonably applies appropriate methodologies and computational tools, and identifies key uncertainties associated with release mechanisms and transport phenomena. The assumptions supporting the analysis and associated uncertainties are also reasonable for the current design stage.

Based on its review, the staff determined the MS analysis to be reasonable because information in PSAR section 3.1.1.13 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance applicable at the CP stage, including the following:

- Radionuclide releases were reasonably grouped into representative source terms or release categories.
- Each release category was assessed with respect to timing, location, magnitude, transport barriers, and mechanisms of release and transport.
- Radiological source terms were developed using methodologies and computational tools to the extent practicable.

- Uncertainties in the MSTs and associated transport phenomena were identified and characterized.
- The MS analysis was documented to ensure traceability of the technical basis.

The staff will review the final MS analysis, including updated assumptions, uncertainties, and results at the OL stage to ensure that it has been performed in accordance with applicable guidance and is acceptable to support the LMP application, should the applicant apply for an OL.

PRA Element 11 - Radiological Consequence Analysis

As described in RG 1.247, the objective of the radiological consequence analysis is to evaluate the potential impacts of radioactive material releases from a nuclear facility under various operational and accident conditions.

To further understand the radiological consequence (RC) analysis described in the PSAR sections 3.1.1.13, 3.2, and 3.3, the staff conducted an audit that included NAT-6161. The audit focused on confirming the characteristics and attributes of the RC analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RAS1.42021 and RG 1.247.

The KU1 RC analysis used source terms developed through the MS analysis as input. The RC analysis quantified weighted individual risk by integrating health effects and dose exceedance probabilities with the annual frequencies of each source and release category. The MACCS code was used to perform the RC calculations.

The radiological consequences evaluated for each source term release category include:

- Total effective dose equivalent (TEDE) to a receptor located at the EAB
- Frequency of exceeding 100 mrem TEDE at the site boundary
- Average individual early fatality risk within one mile of the EAB
- Average individual latent cancer fatality risk within ten miles of the EAB.

The MACCS code was used to compute uncertainties associated with TEDE, early fatality risk, and latent cancer fatality risk. Both mean values and 95th percentile estimates were developed, based on analyses of nominal and bounding source terms to ensure a robust evaluation. The staff's review and conclusions regarding the KU1 RC analysis are presented in section 3.3.1.5 of this SE.

The audit focused on the evaluation of potential impacts of radioactive material releases under various operational and accident conditions. The audit assessed the RC analysis, which incorporates source terms from the MS analysis, site-specific characteristics, meteorological data, atmospheric transport and dispersion, protective actions, dosimetry calculations, and health effects estimation. The audit verified the calculations performed to quantify TEDE, early fatality risk, latent cancer fatality risk, and the frequency of exceeding 100 mrem TEDE at the site boundary. The audit also confirmed the mean and 95th percentile estimates that were

developed to account for uncertainties associated with the source terms. The audit determined that the assumptions relevant to the RC analysis are reasonable at this stage.

Based on its review, the staff determined the RC analysis to be reasonable because information in PSAR and NAT-6161 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance that is applicable at the CP stage, including the following:

- The RC analysis was conducted using justified methodologies and computational tools.
- Uncertainties associated with TEDE and associated risk metrics were identified, characterized, and quantified to the extent practicable.
- The RC analysis was sufficiently documented to ensure traceability of the technical basis.

The staff expects USO to reassess RC analysis assumptions at the OL stage using final design information and either confirm their continued validity or revise the RC analysis accordingly. The staff will review the final RC analysis results at the OL stage to ensure that it was performed appropriately and is acceptable to support the LMP application, should the applicant apply for an OL.

PRA Element 12 - Risk Integration

As described in RG 1.247, the objectives of the risk integration analysis are to: 1) establish criteria for determining the risk significance of event sequences; 2) quantify overall plant risk using appropriate metrics; and 3) characterize and quantify uncertainties associated with those metrics.

PSAR sections 3.1.1.13 and 3.1.1.14 describe the CP application's RI analysis. To further understand the description in the PSAR, the staff audited NAT-7827, "PRA Analysis - Risk Integration (Model)," Rev. B. The audit focused on confirming the characteristics and attributes of the RI analysis to ensure the analysis aligns with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and RG 1.247.

Specifically, the audit concentrated on the integration of the ESQ, spent fuel pool model, and radiological consequence analysis into a unified model with a common top event. Other internal and external hazards (e.g., internal and external fires, internal and external flooding, seismic events, high winds, tornadoes) were not included in this integration, which is consistent with Appendix A of RG 1.253. The audit confirmed that the integrated model was evaluated against the plant-level cumulative risk metrics summarized in chapter 4 of the PSAR and verified the application of truncation thresholds to ensure adequate resolution of low-frequency contributors. The staff's evaluation regarding the KU1 integrated risk results is documented in chapter 4 of this SE.

Based on its review, the staff determined the RI analysis to be reasonable because information in PSAR sections 3.1.1.13 and 3.1.1.14 and table 3.1-1 is consistent with RG 1.253 guidance for CP applications, including that external hazards such as internal fires, internal flooding, seismic events, or high winds were not included, and the analysis was performed in accordance with RG 1.247 guidance applicable at the CP stage, including the following:

- The assumptions and the sources of uncertainty were identified for each PRA element, and their potential impacts on event sequence frequencies and consequence estimates were evaluated at a preliminary level.
- Quantitative sensitivity analyses were performed on a limited set of uncertainties to assess the impact of individual uncertainties and their combinations on key risk metrics.
- The uncertainty was characterized or quantified. The staff recognizes that uncertainties related to portions of the design not modeled at the CP stage will be assessed at the OL stage, when design information becomes available, in accordance with RG 1.253.
- The RI analysis was sufficiently documented to ensure traceability of the technical basis and supporting assumptions.

At the OL stage, the staff will review the final RI analysis and its results to confirm that it has been conducted using appropriate methods and assumptions, and that it provides an acceptable basis for the LMP application, should the applicant apply for an OL.

Spent Fuel PRA

PSAR sections 3.1.1.3 and 3.1.1.6 describe the spent fuel PRA. To further understand the description in the PSAR, the staff conducted an audit that included NAT-7547, "Natrium Demonstration - Scoping PRA Analysis for Spent Fuels from EVST to Spent Fuel Pool (SFP) and in the SFP," Rev. B, which documents the development and results of the spent fuel (SF) PRA.

The staff's audit focused on reviewing the assumptions, methodologies, and scope of the KU1 SF PRA, with emphasis on the identification and characterization of spent fuel POSs, the appropriateness of IEs considered, and the completeness of ES and systems analyses. The staff assessed whether the SF PRA reasonably addressed key support systems, radionuclide release barriers, components, and operator actions necessary to mitigate potential damage during fuel transfer and storage in the spent fuel pool.

The audit concentrated on the treatment of IEs, the linkage of POS events from the internal events PRA, and the adequacy of event tree and fault tree development. The audit verified whether IEs affecting spent fuel handling were appropriately modeled, including those associated with the bottom loading transfer cask (BLTC), pool immersion cell (PIC), cask loading pit, pool handling machine, and spent fuel pool cooling system.

The audit also confirmed quantification methods, data, including the treatment of failure rates, demand probabilities, truncation limits, uncertainty characterization, and identification of significant risk contributors. The audit assessed how uncertainties were addressed in the modeling and assumptions, and whether the documentation provided sufficient traceability of the technical bases.

Overall, the staff's audit confirmed the reasonableness and completeness of the SF PRA at the current design stage.

Based on its review, the staff determined the SF PRA analysis to be reasonable because information in PSAR sections 3.1.1.3 and 3.1.1.6 pertinent to the analysis is consistent with RG 1.253 guidance for CP applications and the analysis was performed in accordance with RG 1.247 guidance applicable at the CP stage, including the following:

- The set of spent fuel POSs were identified and characterized.
- The POS events from the internal events PRA that were also used in the spent fuel PRA are reasonable.
- The IE analysis for spent fuel activities included a detailed identification and characterization of IEs and appears reasonable.
- The IE frequencies for spent fuel scenarios appear reasonably estimated.
- The ES analysis identified and characterized the relevant release barriers, safety functions, and release categories, based on the preliminary state of the design.
- The SC analysis considered key safety functions, support systems and structures, radionuclide release barriers, components, and human actions.
- The SY reflects the as-designed and as-to-be-built system configurations and appears reasonable.
- The reliability DA, including failure rates, demand probabilities, and supporting justifications, appears reasonable.
- The software tools used for quantifying the KU1 SF PRA are appropriate.
- The truncation limits set in the SF PRA quantification are appropriate.
- The identification of significant risk contributors was performed.
- The uncertainties associated with the quantification results were characterized and quantified.
- The SF PRA was sufficiently documented to ensure traceability of the technical basis and supporting assumptions.

By conforming to the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as stated in in PSAR section 3.1.1.1 and table 1.4-5, USO is expected to fully develop the SF PRA at the OL stage, including updating assumptions, system modeling, and uncertainty characterizations, as necessary based on final design information. The SF PRA will be integrated with the other PRA models to produce a comprehensive plant-level PRA and to quantify overall plant risk. The staff will review the final SF analysis at the OL stage to confirm its acceptability, should the applicant apply for an OL.

Risk Integration Sensitivities

To further confirm that the KU1 CP PRA was performed in accordance with RG 1.247 guidance that is applicable at the CP stage, the staff conducted an audit that included NAT-8076, “Natrium PRA - Risk Integration Sensitivities,” Rev. A. This document presents a sensitivity analysis that examines the impact of key assumptions and model elements used in the risk integration portion of the KU1 PRA.

Through the audit of this and other related documents (as identified in the staff audit report), the staff gained additional insights into the KU1 PRA and how variations in key assumptions and estimated frequencies affect the integrated risk results. The audit focused on the assumptions identified across the PRA elements and their influence on the overall evaluation of plant risk. This audit provided the staff with a deeper understanding of how sensitivity analyses were conducted, documented, and used within the PRA framework to evaluate the robustness of the integrated model.

The staff also examined the overall approach to sensitivity analysis, including the identification of key risk contributors, the influence of individual assumptions on risk metrics, and the treatment of uncertainties associated with the risk integration process. While the preliminary nature of the KU1 design limits the ability to draw definitive conclusions regarding the sensitivity results, the staff’s examination of the sensitivity analysis found that it was useful in assessing the relative importance of key assumptions and data. The staff confirmed that the sensitivity analysis supports ongoing development and refinement of the integrated risk model as the design progresses toward the OL stage.

PRA Self-assessment

The KU1 PRA self-assessment is described in PSAR section 3.1.1.2, which states that the self-assessment covers the twelve technical elements of the PRA from ASME/ANS RA-S-1.4-2021, which are also listed above in this SE. The twelve elements are also the subject of corresponding regulatory positions in RG 1.247.

To further understand the description in the PSAR, the staff conducted an audit that included NAT-8218, “Natrium PRA Self-Assessment Against ASME/ANS RA-S-1.4-2021,” Rev. B, and included the self-assessment in the scope of the staff’s three-day, in-person audit of the KU1 PRA model and its supporting documentation. The audit focused on confirming that the self-assessment was performed in accordance with RG 1.247 and RG 1.253, as provided in section 3.1.1.2 of the PSAR. The staff made several observations regarding the self-assessment including the following:

- The scope of the self-assessment was consistent with the guidance in RG 1.253, including the twelve elements listed above in this SE. Consistent with RG 1.253, the following PRA elements were not analyzed for the CP application and were therefore excluded from the KU1 PRA self-assessment: internal flood, internal fire, seismic, high wind, external flooding and other external hazards PRAs.
- The assessment results were documented and made available for staff audit.
- The self-assessment appeared to be conducted primarily as individual comparisons of PRA standard supporting requirements against documentation. Enhanced team-based

holistic evaluation of the adequacy of the PRA models and the underlying technical assessments may result in enhanced coordination on resolution plans to address identified self-assessment findings.

- Portions of the self-assessment appeared to have been performed by individuals with limited PRA expertise, which could lead to varying interpretations of the PRA supporting requirements and potentially influence the overall conclusions and grading outcomes.
- While recognizing that the KU1 PRA self-assessment serves as a valuable initial tool for evaluating PRA quality, the staff does not rely solely on it to determine PRA adequacy or acceptability, even for PRA supporting requirements assessed as “Met.”

The staff audited the grading of PRA supporting requirements, identification of gaps, and documentation of assumptions to ensure that the self-assessment appropriately captured areas where PRA supporting requirements were categorized as “Met,” “Met with Gap,” “Not Met,” or “Not Applicable.” The staff’s audit emphasized understanding the preliminary nature of the design information and PRA, and the extent to which assumptions, uncertainties, and gaps were identified and characterized to support future resolution.

The staff also audited the gap resolution, potential impacts on risk-informed applications, and verification of configuration control practices. While acknowledging limitations in the self-assessment, such as its individual-based approach and the variable level of PRA expertise involved, the self-assessment appeared to be systematic, sufficiently documented, and provided useful insights to support future refinement and verification of the PRA.

Based on its review of the PSAR and confirmation via the audit described above, the staff finds that the KU1 PRA self-assessment was conducted per RG 1.247 and RG 1.253. In PSAR section 3.1.1.2, USO stated that a PRA peer review will be conducted to support the OL stage PRA, which will replace the KU1 PRA self-assessment and provide an additional level of confidence in the PRA model and results. The staff will review the results of the KU1 PRA peer review at the OL stage and consider them to inform its conclusions regarding PRA quality and acceptability.

PRA Configuration Control Program

PSAR section 3.1.1.2 describes the KU1 PRA configuration control process. The non-LWR PRA standard ASME/ANS RA-S-1.4-2021 describes that PRA results used to support licensing or risk-informed applications, including the LMP, must be derived from a PRA model that represents the as-designed, as-to-be-built, and as-to-be-operated or the as-built and as-operated plant. The model’s level of detail should be commensurate with the intended application and its associated decision-making requirements. Consequently, as discussed in RG 1.247 and RG 1.253, the design-specific PRA should be maintained and updated as necessary to ensure ongoing consistency with the plant’s design and operational status throughout the various licensing stages, in alignment with the availability of plant information.

Accordingly, USO has established its PRA configuration control procedure, documented in NAT-13718, “Natrium PRA Configuration Control Procedure,” Rev. 0, which defines the process for maintaining and managing the KU1 PRA model, its documentation, model change tracking, and supporting information.

In response to staff audit questions regarding the KU1 PRA configuration control program, USO updated the PSAR section 3.1.1.2 to describe that the PRA documentation and software are maintained under the PRA configuration control process that complies with the Quality Assurance Program described in PSAR section 8.1.

The staff's audit of the KU1 PRA configuration control program confirmed its ability to effectively maintain and manage the PRA model and its supporting documentation throughout the plant's design and operational lifecycle. The audit focused on the processes for:

- monitoring changes in plant design, operations, PRA technology, and industry experience;
- maintaining alignment of the PRA model with the as-designed, as-to-be-built, and as-to-be-operated plant; and
- systematically evaluating the cumulative impact of pending changes on risk-informed applications.

The staff also examined how configuration control is applied to computer codes and associated files used for PRA quantification, and how updates and modifications are documented to ensure traceability. The staff's audit included review of USO's design change control and corrective action programs used to track design modifications, discrepancies between the plant and the PRA model, and relevant operating experience. The staff's audit confirmed that the KU1 PRA configuration control procedure is reasonably structured, includes defined criteria and triggers for model updates, and maintains effective controls over PRA model changes.

Based on the staff's confirmation via its audit of NAT-13718, the staff determined for the CP application, that:

- The KU1 PRA configuration control procedure is reasonably structured to monitor PRA inputs and support the integration of new information that may influence the PRA.
- The cumulative impact of pending plant changes is to be systematically evaluated.
- Configuration control over computer codes and associated files used for PRA quantification is maintained.
- The process defines criteria and triggers for updating the PRA model in response to new information.

Overall, the staff determined the KU1 PRA configuration control procedure to be reasonably developed. It establishes effective processes and controls to maintain and enhance the PRA as the plant design evolves, thereby ensuring sustained support for the LMP and other risk-informed initiatives.

Assessment of Conformance with the Staff's Positions in RG 1.253 Regarding PRA and Its Results

The staff also evaluated the KU1 PRA and its supporting PRA-related documentation available through the PRA audit, as mentioned above, against the applicable regulatory positions outlined

in RG 1.253, which describes an approach that is acceptable to the staff for using a technology-inclusive content-of-application methodology to inform the PSAR. The staff's determinations are summarized as follows:

1. RG 1.253, section A.2.2, states that the CP applicant should clearly document in the PSAR the key assumptions made during the development of the LMP-based safety analysis and in selecting PRA model elements.

Staff Determination: The staff determined the regulatory position in RG 1.253, section A.2.2 is satisfied because PSAR table 3.1-1 identifies the key assumptions used in the PRA development.

2. RG 1.253, section A.2.3, states that the CP applicant considers both near-term and long-term uses of the PRA during its development to ensure continued applicability. Besides supporting LMP implementation, PRA results may be utilized to demonstrate regulatory compliance and support other voluntary risk-informed applications.

Staff Determination: Although no specific regulation mandates a PRA to support a CP application under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," USO developed the KU1 PRA in part to demonstrate compliance with applicable regulatory requirements, including:

- 10 CFR 50.34(a)(1)(ii): Expectation that reactor design, construction, and operation will reflect an extremely low probability of accidents resulting in significant radioactive releases.
- 10 CFR 50.34(a)(4): Requirement that the PSAR includes a preliminary analysis and evaluation of the design and performance of SSCs of the facility with the objective of assessing risk to public health and safety and the adequacy of accident prevention and mitigation features.

Beyond supporting the LMP methodology, the KU1 PRA informs other licensing activities that are described in the KU1 CP application, such as the design reliability assurance program, reliability and integrity management program, and environmental reviews.

Since the KU1 CP PRA only addresses internal events and relies on key assumptions not yet validated at the CP stage due to the preliminary nature of the design, its acceptability to support full implementation of all cited intended uses is not yet determined. However, based on the evaluations in the previous subsections of SE section 3.1.3, the staff determined the methodologies and processes used in developing the internal events and LPSD PRA portions are generally reasonable and consistent with RG 1.247 guidance for the CP stage.

3. RG 1.253, section A.3.1, states that a CP applicant using the LMP methodology should estimate the risk metrics described in sections C.3.2.1, C.4.1.1, C.4.1.2, and C.4.1.3 of NEI 21-07 on either a qualitative or quantitative basis consistent with the information available when the CP application is prepared. Such an estimation should include an explanation of how the Commission's QHOs from the Commission Safety Goal Policy Statement will be met in support of the OL application.

Staff Determination: PSAR section 3.5 provides USO's summary evaluation of LBEs against the F-C target, consistent with NEI 21-07 section C.3.2.1, with further detail on each LBE provided in PSAR sections 3.6 through 3.8. The staff's evaluation of the LBE summary and detailed evaluation of LBEs is provided in SE sections 3.4 through 3.7. PSAR section 4.1 provides the three cumulative plant risk performance metrics, including: 1) site boundary risk, 2) EAB early fatality risk, and 3) latent cancer risk. The staff's evaluation of these risk metrics is provided in chapter 4.1 of this SE. Based on the staff evaluation of the calculated KU1 LBE evaluation against the F-C target and cumulative risk metrics, the staff determined that regulatory position A.3.1 in RG 1.253, is fulfilled.

4. RG 1.253, section A.3.2, states that the CP applicant should:
 - a) Identify all radiological sources, hazards, and POSs by performing a comprehensive and systematic search.
 - b) Disposition the search results by a combination of PRA logic modeling, acceptable screening methods, risk-informed supplemental evaluations, and crediting design-basis hazard levels.

Staff Determination: PSAR section 3.1.1.3 identifies the radiological sources evaluated in the PRA. PSAR section 3.1.1.5 summarizes the hazards that were addressed by detailed PRA or DBHLs. PSAR section 3.1.1.6 lists the POSs that were included in the scope of the KU1 PRA. Based on the evaluation in the previous subsections of SE section 3.1.3 pertaining to source, hazard, and POS identification, the staff determined that the applicant satisfied the regulatory position in item (a) above.

In addition to the KU1 PRA development, USO implemented a screening analysis as documented in PSAR table 3.1-2 and considered site-specific DBHLs, as presented in PSAR table 6.1-1, thereby the applicant has addressed the regulatory position in item (b) above.

5. RG 1.253, section A.3.3, states that the minimum scope of the CP stage PRA logic model should include the internal events hazard group for the reactor in the at-power POS.

Staff Determination: The KU1 CP stage PRA addresses internal events by including all eleven internal event elements identified in RG 1.253 table A-1, as well as an additional element covering POSs analysis. The PRA scope aligns with RG 1.253 guidance, with any deviations appropriately documented and justified as discussed above in this SE. Therefore, the staff determined that the applicant fully satisfies RG 1.253, section A.3.3.

6. RG 1.253, section A.3.4 states that the PRA high level requirements¹² (HLRs) and supporting requirements from the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as endorsed in RG 1.247, provide an acceptable approach for PRA logic model development.

¹² RG 1.253, section A.3.4, also includes a footnote that states, "The non-LWR PRA standard uses the terms "requirement," "require," and other similar mandatory language. However, the use of this language in this RG does not imply that this RG imposes any regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements."

Staff Determination: USO developed the CP stage PRA in accordance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and conducted a PRA self-assessment against the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 HLRs and supporting requirements, as discussed in PSAR section 3.1. For those PRA supporting requirements identified as “Not Met” or “Not Fully Met”, USO provided justifications consistent with the standard guidance. The staff determined USO has satisfactorily met the regulatory position of RG 1.253, section A.3.4.

7. RG 1.253, section A.3.5, states that section 4.3.11 of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 provides an acceptable approach for performing hazards screening analyses.

Staff Determination: The HS analysis was performed and documented in PSAR table 3.1-2. The staff verified that USO followed the guidance in the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 when conducting the HS. While some initial screening justifications were found to be unacceptable by the staff, USO will revise the justifications and update NAT-8218 accordingly to support the OL application. The staff determined that USO’s HS approach is consistent with the regulatory position in RG 1.253, section A.3.5, for a CP stage PRA.

8. RG 1.253, section A.3.6, states that non-LWR CP applicants utilizing NUREG-1855 guidance to develop risk-informed supplemental evaluations should (1) describe and justify the use of reactor-technology-specific screening criteria, and (2) explain how specific sources of uncertainty were identified and addressed.

Staff Determination: This regulatory position is not applicable because USO does not follow NUREG-1855 guidance. However, the staff notes that USO quantifies uncertainty using the CAFTA uncertainty analysis module. Since the use of NUREG-1855 is optional, the applicant is not subject to the position.

9. RG 1.253, section A.3.7, states that CP applicants may disposition certain hazards by crediting design basis hazard levels instead of explicitly modeling these hazards in the PRA or addressing them through risk-informed supplemental evaluations.

Staff Determination: In response to staff audit questions, USO updated PSAR section 6.1.1 to state that the DBHLs are selected in accordance with existing NRC approved methods and are consistent with the DBHLs described in section 6.1.1 and table 6-1 of NEI 21-07 and with the guidance on the scope of hazards found in chapter 3 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.” The staff determined that USO’s approach aligns with the regulatory position in RG 1.253, section A.3.7.

10. RG 1.253, section A.4.1, identifies the minimally acceptable PRA elements and additional elements recommended for full LMP methodology implementation at the CP stage.

Staff Determination: USO addressed all minimally acceptable PRA elements and included the POS element from the additional list, as described in PSAR section 3.1.1.2. Accordingly, the staff determined that USO has met the regulatory position of RG 1.253, section A.4.1.

11. RG 1.253, section A.4.2, states that PRA elements should be developed consistently with the PRA HLRs and supporting requirements of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, with all HLRs expected to be met.

Staff Determination: The staff determined that the regulatory position in RG 1.253, section A.4.2 is satisfied because USO addressed all HLRs applicable at the CP stage through its self-assessment and provided justifications for any gaps, primarily due to undeveloped areas such as operator and maintenance procedures, and explained their impacts on PRA applications in their PRA self-assessment report.

12. RG 1.253, section A.5.1, states that the CP stage PRA's level of detail should be commensurate with the preliminary plant design and site characteristics described in the PSAR.

Staff Determination: The PSAR includes limited plant design details. While the PRA incorporates many assumptions about some aspects of plant operation, they are generally reasonable given the design maturity at the time of the CP application. While some assumptions regarding SSCs appear non-conservative, overall, the staff determined that KU1's PRA development conforms to RG 1.253, section A.5.1, utilizing available design information as previously described in section 3.1.3 of this SE, under each PRA element.

13. RG 1.253, section A.5.2, states, in part, that when certain PRA elements cannot be met due to the maturity of preliminary plant design and information about site characteristics, the application should justify the adequacy of the internal events PRA logic model for the reactor in the at-power POSs to support the implementation of the LMP methodology.

Staff Determination: USO acknowledged gaps and limitations in the KU1 PRA and will develop a full-scope PRA at the OL stage in accordance with the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, as discussed in PSAR section 3.1.1.2. The KU1 CP stage PRA self-assessment provides justifications regarding the impact of current gaps on LMP implementation. Although some justifications cannot yet be confirmed due to incomplete design information, which is expected due to the preliminary design available at the CP stage, the staff determined USO has met the regulatory position in RG 1.253, section A.5.2, because it provided justification on PRA adequacy.

14. RG 1.253, section A.6.1, states that CP applicants should establish a PRA configuration control program to ensure the CP stage PRA reasonably reflects the preliminary plant design and site characteristics described in the PSAR.

Staff Determination: USO has developed a PRA configuration control program as described in PSAR section 3.1.1.2. The program provides a comprehensive framework for PRA updates and upgrades. Although some areas, such as documentation of data sources, expert elicitation, and routine error identification and correction, are not fully addressed yet, the staff determined the overall configuration control procedure is reasonable. Therefore, the applicant has met the regulatory position in RG 1.253, section A.6.1.

15. RG 1.253, section A.6.2, states that section 5 of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 provides an acceptable approach for establishing a PRA configuration control program.

Staff Determination: The staff's audit confirmed that the KU1 PRA configuration control program was developed in accordance with section 5 of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021. Although the KU1 PRA self-assessment did not explicitly evaluate the established configuration control program against the PRA standard configuration control supporting requirements, the staff considers this acceptable because a comprehensive PRA peer review is planned for the OL stage to address this gap. Because the PRA configuration control program was developed in accordance with ASME/ANS RA-S-1.4-2021, the staff determined that USO has met the regulatory position outlined in RG 1.253, section A.6.2.

16. RG 1.253, section A.6.3, states that the PRA configuration control program may be implemented either as a stand-alone program or included within the quality assurance program required by 10 CFR 50.34(a)(7).

Staff Determination: As stated in PSAR section 3.1.1.2, the KU1 PRA documentation and software are maintained under a PRA configuration control process that complies with the Quality Assurance Program described in PSAR section 8.1. Therefore, the staff determined that USO has satisfied the regulatory position in RG 1.253, section A.6.3.

17. RG 1.253, section A.7.1, states that NEI 21-07 provides an acceptable approach and format for presenting CP stage PRA information in the safety analysis report. The PSAR should include a summary justification of the acceptability of the CP stage PRA. This justification should summarize why, commensurate with the facility design maturity, the scope, level of detail, PRA elements, and plant representation are sufficient to implement the LMP methodology for the CP application. Specifically, the summary justification should describe, but is not limited to, the following:

- How the CP stage PRA was developed in accordance with this appendix, RG 1.247, and non-LWR PRA standard ASME/ANS RA-S-1.4-2021;
- How the PRA configuration control program ensures the CP stage PRA represents the as-designed, as-to-be-built, and as-to-be-operated facility described in the CP application;
- How the PRA configuration control program will ensure that the PRA for the OL application will represent the as-built, as-to-be-operated facility, including all radiological sources, all hazards, and all POSs, and meet all applicable staff positions in RG 1.247 and technical elements of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 not addressed in the CP stage PRA;
- How the applicant's self-assessment of the CP stage PRA was performed consistent with staff positions in section A.8 of RG 1.253.

Staff Determination: As described in PSAR section 3.1.1.2, the KU1 PRA was developed following the non-LWR PRA standard ASME/ANS RA-S-1.4-2021, which the staff confirmed in its audit was documented in NAT-7444, "Natrium PRA Methodology,"

Rev. A. The technical adequacy of the PRA was justified by a PRA self-assessment against the non-LWR PRA standard, as discussed in PSAR section 3.1.1.2.

18. RG 1.253, section A.7.2, states that staff position C.4.1 in RG 1.247 provides an acceptable approach for developing and preserving PRA archival information. PRA documentation providing the detailed justification supporting PRA acceptability should be maintained in archival documentation and, at minimum, include the items described in staff position C.4.1 of RG 1.247.

Staff Determination: As confirmed during the staff audit, while PRA documentation is not fully complete at the CP stage, the justification for PRA acceptability and the items listed in staff position C.4.1 of RG 1.247 are maintained in archival documentation. Therefore, the staff determined that USO has satisfied the regulatory position in section A.7.2 of RG 1.253.

19. RG 1.253, section A.7.3, states that PRA archival information may be controlled through a stand-alone program or the quality assurance program required by 10 CFR 50.34(a)(7).

Staff Determination: As stated in PSAR section 3.1.1.2, the KU1 PRA documentation and software are maintained under a PRA configuration control process that complies with the Quality Assurance Program. Accordingly, the staff determined that USO has satisfied the regulatory position in section A.7.3 of RG 1.253.

20. RG 1.253, section A.8.1, states that the guidance in DANU-ISG-2022-05, section 11.1.1, provides an acceptable approach for describing key management responsibilities for PRA development.

Staff Determination: The staff confirmed through audit that the key management responsibilities, as described in the KU1 PRA configuration control program (PSAR section 3.1.1.2 describes the KU1 PRA configuration control process), are reasonable. Therefore, the staff determined that USO addressed the regulatory position in section A.8.1 of RG 1.253.

21. RG 1.253, section A.8.2, states that the guidance in DANU-ISG-2022-05, section 11.1.1.1, provides an acceptable approach for describing ability of the CP applicant's technical staff to develop the PRA.

Staff Determination: PSAR section 3.1.1.2 describes the KU1 PRA configuration control process. The staff confirmed through audit of NAT-13718 that the responsibilities of the PRA technical staff were documented and that their experience in risk analysis was sufficient to perform PRA. Therefore, the staff determined that the regulatory position A.8.2 was addressed.

22. RG 1.253, section A.8.3, states that the CP applicant should conduct a self-assessment to demonstrate that all PRA logic models, screening analyses, and risk-informed supplemental analyses have been developed and applied in a technically acceptable manner, including the appropriateness of assumptions and approximations.

- a) To support this objective, the self-assessment should provide a basis for asserting that the CP stage PRA is acceptable for implementing the LMP

methodology leading up to the CP application submission. The self-assessment should include a review of:

- i. The comprehensive and systematic search used to identify radiological sources, POSs, and hazards;
- ii. The PRA logic models (including scope, level of detail, and elements), screening analyses, risk-informed supplemental evaluations, and credit for DBHLs;
- iii. The CP applicant's PRA configuration control program used to ensure that the CP stage PRA logic models, screening analyses, and risk-informed supplemental analyses represent the as-designed, as-to-be-built, and as-to-be-operated plant.

Staff Evaluation: The KU1 PRA self-assessment, described in PSAR section 3.1.1.2, addressed these items. However, the staff observed several areas for enhancement, as discussed in the KU1 PRA self-assessment section above. Despite areas for enhancement, since a self-assessment was performed and documented that addresses the items listed above the staff determined that USO sufficiently addressed the regulatory position of RG 1.253, section A.8.3.

3.1.4 Conclusion

The staff reviewed the KU1 PRA as described in section 3.1 of the PSAR and conducted an audit that included supporting PRA documentation and PRA-related information. The staff concludes that the KU1 PRA was developed consistent with the guidance in RG 1.247, RG 1.253, and the applicable supporting requirements and HLRs of the non-LWR PRA standard ASME/ANS RA-S-1.4-2021.

Based on the staff audit, the staff determined that USO and its PRA team responsible for KU1 PRA development have demonstrated general competency in developing the PRA, including the detailed treatment of individual PRA elements. Based on its review of the PSAR as confirmed through audit, the staff concludes that the processes used to develop the KU1 PRA are logical and structured. The staff also notes that, through its self-assessment, USO identified and documented deviations from the non-LWR PRA standard, along with corresponding resolutions, which will be addressed in updates at the OL stage.

The staff's review determined that the PRA information presented in the PSAR:

1. Aligns with the guidance in RG 1.253, Appendix A, and NEI 21-07;
2. Provides a summary of the PRA development, including its scope and level of detail;
3. Describes how PRA results have informed plant design and the safety analysis;
4. Identifies process for updating and finalizing the PRA;
5. Supports the safety conclusions documented in chapter 4 of the staff's SE;
6. Derives insights to support risk-informed licensing programs; and

7. Serves as a foundation for LMP implementation.

The staff concludes that the PRA information described in the PSAR conforms to the guidance documents, referenced above, and that the scope, level of detail, and quality of the KU1 PRA are reasonable for the CP application. This conclusion is based on the understanding that the KU1 PRA is preliminary to support the CP application, and that the applicant is not requesting finality at this stage.

In accordance with RG 1.247 and RG 1.253, the staff expects USO to perform the following in support of its OL application, should the applicant apply for an OL:

1. Reassess the IEs to ensure completeness and to include human-induced initiators.
2. Update and expand the PRA scope to incorporate all applicable radiological sources, all internal and external hazards that may affect plant operations, and all POSs relevant to risk evaluations.
3. Update the level of detail in the CP stage PRA to accurately reflect plant behavior and operations expected at the OL stage.
4. Reevaluate assumptions to ensure their continued validity or appropriately address them in the PRA.
5. Update the PRA to reflect the as-designed, as-built, and as-to-be-operated plant configuration.
6. Ensure that PRA models, quantification, and documentation conform to the non-LWR PRA standard ASME/ANS RA-S-1.4-2021 and the regulatory positions in RG 1.247.
7. Reassess sensitivity studies to ensure relevance and update PRA insights accordingly.
8. Update the PRA notebooks and supporting documentation to maintain technical accuracy and consistency.
9. Implement all activities defined in the KU1 PRA configuration control procedure.
10. Conduct a PRA peer review, address all identified findings, and develop a resolution plan for any long-term issues.
11. Update all applications that rely on PRA results and insights to reflect the revised PRA.

In summary, based on its review and evaluation described above, the staff determined that the KU1 PRA is reasonably developed and maintained, and provides sufficient support for the issuance of KU1 CP.

3.2 Licensing Methodology for Mechanistic Source Term

PSAR section 3.2 describes the analytical methods developed to calculate MSTs for use in the evaluation of the dose consequences from LBEs, including DBAs and non-DBA LBEs.

The following topical report is incorporated by reference into PSAR section 3.2:

- NAT-9392-A, "Radiological Source Term Methodology Report," Rev. 0 (ML25211A271)

The regulatory requirements for the evaluation of the source term are as follows:

- 10 CFR 50.34(a)(1), (a)(4), and
- 10 CFR 50.35.

The applicable guidance for the evaluation of the source term are as follows:

- RG 1.233 which endorses NEI 18-04
- RG 1.253 which endorses NEI 21-07
- RG 1.247, which endorses the non-LWR PRA standard ASME/ANS RA-S-1.4-2021

While RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 1 (ML23082A305), was developed for LWRs and does not give guidance on the development of non-LWR MSTs, RG 1.183 section 2 provides useful information for non-LWRs regarding development of non-scenario-specific source terms for major accidents or maximum hypothetical accidents. Such source terms are used in consequence analyses performed to show compliance with the 10 CFR 50.34(a)(1)(ii)(D) safety analysis requirements, as well as the non-seismic site criteria postulated accident analysis requirement in 10 CFR 100.21(c)(2) which references the safety analysis dose criteria in 10 CFR 50.34(a)(1).

Policy considerations on non-LWR MSTs and functional containment are provided in SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," (ML040210725) and SECY-18-0096, "Functional Containment Performance Criteria for Non-Light Water Reactors," (ML18114A546) and their related staff requirements memoranda (SRM), respectively.

3.2.1 Technical Evaluation

Radiological source term refers to the type, quantity, and timing of the release of radioactive material from a facility during a postulated event. Traditionally, the assumed release of radioactive material from its initial location (e.g., reactor fuel) during an accident has been defined deterministically. For a major accident in an LWR, it was initially defined as an instantaneous release into the containment and more recently as a release overtime (see RG 1.183) into the containment. The composition and chemical form of the released material were also defined. In addition, credit was given for attenuation of the released material prior to it reaching the environment, consistent with the plant design features (e.g., suppression pool, containment sprays) and physical properties (e.g., decay) of the radioactive material.

NEI 18-04 proposes a more realistic (mechanistic) approach for determining the source terms for non-LWR event sequences through an integrated assessment of the integrity of the reactor fuel and, upon fuel failure, the subsequent release, nature, transport and attenuation of the radioactive material prior to reaching the environment. The proposed approach relies on having

sufficient knowledge (e.g., data, analysis, operating experience) regarding the release of the radioactive material from its initial location during an accident so that it can be accurately modeled and used to calculate the release as a function of time (or as a function of some other parameter, such as fuel temperature), including its composition, chemical form, and uncertainty. Source terms may also be specific to an LBE type, plant operating state or location of the radionuclides.

PSAR section 3.2 provides summaries of the source terms for DBAs and non-DBA LBEs with radiological release, presented as time-dependent activity per radionuclide released to the environment. As stated in the PSAR and further described in the referenced TR, the types of events evaluated include events related to significant reduction in flow through the reactor core, localized high power-to-flow conditions within core assemblies, physical damage to the fuel, and reactivity insertion. PSAR section 3.2 contains tables summarizing the source terms developed for the anticipated operational occurrences (AOOs) described in PSAR section 3.6, the design basis event (DBEs) described in PSAR section 3.7, the beyond design basis events (BDBEs) described in PSAR section 3.8, and the DBAs described in PSAR section 3.9.

NEI 18-04 and NEI 21-07 describe the processes and information to justify the use of MSTs in an LMP-based safety analysis. The staff reviewed PSAR section 3.2 against NEI 18-04 and 21-07 guidance, including review to ensure that the MST description in the PSAR was adequate for use in analyses to show compliance with requirements in 10 CFR 50.34(a)(1) and (a)(4).

The staff also reviewed the technical basis for the MSTs proposed for use in the safety analysis described in the PSAR, including the referenced source term methodology TR (NAT-9392-A), with consideration of the information on non-LWR PRA MST analysis in RG 1.247. The staff conducted an audit of supporting source term and consequence analysis documentation (ML25302A443) and confirmed that the development of the MSTs used in DBA and LBE analyses used methodology tools, inputs, and assumptions, consistent with the approved NAT-9392-A methodology TR.

PSAR section 3.2 states that the TerraPower radiological source term methodology TR describes the methodology used to develop MSTs. The PSAR incorporates by reference sections 1 through 8 of the TR NAT-9392-A, which describes the methodology to develop event-specific MSTs. The staff's SE (ML25211A271) approving NAT-9392, was issued on June 16, 2025. In the SE for the TR, the staff determined that NAT-9392, subject to the limitations and conditions (L&Cs) discussed in the SE, provides an acceptable approach for developing MSTs for DBAs, LBEs, and normal operation to support analyses to show compliance with the regulatory requirements in 10 CFR 50.34(a)(1) and (4). The staff's evaluation determined that the NAT-9392 methodology was consistent with the guidance in RG 1.233 and RG 1.247 with respect to MST development for LMP and PRA, respectively. There are eight L&Cs on the use of NAT-9392-A. The staff confirmed the L&Cs identified in the radiological source term methodology TR were either met or evaluated as part of the PSAR. The staff's SE for NAT-9392-A imposed the following L&Cs:

1. The methodology is limited to a Natrium design that has a pool-type, sodium fast reactor (SFR) design with metal fuel and sodium bond as described in TR sections 1.3 and 2.3.1. Changes from these design features will be identified and justified in safety analysis reports of Natrium license applications.

2. The fuel failure fractions during normal operation and transient conditions are subject to the qualification of Type 1 fuel.
3. If bonded sodium is not utilized in subsequent fuel designs, additional information shall be provided to justify the fission product release behavior from metal fuel to the gas plenum.
4. The sodium pool scrubbing and associated radionuclide (RN) retention within the primary sodium coolant is limited to where the bulk sodium is in subcooled conditions.
5. Adequate verification and validation assessment information should be made available to the staff as part of future submittals supporting the codes that make up the evaluation model (EM). This verification and validation information should be justified to reasonably bound the operational envelope for the design for any applicant referencing the source term EM methodology.
6. User inputs to analytical tools used in the source term EM (e.g., parameter values) for which specified values are not provided in this TR should be documented and justified in the analysis supporting a license application referencing this TR methodology.
7. The source term EM described in this methodology results in MSTs intended for use in LMP-based license applications or other best-estimate plus uncertainty analyses. For applications using another process (e.g., conservative deterministic licensing analysis using postulated maximum hypothetical accident), the user must demonstrate that the TR methodology is applicable to that other process.
8. The TR documents that certain activities related to the development of the source term EM have not been completed. These activities are relevant to steps 6, 7, 8, 9, 12, 13, 14, 15, 16, 17, 18, 19, and 20 of the Evaluation Model Development and Assessment Process (EMDAP). This also includes ongoing work related to experimental data to justify the source term phenomena closure models as described in TR section 3. An applicant or licensee referencing the methodology developed in this TR must submit documentation and justify that the identified activities have been completed to a state that is appropriate for the intended licensing application and that the identified information has been provided.

The staff determined that the use of NAT-9392-A in the PSAR meets TR L&Cs 1 through 3 because the preliminary Natrium design described in the PSAR is consistent with the design information in the L&Cs. Consistent with L&C 4, the staff verified through review of the LBE analyses in PSAR chapter 3 that the scenarios in which the MST methodology is used do not include sodium bulk boiling. The staff determined that the code qualification, verification and validation activities discussed in NAT-9392-A for this methodology are sufficient for preliminary analyses supporting a CP application and thus TR L&C 5 is addressed. The staff determined through audit that analyses supporting the MSTs described in the PSAR provide documentation of user inputs to analytical tools used in the methodology. Although the documentation provided justification of user inputs and assumptions for the preliminary analyses, the staff noted that the DBA and non-DBA analyses were inconsistent with each other. In audit discussions, the applicant acknowledged these inconsistencies and stated that as they develop the scenario-specific source terms to support the OL application, they anticipate resolving the differences and providing the necessary justifications. Therefore, the staff could not find that TR L&C 6 is fully met at this time but determined it is reasonable to leave further review of the user

inputs and assumptions in the staff's evaluation of the implementation of NAT-9392-A using final design information in support of an OL application. Because the PSAR summarizes how the NEI 18-04 LMP methodology is implemented for preliminary safety analyses for the Natrium design, the staff determined that the application meets TR L&C 7. The staff determined that TR L&C 8 is addressed because the activities described in the TR are not needed because the source term EM is sufficient for preliminary analysis and PSAR section 3.2 states that the items identified for development in the referenced source term methodology report will be available by the OL stage. Therefore, based on its assessment of the PSAR with respect to the accident source term methodology TR L&Cs the staff determined that the use of NAT-9392-A in analyses supporting the PSAR acceptably addresses the TR L&Cs.

The staff's review of the PSAR non-DBA LBE and DBA source terms is described below. In performing this review, the staff evaluated the applicant's implementation of the NAT-9392-A methodology to develop the source terms, with specific evaluation of the justification of user inputs that were not provided in NAT-9392-A. The discussion of specific LBE source terms is organized by event type under separate SE subheadings.

For the CP application, the applicant developed a limited number of source terms based on the preliminary stage of design and analysis. As shown in PSAR sections 3.6 through 3.9, some source terms were applied to multiple LBEs. The staff confirmed through the regulatory audit that some LBE consequence analyses applied a representative source term based on another related scenario, sometimes with source term scaling factors to account for different LBE conditions. Based on its review, the staff determined that the use of representative source terms is reasonable for the preliminary analyses. The staff anticipates that the applicant will develop LBE-specific source terms as the design and analyses mature and that the final safety analysis report (FSAR) in support of the OL application will include these LBE-specific source terms in the consequence analyses instead of carrying forward the PSAR analyses.

PSAR tables 3.2-1 through 3.2-18 provide the 18 source terms used in the radiological consequence analyses of LBEs with release and DBAs. As stated in PSAR section 3.2.1, these tables list the isotopes which contribute to at least 95 percent of the total dose and the cumulative activity release of each isotope in curies. The staff confirmed through audit of the supporting analyses that the MSTs and consequence analyses include a larger, more complete list of isotopes determined to be important with respect to the dose to an individual for releases from the Natrium reactor. This is consistent with the methodology provided in the radiological release consequences methodology, NAT-9391-A, "Radiological Release Consequences Methodology Topical Report," Rev. 0 (ML25211A267). SE section 3.3.1.5 describes the staff's review of the use of NAT-9391-A. In addition, the staff also confirmed through the audit that the list of isotopes included in the source term calculations are consistent with the radionuclide screening methods described in the NAT-9392-A source term methodology.

Through the audit the staff observed that in implementing the NAT-9392-A MST methodology, the calculations supporting the PSAR used information from two Argonne National Laboratory (ANL) reports developed for SFRs: ANL-ART-38, "Regulatory Technology Development Plan, Sodium Fast Reactor, Mechanistic Source Term – Metal Fuel Radionuclide Release," (2016) and ANL-ART-49, volume 1, "Regulatory Technology Development Plan, Sodium Fast Reactor, Mechanistic Source Term – Trial Calculation," (2016). These reports are also referenced in the approved NAT-9392-A methodology as sources of information for the user of the methodology to provide a basis for the required user input to the source term evaluation models. As stated in the SE for the MST methodology in NAT-9392-A, the staff determined that the referenced national laboratory reports are relevant to SFR designs such as the Natrium reactor.

The staff observed through audit that the MST analyses used radionuclide inventories for the sources of radiological material at risk of release (MAR) which were estimated using the methods described in the NAT-9392-A methodology. Fuel isotopic inventories were developed using appropriate isotope generation and depletion computer codes and preliminary design information for the Sodium Type 1 and Type 1B fuel and core conditions. [[

]]. The applicant noted, and the staff confirmed, that the overall source term modeling is biased conservatively for the spent fuel assemblies. The staff anticipates that consideration of the additional burnup during storage will be included in the updated fuel isotopic inventories to support the event-specific source terms for the OL application. The staff's evaluation of fuel storage in the IVS is discussed in SE section 3.13.

The inventories of radionuclides in the primary sodium and cover gas during normal operations are used as a basis for the MAR for other ex-vessel releases. [[

]]. The staff found the audited information showed that the MAR isotopic inventories were conservatively estimated using the NAT-9392-A methodology and supported the PSAR section 3.2 description of the source term development for LBEs. The staff's evaluation of the PSAR's preliminary analyses of normal radioactive effluent source terms is described in SE chapter 9.

The staff's review found that the PSAR LBE scenarios include many conservatively biased assumptions in the progression including the amount and timing of fuel damage or other MAR release rates. The LBE MST analyses also include many conservatively biased assumptions and code input to model the radionuclide transport across the release barriers and radionuclide retention and removal phenomena within the sodium coolant and in the gas space. Given the preliminary nature of the reactor design and analyses for the CP application, the staff did not find it necessary to perform independent event progression and source term analyses with a comprehensive system analysis code to evaluate the PSAR. However, using information from the PSAR and the supporting MST analyses available in the audit, the staff performed calculations on the LBE MSTs to ascertain the effective radionuclide retention within the credited barriers for the event. This included estimation of the effective removal rate of each of the radionuclide transport and removal phenomena within those barriers, such as bubble scrubbing in the sodium pool, SFP scrubbing, aerosol natural deposition, and filtration systems. The staff's assessment of the LBE source terms and radionuclide retention aided in the staff's review of the applicant's use of the LMP process to classify SSCs and evaluate the relationship

to the retaining radionuclides fundamental safety function and the Sodium reactor functional containment strategy. Discussion of the staff's evaluation of the LBE descriptions follows in later sections of this SE chapter and the safety classification of SSCs is discussed in SE chapter 5.

The staff observed in the audit that the development of source terms was inconsistent when the RADTRAD code was used with conservative assumptions for the DBAs when compared to using the [] and more detailed models (e.g., for pool scrubbing) for the non-DBA LBEs. The staff also noted that although the same base information was referenced [], the analyses used different user input based on those sources. In audit discussions, the applicant acknowledged these inconsistencies and stated that as they develop the scenario-specific source terms to support the OL application, they anticipate resolving the differences and providing the necessary justifications. The staff asked questions on the details of radionuclide transport and retention modeling during the audit, including comparison to the staff's scoping analyses to gain understanding of the PSAR source term analyses. In response to these audit discussions, USO provided supplemental information by letter dated September 17, 2025 (ML25260A002). In this letter USO states:

Additionally, mechanistic source term evaluation model treatment of sodium pool scrubbing and iodine releases will be further examined for the OL stage to confirm the implementation and supporting methodology is appropriate for the application.

Accordingly, based on its review of MST information in the PSAR and docketed supplement, as confirmed through audit, the staff determined the MST analyses are adequate at the CP stage. The staff will review updated MST analyses as part of the OL application.

MST Modeling of Radionuclide Release Barriers and Functional Containment

As described in SECY-93-092, an MST is “the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.” In the development of MSTs for the PSAR, the applicant modeled the release rates from the initial barrier (e.g., fuel pin) into subsequent spaces defined by physical barriers (e.g., reactor vessel system, head access area, reactor building) for the relevant event. Release rates across these barriers were based on preliminary design information or with conservative assumptions, with the expectation that these barrier release rate assumptions will be updated as necessary based on the final design for the FSAR analyses. The modeling of radionuclide transport and removal phenomena within the barriers (e.g., sodium pool aerosol scrubbing) implements the NAT-9392-A radiological source term methodology.

SECY-18-0096, which was approved by the Commission, describes the establishment of functional containment performance criteria for non-LWRs and provides a high level definition of functional containment as “a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment.” Enclosure 2 to SECY-18-0096 describes a proposed technology-inclusive, risk-informed, performance-based approach to derive functional containment performance criteria. The staff's review of the acceptability of the functional containment approach for KU1 is discussed in section 1.3 of the SE.

The MST modeling of the radionuclide barriers and release rates for the non-DBA LBEs and DBAs informs the functional containment performance criteria by providing details on the barrier performance assumed in the radiological consequence analyses, in particular for the SR and risk-significant non-safety-related with special treatment (NSRST) release mitigation SSCs as determined through the LMP process. NAT-9392-A section 2.3.5 describes the functional containment for the Natrium design as consisting of the physical system boundaries or structures for which leakage performance can be specified in design and verified by testing or associated analysis. NAT-9392-A further described a primary functional containment boundary as consisting of “the minimum set of barriers encompassing the core and primary system which prevents a release of radionuclides from exceeding 25 rem TEDE at the [low population zone (LPZ)] or 25 rem TEDE during the worst 2-hr period at the EAB during a Design Basis Accident.” SSCs along this boundary are established as SR. In the Natrium design, the functional containment depends on the operating mode and configuration and comprises a primary barrier and an enveloping barrier. As stated in NAT-9392-A:

A primary barrier is an SSC, or portion of an SSC, which is required to perform a radionuclide retention function to keep offsite doses of DBA scenarios within 10 CFR 50.34 limits and/or keep the consequence of the associated DBE from violating the F-C target curve. SSCs serving as a primary barrier are safety related.

and:

An enveloping barrier is an SSC, or portions of a SSC, which provides a backup radionuclide retention function to primary barriers or the primary functional containment boundary, in the event of leakage or failure of the primary barriers it envelopes. Enveloping barriers working alone or in tandem with other barriers limit the radiological release in AOOs, DBEs, and BDBEs to below established limits of the event type. Typically, enveloping barriers are NSRST or NST.

The Natrium reactor SSCs that are listed as providing functional containment boundaries are discussed in PSAR section 1.3.2.1. PSAR figure 1.3-1 illustrates the diverse barriers in functional containment.

The barriers modeled in the MSTs which provide the radionuclide retention function for the in-vessel LBEs are the reactor vessel and reactor vessel head, the head access area (HAA) boundary, and for some events the reactor building. For the fuel handling events, the EVHM barrier (which is modeled as conservatively representing the BLTC, pin removal cell (PRC), EVST, and PIC for the ex-vessel fuel handling events) is added. Other ex-vessel release events conservatively model the releases to the environment, with some events modeling some holdup in the reactor auxiliary building (RAB) or reactor building (RXB) (including SPS cell, or vapor trap cell), depending on location and some including filtration of aerosols and release by the of NI Heating, Ventilation, and Air Conditioning System (NHV). For additional information on radionuclide release barriers, table 5.5-1 of this SE summarizes the radionuclide retention functions mapped to the MAR they are mitigating and the associated LBEs. The assumptions made on the barrier leakage rates in the MST analyses to support the FSAR will inform the development of the Technical Specification 3.6 “Functional Containment” limiting conditions for operation and surveillance requirements for the OL stage, as listed in PSAR table 11.5-1. The following discussion of the staff’s review of the PSAR source terms includes description of the radionuclide release barriers, including those credited as functional containment barriers in the source term analyses.

3.2.1.1 In-vessel At-power Event Source Terms

The MSTs for in-vessel at-power events are summarized in PSAR tables 3.2-1, 3.2-2, and 3.2-4. These MSTs are representative of LBEs and DBAs with an intact vessel. These events include breach of the fuel cladding, but do not result in fuel melting. The local fault events result in cladding damage to one fuel assembly, while the MST for the protected loss of flow (PLOF) or loss of offsite power (LOOP) events is based on damage to the cladding in 2/3 of the core.

Each of the in-vessel at-power event source terms assumes instantaneous release of the radionuclides present in the fuel gas plenum of the pins in the damaged fuel assemblies. Through the audit of supporting analyses, the staff observed that the [[

]]. The audited information confirms that the MST analyses appropriately justify the specific information used from the referenced report and the information is used in a conservative bounding manner for the DBAs or is used in parameter distributions within the [[]]] for the other LBE analyses, which employ best-estimate with uncertainty methods.

In the applicant's analysis to develop the LBE source terms in PSAR tables 3.2-1 and 3.2-2, the [[

]]. Through evaluation of the PSAR and the referenced reports, the staff determined that the fuel radionuclide release fraction information is relevant to the Natrium design conditions for in-vessel at-power events. The audited information confirms that the MST analyses appropriately justify the specific information used from the referenced reports and the information is used in a conservative bounding manner for the DBAs or is used in parameter distributions within the [[]]] for the other LBE analyses.

After the release from the fuel, the next functional containment barrier is the reactor vessel system, including the reactor vessel head. The MST analyses model the radionuclide transport through the sodium pool to the cover gas space, including bubble transport and aerosol scrubbing. The analyses follow the NAT-9392-A methodology with conservatively biased assumptions to model the radionuclide transport and retention phenomena in the sodium pool. The LBE source terms summarized in PSAR tables 3.2-1 and 3.2-2 modeled the sodium pool scrubbing using the [[]]] detailed bubble transport model which calculates a sodium pool decontamination factor (DF) [[

]]. The staff observed in the audit that the MST analyses provided justification for the pool scrubbing model assumptions and found through its assessment of the referenced reports that the modeling of sodium pool scrubbing is appropriate for the Natrium PSAR.

For the [[

]]. The staff determined this assumption is acceptable because it is consistent with the NAT-9392-A methodology in which the staff found the use of the gas space aerosol removal coefficient value acceptable for use in DBA MST analyses if justified.

Leakage across the intact reactor vessel head barrier from the cover gas space to the HAA is modeled as an assumed leakage rate of 1 percent of the cover gas volume per day. This barrier is part of the functional containment and the leakage rate will be considered in the development of functional containment performance criteria and technical specifications subject to staff evaluation in the review of the FSAR in support of the OL application.

The HAA boundary is the enveloping functional containment barrier for the in-vessel at-power events. For the LBE source terms summarized in PSAR tables 3.2-1 and 3.2-2, the analyses model aerosol deposition through the Henry correlation for gravitational settling as described in NAT-9392-A. The staff audit of the supporting MST analyses showed that the user inputs to the model in the calculational code were based on preliminary design information (i.e., assumed deposition height in the HAA), or were assumptions that are consistent with data on sodium aerosols formation in air (e.g., aerosol density of sodium oxide). The local fault DBA MST analysis conservatively did not model aerosol deposition in the HAA.

Leakage from the intact HAA space was modeled directly to the environment (no credit for the RXB) as an assumed leakage rate of 10 percent of the HAA volume per day. The local fault DBA assumed the degraded HAA barrier leaks to the environment at an elevated rate of 100 percent of the HAA volume per day. For the in-vessel at-power events, the HAA barrier is part of the enveloping functional containment to provide DID to the primary functional containment and

the leakage rate will be considered in the development of functional containment performance criteria and technical specifications (TSS) subject to staff evaluation in the review of the FSAR in support of the OL application.

As discussed in detail in section 5.4.3 of this SE, an SR SCG isolation function was added by USO during the review. As noted in the PSAR, the dose consequences for the local fault events, the PLOF BDBE, and the LOOP BDBE do not include radionuclide release from SCG release pathways. The consequence analyses for these events will evaluate release from SCG release pathways, including the mitigation from SCG isolation, at the OL stage.

During audit discussions the staff noted that the PSAR section 3.8.6.2.2 MST description of the leakage rates from the reactor vessel head as primary functional containment and HAA (as modeled in the PSAR table 3.2-4 source term) were not consistent with the description of the barriers in the supporting analysis of the core blockage and local faults with reactor vessel head failed BDBE. Additional discussion of the staff's evaluation of this inconsistency and the applicant's resolution can be found in SE section 3.7.1.6.

The staff determined through review of the PSAR, and confirmed through the audit of supporting documentation, that the PSAR tables 3.2-1, 3.2-2, and 3.2-4 source terms were developed with conservatively biased radionuclide release and transport assumptions to reasonably represent at-power in-vessel events. The following table summarizes the information used to develop the at-power in-vessel event source terms:

Table 3.2-1: Key Inputs for At-Power In-Vessel Event Source Terms

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3.2.1.2 In-vessel Fuel Handling Event Source Terms

The MSTs for in-vessel fuel handling events are summarized in PSAR tables 3.2-3, 3.2-5, and 3.2-6. These MSTs are representative of fuel handling LBEs and DBAs within the reactor vessel, with the ex-vessel handling machine attached. The source terms assumed damage to a high burnup assembly, with the radionuclide inventory decayed to reflect the event occurring at 0 or 2.5 days post-shutdown, as applicable to the event scenario. These events assume the breach of all cladding on the dropped fuel assembly and in some cases a second impacted fuel assembly. The modeling of releases from the fuel, sodium pool scrubbing, and aerosol deposition in the cover gas space and RXB (instead of the HAA) are the same as described

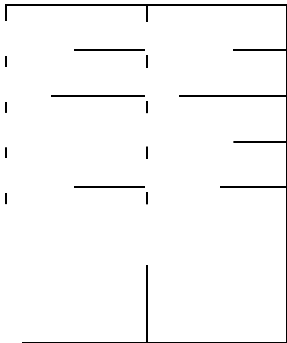
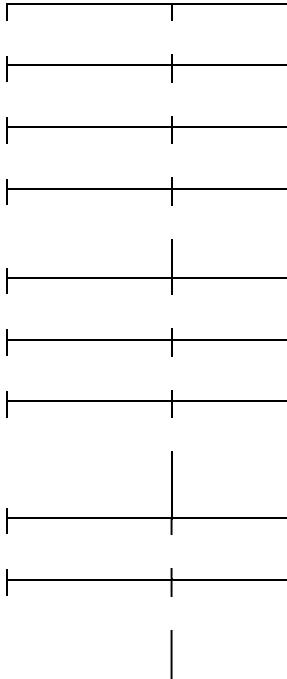
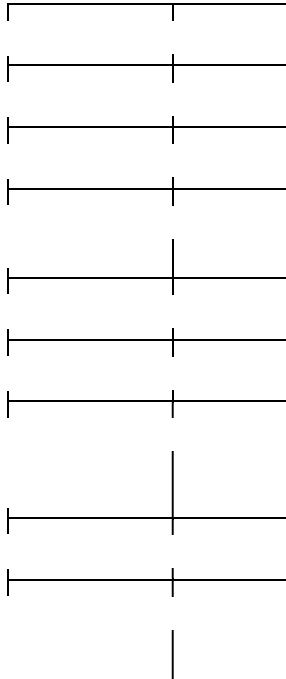
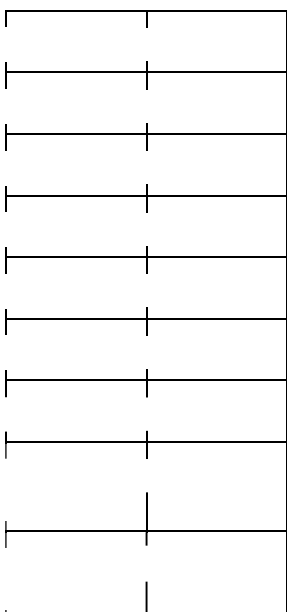
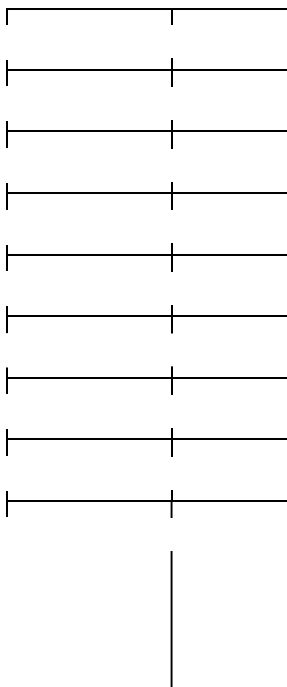
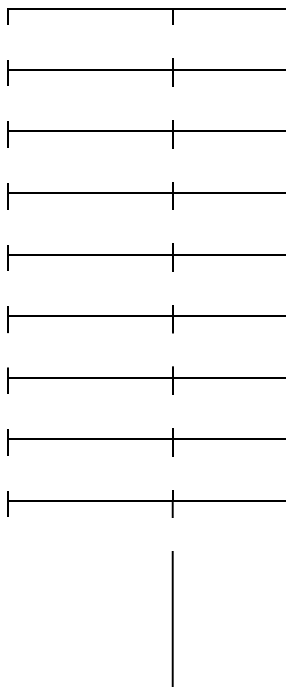
above for the in-vessel at-power events, with adjustments related to the time after shutdown and increased leakage for degraded barriers in the in-vessel fuel drop BDBEs. The source term for the in-vessel fuel drop BDBEs assumes the leakage from the RVH degraded primary barrier is 100 percent of the cover gas volume per day to an enveloping barrier in the RXB which is made up of a portion of the RXB volume. The leakage from the RXB is 100 percent of partial RXB volume per day. The DBA and DBE in-vessel fuel drop accidents model the release directly from the vessel to the environment, without credit for the HAA or RXB.

As discussed in detail in section 5.4.3 of this SE, an SR SCG isolation function was added by USO during the review. As noted in PSAR sections 3.7.4.2, 3.7.4.3, and 3.9.5.1, the dose consequences for the in-vessel fuel handling events with successful functional containment do not include radionuclide release from SCG release pathways. The consequence analyses for these events will evaluate release from the SCG release pathways, including the mitigation from SCG isolation, at the OL stage.

The staff determined through review of the PSAR and confirmed through the audit of supporting documentation that the PSAR tables 3.2-3, 3.2-5, and 3.2-6 source terms were developed with conservatively biased radionuclide release and transport assumptions to reasonably represent in-vessel fuel handling events. The following table summarizes the information used to develop the in-vessel fuel handling event source terms:

Table 3.2-2: Key Inputs for In-Vessel Fuel Handling Event Source Terms

II

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3.2.1.3 Ex-vessel Fuel Handling Event Source Terms

The MSTs for ex-vessel fuel handling events are summarized in PSAR tables 3.2-7, 3.2-8, and 3.2-9. These MSTs are representative of fuel handling LBEs and DBAs occurring in locations other than the reactor vessel. The source terms assumed damage to a high burnup assembly, with the inventory decayed to reflect the event occurring at 2.5, 210, or 310 days post-shutdown, as applicable to the event scenario. These events assume the breach of all cladding on the affected assemblies or fuel pins, with instantaneous release of the radionuclides. Because of the small volume and limited amount of sodium covering the assembly in the EVHM the source term development does not model aerosol scrubbing or radionuclide retention in the sodium or aerosol deposition in the cover gas space.

The source term for releases from spent fuel or the LTA/LDA within the EVHM was used as representative for fuel handling events in other locations during the fuel handling operations,

such as the BLTC, EVST, and PIC. During the audit, the staff confirmed that this is a reasonable modeling choice because the supporting analyses provide justification that shows the modeling of the release from the EVST is conservative based on preliminary design information. The applicant plans to determine scenario-specific source terms using final design information to support the FSAR analyses.

The staff observed in the audit of the [[

]]:

- PSAR section 3.8.5.5,
- PSAR section 3.8.5.9, and
- PSAR section 3.8.5.15.

In PSAR table 3.2-8 [[

]]. The applicant plans to determine scenario-specific source terms using final design information to support the FSAR analyses.

PSAR section 3.6.4.1 describes the [[

]].

The spent fuel drop in the SFP events are represented by the MST summarized in PSAR table 3.2-9. These events are the DBE in PSAR section 3.7.4.4, BDBE in PSAR section 3.8.5.3, and the fuel handling accident (FHA) in the SFP DBA in PSAR section 3.9.5.2. The MST assumed cladding failure of all pins in a high burnup assembly which has undergone 310 days of decay. For the PSAR section 3.8.5.3 fuel handling event in the SFP consequence analysis, the MST was doubled to account for the failure of two spent fuel assemblies. The radionuclides in the gas plenum of the fuel pins are released instantaneously to the SFP with credit for aerosol scrubbing in the water pool and subsequent instantaneous release from the SFP to the fuel handling building (FHB). There is no credit for FHB holdup or retention in the assumed instantaneous release from the FHB to the environment. The staff determined these release assumptions in the modeling of the fuel drop accidents to be conservative.

As stated in PSAR sections 3.7.4.4 and 3.9.5.2, when the fuel cladding is damaged, the fuel pin sodium bond undergoes a gas-producing exothermic reaction with the water in the SFP. The PSAR also states that the applicant is studying the nature of the reaction for further characterization of the radionuclide retention in the SFP. In the PSAR MST analysis, the modeling of the SFP is informed by RG 1.183 Appendix B guidance on radionuclide retention in the SFP for LWR FHAs, with a reduced DF to address the sodium bond-water reaction. The staff [[

]]. The staff's judgment is that the modeling of radionuclide retention in the SFP is reasonable for the PSAR taking into consideration the applicant's continuing investigation of the phenomena.

Excessive sodium-water reaction (ESWR) events in the PIC were represented by the source terms for fuel handling events in the EVHM or the SFP. The source term for the fuel handling event in the EVHM was used as representative of the PSAR section 3.9.5.3 ESWR DBA and the PSAR section 3.8.5.5 ESWR BDBE, where there is no credit for sodium scrubbing or holdup or retention in the enveloping containment barriers such as the FHB. The PSAR table 3.2-9 source term for the spent fuel drop in the SFP was used as representative of the PSAR section 3.7.4.6 ESWR DBE to account for limited sodium scrubbing in the PIC. The [[

]].

By applying the FHA source terms to the ESWR events, the PIC functional containment barrier and PIC/BLTC mated barrier are effectively modeled as isolated. This is contrary to the event scenario which instead states that the PIC or PIC/BLTC fails to contain the radionuclide release and there is a filtered release from the NHV that is not isolated. In other words, the source term scenario applied differs significantly from the system design. As noted in the previous paragraph, the [[

]].

During the audit, USO identified that the PIC design has changed since the PSAR submittal and that the analysis of the ESWR events in PSAR are based on the outdated preliminary design information. By letter dated September 17, 2025 (ML25260A002), USO provided supplemental information to describe the differences in event progression modeling since the PSAR was submitted and the resulting effect on the ESWR LBEs. In that letter USO stated that event-specific source term models will be developed that represent the detailed system configuration of PIC in the final design to support the OL application. USO further stated that the development of the source term analysis methodology will include consideration of the pressure and temperature effects driving radionuclide releases from the fuel for events involving the sodium-water reaction. Taking into consideration the supplemental information describing the applicant's plans to develop event-specific MSTs based on final design information and based on its review of the preliminary design information and event descriptions in the PSAR and audited supporting documents, the staff determined that the application of the fuel handling event source terms to the ESWR events is reasonable for the PSAR.

The staff determined through review of the PSAR and confirmed through the audit of supporting documentation that the PSAR tables 3.2-7 through 3.2-9 source terms are developed with conservatively biased radionuclide release and transport assumptions to reasonably represent ex-vessel fuel handling events. The following table summarizes the information used to develop the ex-vessel fuel handling event source terms:

Table 3.2-3: Key Inputs for Ex-Vessel Fuel Handling Event Source Terms
[I]

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3.2.1.4 Ex-vessel Release Event Source Terms

The remainder of the MSTs listed in PSAR section 3.2 are used in the analysis of other ex-vessel release LBEs. The staff's evaluation of these MSTs is described below in three groups based on the systems from which the releases occur: SPS, SCG, and gaseous radwaste processing system (RWG).

Primary Sodium Processing System (SPS-P), Intermediate Sodium Processing System (SPS-I) releases

The MSTs for SPS release events are summarized in PSAR tables 3.2-10, 3.2-11, and 3.2-18.

The PSAR table 3.2-10 source term is representative of a DBE leak from the primary sodium processing system (SPS-P), either to the RXB or RAB. The MST is modeled with the assumption that a small break in the SPS-P results in an airborne release equivalent to 20 gallons of aerosolized primary sodium to the SPS cell in the RAB. The staff performed calculations to determine the quantity of sodium aerosol produced from leakage into an inert cell and identified that the quantity assumed in the analysis is conservative relative to the quantity of sodium expected to be available for release into the SPS cell. The intact SPS cell leaks to the RAB at a leakage rate of 10 percent of the SPS cell volume per day, and the RAB exhausts fully to the environment every two hours.

The PSAR table 3.2-11 source term is representative of a DBA leak from the SPS-P, either to the RXB or RAB. The MST is modeled the same as the SPS-P leak DBE source term with the exception that the SPS cell barrier is assumed to be degraded. The SPS cells leaks to the RAB at a leakage rate of 50 percent of the SPS cell volume per day, and the RAB exhausts to the environment instantaneously.

Because the dominant isotope for these source terms is sodium-24, which has a half-life of approximately 15 hours, the leak rate assumptions of 10 percent and 50 percent have significant impacts on the source term results. Further discussion on the impacts to specific LBEs can be found in SE sections 3.6.1.1 and 3.8.1.4. As discussed in SE section 5.1, USO will need to provide bases for the nominal and degraded performances of radionuclide retention barriers used in the source term analyses at the OL stage.

The PSAR table 3.2-12 source term is representative of either the [I

]].

The staff determined through review of the PSAR and confirmed through the audit of supporting documentation that the PSAR tables 3.2-10, 3.2-11, and 3.2-18 source terms reasonably represent SPS release events. Assumptions for the amount of sodium vaporized and amount of tritium released appear conservative. However, assumptions for nominal and degraded leak rates of the SPS cell may be non-conservative and will be evaluated further at the OL stage when additional design information is available. The following table summarizes the information used to develop the SPS release event source terms:

Table 3.2-4: Key Inputs for SPS Release Event Source Terms

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Sodium Cover Gas System (SCG) Releases

The MSTs for SCG release events are summarized in PSAR table 3.2-12, table 3.2-13, and table 3.2-14. These MSTs are representative of SCG leaks and are applied to SCG leak DBEs (with confinement success) and BDBEs (with confinement bypass) either inside the HAA or downstream of the SCG cell, and the SCG leak DBA.

The three SCG release event MSTs are each modeled as an instantaneous release of the entire cover gas radionuclide inventory to the HAA using the HAA leakage rate either to the RXB or directly to the environment.

For the PSAR table 3.2-12 SCG leak with confinement success DBE source term, the releases are filtered by the NHV using preliminary design NHV flow rates from the HAA to the environment before NHV isolation at 30 minutes and then from the RXB after NHV isolation. The release rate from the HAA to the RXB after NHV isolation is assumed to be 10 percent of the HAA volume per day.

The PSAR table 3.2-13 SCG leak with confinement bypass DBE source term assumes bypass of the RXB. The release from the HAA to the environment credits filtration by the un-isolated NHV.

For the cover gas release DBA, the PSAR table 3.2-14 source term assumes the HAA volume fully exhausts to the environment over a 2-hour period, with no filtration or credit for other radionuclide removal processes.

The staff determined through review of the PSAR and confirmed through the audit of supporting documentation that the PSAR tables 3.2-12 through 3.2-14 source terms were developed with conservatively biased radionuclide release and transport assumptions to reasonably represent SCG release events. The following table summarizes the information used to develop the SCG release event source terms:

Table 3.2-5: Key Inputs for SCG Release Event Source Terms

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Gaseous Radwaste Processing System Releases

The MSTs for RWG release events are summarized in PSAR table 3.2-15, 3.2-16, and 3.2-17. These MSTs are representative of releases from the RWG and are applied to the DBEs (PSAR tables 3.2-15 and 3.2-16) or DBA (PSAR table 3.2-17).

The three RWG release event MSTs each model the release of the entire RWG holdup tank radionuclide inventory, either to the tank vault in the FHB or to downstream RWG piping.

For the RWG release DBEs, the releases to the environment are filtered. The DBA source term assumes the release to the tank vault is released directly to the environment over a 10 minute period via the plant stack with the filter bypassed.

The staff determined through review of the PSAR and confirmed through the audit of supporting documentation that the PSAR tables 3.2-15 through 3.2-17 source terms were developed with conservatively biased radionuclide release and transport assumptions to reasonably represent RWG release events. The following table summarizes the information used to develop the RWG release event source terms:

Table 3.2-6: Key Inputs for RWG Release Event Source Terms

[[

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3.2.2 Conclusion

The staff reviewed the event-specific MST information described in PSAR section 3.2 and determined that the information is consistent with the guidance in RG 1.233, RG 1.247, and RG 1.253, and used an approved methodology NAT-9392-A to develop MSTs for KU1. The staff determined that the information provided in the PSAR regarding the use of MSTs in the analysis of LBEs: (1) is consistent with the guidance in RG 1.253 and NEI 21-07; (2) provides a comprehensive and technically acceptable description of the technical basis for the

event-specific MSTs used in the safety analysis; (3) identifies the applicant's plans for finalizing the source terms prior to submitting the OL; and (4) supports the safety conclusions made in SE chapters 3 and 4. Therefore, the staff determined that the use of the MSTs described in the PSAR supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1), (4), and 10 CFR 50.35, as applicable.

3.3 Licensing Methods for Evaluation of Licensing Basis Events

PSAR section 3.3 provides an overview of the methodologies used in the evaluation of LBEs including a description of the computer programs and calculation framework with a specific set of transients or accidents, and provides information such as the mathematical models used, assumptions included in the programs, and procedures for treating program inputs and outputs. The methodologies also include required assumptions about plant equipment availability and any other information necessary to specify the calculation procedure.

PSAR section 3.3 describes the methodology used in the evaluation to select and categorize LBEs. Analysis of DBAs, including event descriptions and mapping to DBEs, is addressed in PSAR section 3.9. Other LBEs (e.g., AOOs, DBEs and BDBEs), including event description, frequency, and, if applicable, consequence estimates, are addressed in PSAR sections 3.6, 3.7 and 3.8.

PSAR section 3.3.1 describes the analytical methods and their qualification used to evaluate DBAs. PSAR section 3.3.2 describes the analytical methods and their qualifications used to evaluate AOOs, DBEs, and BDBEs. PSAR section 3.3.3 describes the analytical methods and their qualification used to evaluate the major accident to meet the requirement of 10 CFR 50.34(a)(1)(ii)(D). The LBE plant response and analysis overview is given in PSAR section 3.3.4.

The applicable regulatory requirements for the evaluation of the DBA analytical methods are as follows:

- 10 CFR 50.34(a)(1) and (4),
- 10 CFR 50.35, and
- 10 CFR 50.43(e).

The applicable guidance for the evaluation of the DBA analytical methods are as follows:

- RG 1.203, "Transient and Accident Analysis Methods," Rev. 0 (ML050230008);
- RG 1.233;
- RG 1.253;
 - RG 1.253 item C.3.b states that an applicant needs to address the requirements of 10 CFR 50.34(a)(1)(ii)(D) for a major accident deterministic analysis.

3.3.1 Technical Evaluation

PSAR section 3.3 discusses five methodologies used to analyze DBAs, which were submitted to the NRC and approved for use with L&Cs. The PSAR incorporates the following TRs by reference:

- NAT-9390-A, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," Rev. 2 (ML25211A127), sections 1 through 8
- NAT-9391-A, sections 1 through 6
- NAT-9392-A, sections 1 through 8
- NAT-9394-A, "Design Basis Accident Methodology for Events with Radiological Release," Rev. 0 (ML25251A090) sections 1 through 7
- NAT-9395-A, "Partial Flow Blockage Methodology," Rev. 0 (ML25251A084) sections 1 through 7

The methodologies in NAT-9390-A, NAT-9394-A, and NAT-9395-A evaluate in-vessel DBAs without radiological release, in-vessel and ex-vessel DBAs with potential radiological release, and partial flow blockage LBEs, respectively. The DBA without radiological release methodology analyzes in-vessel DBAs against defined screening criteria to determine if a transient causes fuel failure. The partial flow blockage methodology uses similar criteria as the DBA without radiological release methodology to determine if a partial flow blockage causes fuel failure. If these screening criteria are exceeded for a given transient in either methodology, the transient is analyzed using the DBA with radiological release methodology. This methodology determines the extent of cladding or fuel failure and also quantifies liquid sodium or gas leaks, which are inputs into the source term methodology discussed in NAT-9392-A. The output of the source term methodology is radiological releases to the atmosphere (source terms), which are input to the radiological consequence methodology discussed in NAT-9391-A.

3.3.1.1 DBA without Radiological Release Methodology

In the SE for NAT-9390-A, the staff determined that the DBA without radiological release methodology provides an acceptable approach for future applicants using the Natrium design to evaluate in-vessel DBAs without radiological release, subject to L&Cs. The staff evaluated these L&Cs to confirm they were either met or evaluated as part of the PSAR. The staff's SE for NAT-9390-A imposed the following L&Cs:

1. The NRC staff's determinations in this SE are limited to the Natrium design described in section 1.2 of the TR and this SE, including the use of Natrium Type 1 fuel. An applicant or licensee referencing the methodology developed in this TR must justify that any departures from these design features do not affect the conclusions of the TR and this SE. Additionally, this methodology was developed to analyze certain DBAs as discussed in TR section 2.1 and this SE (and as defined in NEI 18-04 [9]); use of this methodology for other kinds of analyses must be justified.
2. The NRC staff noted that execution of the steps 6, 7, 8, 9, 12, 13, 14, 15, 16, 17, 18, 19, and 20 of the EMDAP, as well as sensitivity studies discussed in section 2.5 of the TR

and section 3.1.4 of this SE, have not been completed. An applicant or licensee referencing the methodology developed in this TR must submit documentation and justify that these steps of the EMDAP have been completed to a state that is appropriate for the intended licensing application.

The staff's evaluation of the L&C implementation is summarized in the bullets below:

- The staff determined that the PSAR's use of NAT-9390-A meets L&C 1 because the preliminary Natrium design described in the PSAR is consistent with the design information in the TR and because the PSAR uses NAT-9390-A to analyze in-vessel DBAs identified through the NEI 18-04 LMP methodology.
- For L&C 2, the steps that have not yet been completed relate to the verification and validation of the EM, which, as discussed in DANU-ISG-2022-01 (ML23297A158) is not required to be complete for a CP application because it represents research and development work. As such, the staff determined that the methodology in NAT-9390-A is sufficiently developed for preliminary analysis for a CP application, and thus the USO adequately addresses L&C 2.

Since USO addressed all L&Cs in NAT-9390-A as described in the PSAR, the staff determined that the use of the DBA without radiological release methodology for analyses supporting the PSAR is acceptable.

3.3.1.2 DBA with Radiological Release Methodology

In the SE for NAT-9394-A, the staff determined that the DBA with radiological release methodology provides an acceptable approach for future applicants using the Natrium design to evaluate in-vessel and ex-vessel DBAs with radiological release, subject to L&Cs. The staff evaluated these L&Cs to confirm they were either met or evaluated as part of the PSAR. The staff's SE for the NAT-9394-A imposed the following L&Cs:

1. The NRC staff's determinations in this SE are limited to the Natrium design described in section 2.2 of the TR and this SE. An applicant or licensee referencing the EMs developed in this TR must justify that any departures from these design features do not affect the conclusions of the TR and this SE. Additionally, this methodology was developed to analyze certain DBAs as discussed in TR section 1.0 and this SE (and as defined in NEI 18-04); use of this methodology for other kinds of analyses must be justified.
 - a. For the FHA EM, the NRC staff's determinations are limited to the Natrium design [[

]].
 - b. For the in-vessel transients with radiological release, partial flow blockage, FHA, and fuel misload EMs, the NRC staff's determinations in this SE are limited to the Natrium design using Natrium Type 1 fuel.
2. As discussed in section 2.4 of the TR, the DBAs with radiological release methodology does not contain event-specific EMs for events associated with excessive sodium-water reaction in the PIC, loss of EVST cooling while storing fuel assemblies, and leakage from the RWG. Use of this methodology for these events requires further justification.

3. Section 2.4 of the TR states that “DBAs which are not in-vessel are evaluated using the appropriate methodology in the DBA with release EM, an appropriate event-specific method, or evaluated with the source term EM using conservative assumptions.” As such, applications involving ex-vessel release analyses referencing this TR for their basis must provide sufficient detail to demonstrate that the methodology used is suitable.
4. An applicant or licensee referencing this methodology must submit documentation and justify that code qualification, verification, and validation activities have been completed to a state that is appropriate for the intended licensing application for each of the EMs discussed in the TR.
5. Consistent with section 6.2 of the TR, applicants or licensees referencing this methodology must appropriately justify that the initial and boundary conditions and other input modeling parameter values are conservatively selected. This includes the selection of **[[** **]]**.
6. As discussed in section 5.1.3 of the TR, the applicability of the in-vessel transients with radiological release methodology for licensing analyses is restricted to those events that do not experience severe accident phenomena (e.g., coolant boiling, gross cladding failure, significant fuel melting and relocation).
7. An applicant or licensee referencing the methodology described in TR section 5.4 for performing detailed mechanical analysis for FHAs must submit documentation and justify that the development and assessment of this methodology has been completed to a state appropriate for the intended licensing application.

The staff’s evaluation of the L&C implementation is summarized in the bullets below:

- The staff determined that USO’s use of NAT-9394-A, as described in the PSAR, meets L&C 1 because the preliminary Natrium design in the PSAR is consistent with the design information in the TR and because the PSAR uses NAT-9394-A to analyze ex-vessel and in-vessel DBAs identified through the NEI 18-04 LMP methodology.
- With respect to L&C 2, the ESWR and RWG release events do not apply the NAT-9394-A methodology but instead directly apply the source term and consequence methodologies as discussed further in section 3.2.1.3 and 3.2.1.4 of this SE, respectively. Further, the loss of EVST cooling event does not apply the NAT-9394-A methodology but is instead analyzed with an event-specific heatup analysis demonstrating fuel does not fail after an extended period of time, as discussed in section 3.6.1.4 of this SE. As such, the staff considers L&C 2 satisfied because USO does not apply the EM discussed in NAT-9394-A.
- The staff determined that the use of this methodology for selected ex-vessel DBAs, not including the PIC and EVST events discussed previously, was appropriately justified as described in the PSAR and confirmed via audit of the supporting documentation, meeting L&C 3.
- For L&C 4, the staff determined that the code qualification, verification and validation activities discussed in NAT-9394-A for this methodology are sufficient for preliminary

analyses supporting a CP application for the same reasons discussed in section 3.3.1.1 of this SE and, thus, USO adequately addressed L&C 4.

- The staff determined that the applicant selected appropriately conservative input modeling parameter values for analyses using this EM as described in the PSAR and confirmed via audit of supporting documentation, which addresses L&C 5.
- For L&C 6, USO demonstrated in the PSAR that no DBAs analyzed with this EM experience severe accident phenomena, which was verified by the staff in audit, thus the staff determined USO adequately addressed L&C 6.
- USO discusses fuel handling DBAs in PSAR section 3.9.5, which assume an instantaneous release of all radionuclide inventory rather than using this methodology which the staff determined adequately addressed L&C 7.

Since USO adequately addressed all the L&Cs included in NAT-9394-A as described in the PSAR, the staff determined that the use of the DBA with radiological release methodology for analyses supporting the PSAR is acceptable.

3.3.1.3 Partial Flow Blockage Methodology

In the SE for NAT-9395-A, the staff determined that the contained methodology provides an acceptable approach for future applicants using the Natrium design to evaluate partial flow blockage LBEs, subject to L&Cs. The staff evaluated these L&Cs to confirm they were either met or evaluated as part of the PSAR. The staff's SE for the NAT-9395-A imposed the following L&Cs:

1. The staff noted that execution of steps 7, 8, 9, 13, 14, 15, 16, 17, 18, and 19, and adequacy decision of the EMDAP have not been completed. An applicant or licensee referencing the methodology developed in this TR must justify that these steps of the EMDAP have been completed to a state that is appropriate for the intended licensing application.
2. The staff's determinations in this SE are limited to the Natrium design described in the TR, including the operating conditions. An applicant or licensee referencing the methodology developed in this TR must justify that any departures from design features or operational conditions, such as core geometry, power, temperature, or flow rate, do not affect the conclusions of the TR and this SE.
3. Applicability of this TR is limited to the flow blockage events bounded by those explicitly identified and analyzed in section 2.1 of the TR. If an applicant implementing this methodology identifies a credible flow blockage event(s) not bounded by those defined in this TR, the applicant must justify the applicability of the TR methodology.

The staff's evaluation of the L&C implementation is summarized in the bullets below:

- For the same reasons provided in section 3.3.1.1 of this SE, the staff determined that the methodology in NAT-9395-A is sufficiently developed for preliminary analysis for a CP application, and USO, therefore, adequately addresses L&C 1.

- The staff determined that the USO's use of NAT-9395-A meets L&C 2 because the preliminary Natrium design in the PSAR is consistent with the design information in the TR.
- In NAT-9395-A, credible flow blockage events are identified and preliminary analysis demonstrates that no fuel failure is expected. For core blockage LBEs, the PSAR conservatively assumes all fuel pins in a single high burnup assembly fail. This conservative assumption bounds the flow blockage events discussed in the TR and the staff determined this adequately addresses L&C 3.

Since USO adequately addressed all the L&Cs included in NAT-9395-A, the staff determined that the use of the partial flow blockage methodology for analyses supporting the PSAR is acceptable.

3.3.1.4 Source Term Methodology

The source term methodology is described in section 3.2 of this SE.

3.3.1.5 Radiological Release Consequences Analysis Methodology

PSAR sections 3.3.1.4 and 3.3.2.2 state that the TerraPower radiological release consequences methodology TR describes the methodology used to evaluate the radiological consequences of DBAs and other LBEs, respectively. The PSAR incorporates by reference sections 1 through 6 of NAT-9391-A to describe the radiological consequence analysis methodology. In the SE for the TR, among other conclusions the staff determined that NAT-9391-A used dose estimation models consistent with those described in the generally applicable dose calculation methodology guidance in section 4 of RG 1.183. The staff evaluated the L&Cs identified in the radiological release consequences methodology TR to confirm the L&Cs were either met or evaluated as part of the PSAR. The staff's SE for the NAT-9391-A imposed the following L&Cs:

1. Application of the methodology in this TR with respect to the described deterministic and probability-based atmospheric dispersion modeling analyses and use of generic meteorological data is limited to sites within the contiguous United States unless technical justification for their applicability is provided.
2. The conclusions reached in this SE are not valid if a process other than that described in NEI 18-04 is used to perform the Natrium safety analysis.

The staff's evaluation of the L&C implementation is summarized in the bullets below:

- The KU1 site is located in the contiguous U.S., therefore the staff determined the use of the NAT-9391-A methodology for the PSAR analyses meets L&C 1.
- USO uses the NEI 18-04 LMP methodology in the safety analyses for the Natrium design, therefore the staff determined the application meets L&C 2.

The staff determined USO adequately addressed all the L&Cs included in NAT-9391-A. Additional considerations for the radiological consequence analysis methodology are discussed below.

The NAT-9391-A methodology includes EMs for radiological release consequence analyses given a source term input. The methodology used to develop DBA and non-DBA LBE event-specific source terms is discussed in PSAR section 3.2. The staff's evaluation of the LBE event-specific source terms described in the PSAR is given above in SE section 3.2.

Through audit of the LBE consequence analyses, the staff confirmed that the analyses used methodology tools, inputs, and assumptions consistent with the approved NAT-9391-A methodology TR and used event-specific source terms as described in PSAR section 3.2. Through the audit the staff also confirmed that the site characterization used in the consequence analyses is representative of the Kemmerer 1 location. For the LMP process the LBE analyses must provide the uncertainty range on the consequence as the mean, 5th, and 95th percentile results. The staff confirmed in the audit that for the PSAR analyses, the majority of uncertain parameters in the consequence analyses were handled with bounding conservative values with the exception of the meteorological conditions and the source term uncertainty. The staff [[

]] The staff

determined the treatment of uncertainty in the PSAR consequence analyses is acceptable based on conservative methods or in accordance with approved methodology.

The staff's audit showed that the [[

]] The staff determined that the use of enveloping atmospheric dispersion factors in the DBA radiological consequence analyses to bound the site characteristic atmospheric dispersion is conservative and, therefore, acceptable.

Through audit of the LBE consequence analyses, the staff confirmed that the AOO, DBE, and BDBE radiological consequence analyses used generic meteorological data based on the EPRI Utility Requirements Document (URD) in the evaluation. As described in the SE for the approval of NAT-9391-A, the staff determined that the meteorological data in the TR LBE EM are representative of a reasonable number of sites in the contiguous U.S. that may be considered. The KU1 site is within the contiguous U.S. and therefore meets L&C 1 for use of NAT-9391-A. The staff also confirmed through audit that other quantified event dose analyses (for example used in the plume exposure pathway EPZ size determination) used the generic meteorological data based on the EPRI URD. However, the LBE and other quantitative event (OQE) consequence analyses did not explicitly show that the use of generic meteorological data is conservatively bounding for the KU1 site location as discussed in the radiological release consequences analysis methodology. PSAR section 3.3.2.2 states that at the OL stage the meteorological data used to evaluate the radiological consequences of AOOs, DBEs, and BDBEs must be shown to be conservatively representative of the site-specific meteorological data as described in section 3.6.1 of NAT-9391-A. The staff expects that use of the generic meteorological data based on the EPRI URD in the non-DBA LBE and OQE consequence analyses will result in doses that are likely to be bounding for KU1 and sufficient to use in the

NEI 18-04 risk-informed process. Additionally, the staff anticipates the consequence analyses supporting the FSAR for the OL application will include specific justification that the generic meteorological data and modeling of the atmospheric dispersion are representative of conditions at the KU1 site.

TR section 3.3.2.2 identifies that the PSAR analyses deviate from the NAT-9391-A method for estimating the consequences of chronic exposure. The long-term phase was not evaluated for the AOOs, DBEs, and BDBEs. Only the evaluation of the risk of latent cancer fatality is affected by the omission of this exposure pathway. The PSAR states that the increase in consequence is expected to be well within the margin to the LMP criterion for the integrated latent cancer risk presented in PSAR section 4.1.3 and that the long-term chronic exposure will be evaluated at the OL phase. Based on its experience with consequence analysis, the staff considers that the deviation from the consequence methodology with respect to the chronic exposure is not likely to challenge the LMP individual risk of latent cancer cumulative risk target. Therefore, the staff determined that the PSAR analyses are acceptable with this deviation, given that the FSAR analyses are anticipated to correct the omission of the long-term exposure in the final calculation of the risk of latent cancer fatality.

Based on its review as described above, the staff determined that the methodology to evaluate the radiological consequences of DBAs and other LBEs is acceptable and provides results that are representative of the potential radiological consequences of events at KU1 for use in the LMP consistent with the guidance in RG 1.233 that are sufficient to address the safety analysis regulatory requirements in 10 CFR 50.34(a)(1) and (a)(4).

3.3.1.6 Licensing Basis Event Analysis Methodologies

PSAR section 3.3.2.1 states the methodologies used to analyze AOOs, DBEs, and BDBEs are similar to the DBA methodologies described above, including the codes used for the analysis, but account for expected response of all SSCs (not just SR SSCs). The applicant also stated that LBEs could use a conservative approach to identifying scenarios or could use a best-estimate plus uncertainty (BEPU) approach. The PSAR provides a high level description of the BEPU approach for analyses, which is a hybrid approach that includes some conservatively biased parameters and some parameters that are sampled. The PSAR describes that a conservative approach was used for the system response portion of most non-DBA LBE analyses, with the exception of BDBEs DHP-LOOP-3 and -4. Several source term analyses also use the BEPU approach, including DBE LFF-SAO-BL and BDBEs DHP-LOOP-3 and -4 and RFH-FDIV-2 and -4. The staff identified these events as the in-vessel events that result in radionuclide releases. NAT-9392-A specifies a computer code for this kind of LBE which has uncertainty quantification capabilities as discussed in the TR, though those were not fully developed or implemented for the DBA analyses.

The staff verified by auditing non-DBA LBE analyses that the conservative approach described in PSAR section 3.3.2.1 is consistent with the DBA methodologies discussed above except that it credits non-SR SSCs. This is necessary to model the system response to with consideration of the successes and failures from the PRA. Because it is consistent with the approved methodologies, the staff considers this approach to be acceptable for the preliminary analyses included in the PSAR.

The staff audited the supporting documentation of the LBEs and the uncertainty analysis methodology to verify that the BEPU approach explicitly accounts for certain sources of uncertainty. The staff also verified that the uncertain parameters are consistent with the

parameters discussed in the DBA analysis methodologies described above. Finally, the staff verified that the uncertainty analysis methodology includes preliminary estimates for the parameter sampling distributions, which the staff notes need further justification to be finalized. In total, the staff determined that the BEPU methodology is reasonable for the PSAR preliminary analyses because it contains the elements needed to quantify uncertainties. However, the staff expects to review the uncertainty analysis methodology and the associated sampled parameter distributions in detail prior to issuance of an OL.

The staff notes that the applicant determined it was necessary to add a SR function for isolation of the SCG, as discussed further in section 5.4.3 of this SE. Application of this function is most relevant to those in-vessel LBEs resulting in releases, which are the same LBE analyses that employ the BEPU methodology as discussed above. Additionally, as noted in section 3.2.1 of this SE, some inconsistencies were identified in the MST EM treatment of sodium pool scrubbing. The applicant indicated they will confirm the implementation and supporting methodology is appropriate for the application prior to the OL. With these considerations, the staff expects that these events will be re-analyzed to consider the effects of the new SCG isolation function and confirm the impacts of pool scrubbing for the OL application.

3.3.1.7 Major Accident Analysis

RG 1.253 position C.3.b identifies that in addition to the LMP-based analyses, the safety analysis report should describe the analysis methods and assumptions for the total calculated radiological consequence dose to demonstrate that the facility meets the requirements of 10 CFR 50.34(a)(1)(ii)(D). Specifically, 10 CFR 50.34(a)(1)(ii)(D) requires, in part, that the PSAR analyses consider safety features engineered into the facility and barriers that must be breached as a result of an accident. The regulation further describes that an applicant shall perform an evaluation and analysis of the postulated fission product release from the core into the containment. Footnote 3 to 10 CFR 50.34 further describes that the assumed fission product release should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. The non-seismic site criteria in 10 CFR 100.21(c)(2) also states that site atmospheric dispersion characteristics must be evaluated and dispersion parameters established such that the radiological dose consequences of postulated accidents shall meet the criteria set forth in 10 CFR 50.34(a)(1). The plant design features intended to mitigate the radiological consequences of accidents, the site atmospheric dispersion characteristics, and the distances to the EAB and to the LPZ outer boundary are acceptable if the total calculated radiological consequences for the postulated fission product release meet the reference values for public dose given in 10 CFR 50.34(a)(1)(ii)(D).

RG 1.253, staff position C.3.b gives guidance on possible approaches to the radiological consequence analysis to address the regulatory requirements that may be considered in the context of an LMP-based approach. RG 1.253 staff position C.3.b provides two options but also states that applicants are free to propose different approaches. The two options provided are summarized as:

Option 1: Use the DBA dose consequence results from an LMP-based approach to establish the acceptability of the EAB and LPZ... [t]he DBA analysis under an LMP-based approach is a deterministic, conservative analysis that is analogous to the DBA analyses performed for new LWRs and operating reactors. Under this option, depending on the nature of the DBA, the application may need to include an exemption from the regulations in 10 CFR 50.34 or 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," that require an assumed "major accident" to

demonstrate containment performance and to confirm that the EAB and LPZ doses are below the reference values in the regulations.

Option 2: Use the greater of the dose consequence results from the bounding DBA and from a bounding BDBE, as identified in the LMP-based approach, to establish the acceptability of the EAB and LPZ... This option provides an acceptable approach to compliance with 10 CFR 50.34 and 10 CFR 52.79 that precludes the need for an exemption from these requirements, as long as the bounding BDBE involves or bounds an ES meeting the description of a major accident and the offsite consequences are below the reference values for public dose....

The major accident has historically been looked at as a conservative event to be analyzed with respect to dose to the public, and as a key event for identifying requirements for containments and related fission product release mitigating features for the facility. For large LWRs, the major accident has traditionally been a postulated core melt accident with substantial fission product release into the containment thus meeting the goals of being a bounding event and testing the key release mitigating feature of the facility (i.e., the containment). However, for advanced reactors use of the traditional analysis of the major accident developed for light-water reactors may not be possible or may not be needed to support design and licensing activities. Depending on fuel design, locations of MAR, and containment strategy (e.g., functional containment), the traditional definition and analysis for the major accident may not be applicable. Additionally, with the application of the LMP framework with a valuation of the range of possible LBEs and related concepts such as functional containment, the value provided by the definition of a single major accident is diminished.

As non-LWR designs were proposed by industry and evaluated by NRC in the late 1980s and early 1990s, it was recognized that the traditional approach to evaluation of the major accident for LWRs was not necessarily appropriate for these advanced reactor designs. The review approaches used in preapplication reviews for non-LWR designs documented in NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," (ML052780497) and NUREG-1368, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," (ML063410561) describe evaluation of a range of events using MSTs instead of evaluating only one single core melt scenario. A difference in the approach to containment for the PRISM was also addressed with the following statement from NUREG-1368:

Similarly, the review of a design without a conventional containment building was based on a mechanistic analysis of a range of low-probability events and on the potential for demonstrated capability of the design (via prototype testing) to perform as predicted. Inherent in this approach is a shift in emphasis in defense in depth from accident mitigation to accident prevention and plant protection.

While these advanced reactor projects were eventually cancelled, the preliminary evaluations demonstrate the staff's acceptance of a shift from evaluating a single bounding event using a predefined source term into a containment building to the evaluation of facility-specific bounding events using an MST approach. As a matter of policy, in the SRM to SECY-93-092 (ML003760774), the Commission approved the staff's recommendation that source terms for non-LWRs be based upon a mechanistic analysis. SECY-93-092 also states that the design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

PSAR section 3.3.3 describes the major accident deterministic analysis that is used to show compliance with the requirements of 10 CFR 50.34(a)(1)(ii)(D) for KU1. To characterize the major accident for KU1, the PSAR states that the applicant chose to use option 1 described in RG 1.253, staff position C.3.b. In the selection of the major accident, the applicant considered the subset of DBAs from the LMP analyses that are the in-vessel DBAs with radiological release. The applicant narrowed the consideration to in-vessel DBAs to be consistent with the 10 CFR 50.34(a)(1) description of the accident analysis as evaluating the postulated fission product release from the core into the containment. The applicant selected the DBA with maximum core damage and associated in-core release from the in-vessel DBAs as the major accident. The PSAR states that the core blockage and local faults DBA described in PSAR section 3.9.6.1 was selected as the major accident because it represents the potential hazard not exceeded by any other in-vessel DBE or in-vessel DBA. The core blockage and local faults DBA is an at-power in-vessel scenario with releases from core damage. The selection of the core blockage and local faults DBA as the major accident addresses the description of the major accident postulated fission product release in the footnote to 10 CFR 50.34.

The staff evaluated the description of the major accident in PSAR section 3.3.3, taking into consideration the purpose of the safety analysis described in 10 CFR 50.34(a)(1)(ii)(D), as well as its use in power reactor siting, as referenced by 10 CFR 100.21(c)(2). The staff's review included an audit of supporting information. Review of PSAR section 3.9 shows that of the nine DBA with release scenarios that are described, only the at-power in-vessel DBAs with core damage and release into the functional containment are consistent with the 10 CFR 50.34(a)(1)(ii)(D) description of a postulated fission product release from the core into containment. In the LMP, the DBA consequence analyses are performed using conservative deterministic assumptions and represent the plant response to credible events crediting only SR SSCs. These DBA analysis methods are similar to those traditionally used to show compliance with the offsite dose criteria in 10 CFR 50.34.

The staff notes that although the offsite radiological consequences of the selected major accident are bounding for at-power in-vessel DBAs, that is not true for the ex-vessel and fuel handling DBAs. This means that the consequences of the major accident analyzed by USO do not bound the consequences of all DBAs as was done in reactor licensing precedents. This would necessitate additional justification as discussed in footnote 10 to RG 1.253 option 1. The staff's review of the PSAR safety analyses found that, even in the preliminary evaluation for the CP application, the LMP provides a more detailed and comprehensive look at the plant design safety and siting than potentially given by a bounding major accident deterministic evaluation. The LMP evaluates a range of in-vessel and ex-vessel scenarios in all modes, including frequency and consequence uncertainty to evaluate the range of probable events, classify the safety classification and treatment of SSCs and evaluate DID. In addition, the LMP assesses the DBAs using deterministic methods to compare to against the 10 CFR 50.34(a)(1)(ii)(D) offsite dose criteria and assesses integrated plant risk. Review of the DBAs in PSAR section 3.9 shows that the offsite dose results for the selected major accident are higher than other at-power in-vessel DBAs.

For further assessment of possible credible accidental events, the staff looked at the PSAR DBEs with release determined through the LMP. Review of the DBEs in PSAR section 3.7 shows that the major accident 30-day dose at the LPZ (which is the same distance as the EAB) calculated using the conservative DBA consequence analysis methodology is higher than the 95th percentile 90-day doses at the EAB for the at-power in-vessel DBEs (and in fact all DBEs) using the more realistic non-DBA LBE consequence analysis assumptions. Because of the differing analysis methodologies, the LBE dose results should not be directly compared to the

DBA results to determine which release is bounding. However, considering the purpose of the major accident is to evaluate the safety design and siting, and the purpose of the LMP LBEs is to aid in the SE with explicit comparison of the each of the DBAs to the 10 CFR 50.34(a)(1)(ii)(D) offsite dose criteria, the staff's judgment is that the PSAR DBAs, as supported by evaluation of the LBEs, comprehensively address the safety and siting analyses consequence analysis requirement. Therefore, based on its assessment of the PSAR information in the context of the regulatory description of the required safety analysis, the staff determined the selection of the PSAR section 3.9.6.1 core blockage and local faults DBA as the major accident, as supported by the assessment of DBAs in the LMP methodology, appropriately addresses the purpose and intent of the 10 CFR 50.34(a)(1)(ii)(D) safety analysis of a major accident postulated from considerations of possible credible accidental events.

PSAR section 3.3.3 states that the radiological consequences at the EAB and LPZ for all the DBAs meet the 10 CFR 50.34 offsite dose criteria, using the DBA source term and radiological consequences methodologies described in PSAR sections 3.2 and 3.3.1, respectively. The staff's evaluation of the MST methodology is described in SE section 3.2, and its evaluation of the radiological release consequences methodology is described in SE section 3.3.1.5. The DBA analyses are discussed in PSAR section 3.9. The results show that the offsite dose consequences for each of the DBAs with radiological release meet the dose criteria in 10 CFR 50.34(a)(1)(ii)(D) with margin. This includes the results of the core blockage and local faults DBA selected as the major accident. Because the major accident analysis is chosen from the DBAs, based on the staff's determination in SE sections 3.2 and 3.3.1.5 that the DBA source terms and radiological consequence analyses methodologies are acceptable the staff also determined that the offsite radiological consequences of a major accident are acceptable and meet the offsite dose criteria in 10 CFR 50.34(a)(1).

Based on its review as described above, the staff determined that the PSAR described an evaluation of the major accident for the KU1 facility that addresses the safety analysis regulatory requirements related to the evaluation of the radiological consequences at the EAB and LPZ. The staff determined that the PSAR discussion of the major accident is consistent with the guidance for LMP-based safety analysis report contents given in RG 1.253. Therefore, the staff determined that the major accident analysis sufficient to address the safety analysis regulatory requirements in 10 CFR 50.34(a)(1)(ii)(D), which are also referenced by the non-seismic siting criteria in 10 CFR 100.21(c)(2).

3.3.1.8 LBE Plant Response and Analysis Overview

Section 3.3.4 of the PSAR provides a summary of the overall plant response to LBE IEs, the initial conditions assumed in LBE analyses, and the LBE analysis end states.

PSAR sections 3.3.4.1.1 through 3.3.4.1.7 provide overviews of the plant response to the different types of transients, including increase in heat removal, decrease in heat removal, reactivity, loss of primary flow, local fuel fault, release from ex-vessel systems, and fuel handling. These correspond to the analysis types that form each subsection of the LBE analysis sections in the PSAR. The staff found the plant response overviews to be useful because they describe the expected plant response including NST systems, which are not generally described in the PSAR in much detail. The staff reviewed the responses and found them to be consistent with the expected plant response to the different types of transients. The staff notes that PSAR section 3.3.4.1.6, which describes the release from ex-vessel systems LBEs, has minimal detail; this is reasonable because there is no generic plant response to these kinds of IEs, which rely solely on functional containment barriers to mitigate release.

PSAR section 3.3.4.2 provides the LBE analysis initial conditions. Chapter 3 of NEI 21-07 specifies that these should be provided for each transient, however, the staff determined it is reasonable to specify the initial conditions once, if they are used consistently between analyses. For transients involving the core, the conditions specified include a low-power condition, which is also used to represent shutdown conditions, and a full-power condition. The staff expects the plant to mostly operate in these states, because full-power operation is normal, and the low-power condition analyzed represents the plant condition following a runback. The staff also expects that events initiated from the low-power operating condition will bound those events at lower power but not fully shut down because they start at a higher power-to-flow ratio and would thus be conservative for loss of flow, loss of heatsink, etc. Finally, the staff identified that accidents occurring during startup conditions would be likely to depend on operating procedures that have not been developed at the CP stage. As such, the staff considers it reasonable for the CP to only analyze the two initial conditions discussed in the PSAR. However, the staff anticipates the OL will analyze all POSs, including plant startup, and provide additional justification that any POSs that are not explicitly analyzed are bounded by the analyzed conditions.

NEI 21-07 section 3.1 also specifies that definitions should be provided for the safe and stable end state of the analysis. Consistent with the guidance, the LBE analysis end states are detailed in PSAR section 3.3.4.3. The safe shutdown end state consists of a cladding temperature criterion and a reactivity criterion. The cladding temperature criterion is consistent with discussion in NUREG/CR-7305, "Metal Fuel Qualification: Fuel Assessment Using NRC NUREG-2246, "Fuel Qualification for Advanced Reactors", (ML23214A065) that fuel cladding integrity is expected to be maintained for temperatures below 650°C, and the staff therefore considers it to be acceptable. The reactivity criterion ensures the reactor is, at most, critical, and that power is stable or decreasing. The staff considers this to be appropriate for short-term transients, which typically result in reactor scrams and are significantly subcritical. For longer term transients, the PSAR introduces that heat removal needs to exceed the total heat produced; this ensures the reactor will be in a stable condition and is therefore acceptable. The PSAR also includes a separate safe and stable end state for transients that do not result in scrams of k_{eff} equal to one and heat removal equal to heat production. The staff determines this end state to be acceptable because it ensures, absent a perturbation to the system that the reactor will remain in a stable configuration.

The LMP methodology includes various steps where LBE consequence results are compared to the 10 CFR 50.34 dose criteria. The dose criteria in 10 CFR 50.34 include a 2-hour dose criterion of 25 rem TEDE at the EAB and a 25 rem TEDE criterion at the LPZ boundary when calculated for the full duration of the event release. The F-C target curve is anchored to 25 rem TEDE at a frequency of 10^{-4} /plant-year based on the 10 CFR 50.34 dose criteria, but the F-C target uses a 30-day TEDE at the EAB versus the 2-hour TEDE at the EAB in 10 CFR 50.34. This difference in evaluation timeframes could lead to differences in interpretation of the methodology. The time period USO used for non-DBA LBE analyses was generally 30 days, unless the source term reaches steady-state (i.e., all material that will be released has been released) in a shorter timeframe. For the step within the LMP methodology that involves the identification of high consequence BDBEs (those with doses exceeding 10 CFR 50.34 dose criteria), USO used the 30-day TEDE at the EAB that is aligned with the F-C target curve. For comparison of DBA analysis results, USO used the dose criteria and evaluation timeframes in 10 CFR 50.34. The staff determined that this aligns with what was intended for those steps in the LMP methodology.

3.3.1.9 Licensing Basis Event Selection Methodology

PSAR section 3.4 states that LBEs are selected according to the NEI 18-04 methodology without deviation. NEI 21-07 section 3.1 states that, in addition to affirming that NEI 18-04 is used and describing deviations, “the specific manner in which the methodology was applied should be described as necessary to provide an adequate description of the grouping of ES families that are used to define the AOOs, DBEs, and BDBEs.”

The applicant’s LBE selection methodology is provided in NAT-5599, “Natrium LBE Selection Guide,” Rev. 0, and the results provided in NAT-4767, “Licensing Basis Event (LBE) Selection Report,” Rev. 2, both of which were made available to staff during the audit. The staff verified that the initial LBE selection process is consistent with NEI 18-04, in that an initial set of LBEs was proposed based on a review of the plant design and the literature and iterated with the PRA to arrive at a final set of LBEs. Consistent with NEI 18-04, LBEs do not directly correspond to single event sequences or even event sequence families (ESFs), but groups of similar event sequences or ESFs that were considered bounded by the final LBE. The LBEs are then analyzed outside of the PRA to determine their consequences. Classification of LBEs and identification of DBAs corresponding to LBEs with frequency bands in the DBE region are discussed in section 3.4.1 of this SE, with two deviations from the NEI 18-04 methodology identified and discussed in the staff evaluation.

3.3.2 Conclusion

As discussed above, the staff reviewed the LBE analysis methodology described in PSAR section 3.3 and determined that the general approach to analyzing LBEs was consistent with NEI 18-04 and NEI 21-07, providing a comprehensive, consistent and integrated summary of the event sequences that form a large portion of the plant’s design bases. Therefore, the staff determines that the LBE analysis methodologies adequately support the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1) and (4) and 10 CFR 50.35, as applicable.

3.4 Licensing Basis Event Summary

PSAR section 3.5 summarizes the results of the selection, categorization and analysis of the LBEs, including DBAs, identified for inclusion into the facility’s design basis. Methods for evaluating LBEs are found in PSAR section 3.3. The LBE selection methodology is described in PSAR section 3.4. PSAR sections 3.6 through 3.8 describe the evaluations of AOOs DBEs, and BDBEs. PSAR section 3.9 describes the evaluations of the DBAs.

PSAR section 3.5.1 lists the tables used to summarize AOOs, DBEs, and BDBEs. It also describes conditions to define LBEs as risk significant. The dose requirements listed in 10 CFR 50.34 are used to classify a BDBE as high-consequence. PSAR section 3.5.2 describes a table that summarizes DBAs and the connection to DBEs.

The applicable regulatory requirements for the evaluation of the LBE summary are as follows:

- 10 CFR 50.34(a)(1) and (4), and
- 10 CFR 50.35.

The applicable guidance for the evaluation of the LBE summary are as follows:

- RG 1.233; and
- RG 1.253.

3.4.1 Technical Evaluation

The LBEs are derived from the event sequences developed in the site and design-specific PRA and should be categorized according to their frequency into one of the following three groups:

- AOOs
- DBEs
- BDBEs

DBAs are also considered LBEs; however, they are derived from the set of DBEs and AOOs and BDBEs with frequency bands extending into the DBE region by deterministically assuming that only safety-related SSCs are available to respond to the event. Section 3.8 of this SE evaluates DBA selection and analysis.

The staff reviewed PSAR section 3.4 and the overall LBE classifications documented in chapter 3 of the PSAR against the guidance in RG 1.233 and NEI 18-04, including the technical basis for the LBE selection and categorization methodology. Two deviations from the methodology were identified based on the staff's review.

The first deviation was in the lower range used in identifying BDBEs. RG 1.233 states, "The F-C target and related discussions in NEI 18-04 include an upper bound event sequence frequency (i.e., 95th percentile) of 5×10^{-7} /plant-year to define the lower range of BDBEs." While USO did use the 95th percentile frequency when determining which events to evaluate as BDBEs, mean frequency was used when determining which events would be classified and included in the CP application as BDBEs. Frequency and consequence information for the additional events that would be categorized as BDBEs if the 95th percentile were used are provided in SE table 3.4-1 below. USO confirmed these values and the intention to continue using mean frequency to determine classification of BDBEs in response to the request for confirmation (RCI)-1 (ML25259A180).

The second deviation from RG 1.233 was that USO did not generate DBAs for AOOs and BDBEs with frequency bands that extend into the DBE region when identifying and classifying LBEs. When performing the review, the staff considered the following statements within the guidance documents related to the generation of DBAs for these events:

- Section 3.2.2 of NEI 18-04: "The mean values of the frequencies are used to classify the LBEs into AOOs, DBEs, and BDBE categories. However, when the uncertainty bands defined by the 5th and 95th percentile of the frequency estimates straddles a frequency boundary, the LBE is evaluated in both LBE categories."
- Appendix A of NEI 21-07: "The classification of AOOs, DBEs, and BDBEs is based on the mean frequencies of the underlying uncertainty distributions. When the uncertainty band on the frequency defined by the 95th and 5th percentiles of the distribution straddle one of the frequency boundaries, the LBEs are evaluated on each side of the boundary,

per NEI 18-04. For example, if a BDBE has a 95th percentile estimate above 1×10^{-4} per plant year, it is treated as a DBE for the purposes of defining the [required safety functions] RSFs and defining the DBAs.”

The language from NEI 18-04 above is not explicit on which evaluations it refers to when it states “the LBE is evaluated.” USO’s original approach was that this evaluation only applied to Task 7a, “Evaluate LBEs Against the [frequency-consequence] F-C Target,” and the identification of RSFs¹³ by evaluating the results of DBEs relative to the F-C target as individual functions are removed. By limiting this evaluation Task 7a, no DBAs were created for AOOs with 5th percentile frequencies or BDBEs with 95th percentile frequencies that extend into the DBE region.

A primary objective of the deterministic DBA analysis is to verify that the set of SR controls is sufficient to ensure that the 10 CFR 50.34 dose criteria are not exceeded when only SR controls are available. Therefore, identifying and evaluating DBAs for these AOOs and BDBEs that extend into the DBE region is an important verification step and the statement in NEI 21-07 highlighted above confirms this approach. As described in the RCI response (ML25259A180), USO updated its process to identify DBAs for AOOs and BDBEs with uncertainty bands that extend into the DBE region for its OL application as stated in section 3.9 of the PSAR. Additional information on the impact of this deviation is documented in section 3.8.1 of this SE.

The applicant divided each set of AOOs, DBEs, BDBEs, and DBAs into the following types of transients:

- Loss of primary flow
- Increase or decrease in heat removal
- Reactivity
- Release from Ex-Vessel Systems
- Fuel Handling
- Local Fuel Faults
- Other

The LBEs in these categories are discussed in sections 3.5 through 3.8 of this SE. The staff reviewed PSAR section 3.5 against the guidance in RG 1.253 and NEI 21-07 for summarizing the LBE information described in the PSAR. The staff’s review considered the multiple tables categorizing the LBEs and a F-C chart, along with the response to RCI-3 (ML25259A180) which provides numerical values for the data represented in the F-C chart. The tables consisted of a description of the LBEs, risk significance, and dose consequences. The F-C chart compares the LBE results to the F-C target in NEI 18-04 figure 3-1.

¹³ PSAR section 5.2.1 notes that the term “safety-related function” is synonymous with the term “required safety function” as defined in NEI 18-04.

In addition to reviewing the LBEs based on their categorization (i.e., AOO, DBE, BDBE, and DBA), the staff reviewed the LBEs by IE family, which helps demonstrate the relationships between LBEs, and the credit attributed to each preventative and mitigative function within the safety analysis. Not unexpectedly at the CP stage, the staff found that not all event sequences had unique source terms and consequence modeling performed. In the fuel handling events in particular, the applicant chose analyses that had already been performed as representative or bounding of similar events in other release locations. The staff determined that this approach is acceptable at the CP stage when design information is limited, but additional analyses will be needed at the OL stage to appropriately capture differences in the MAR, mechanisms for release, and effectiveness of mitigative controls for the mechanistic source terms. See section 3.2 of this SE for additional information on the MST evaluation.

In reviewing the LBE analyses throughout chapter 3, the staff identified that IE families have the same root identifier, with the success case (i.e., all controls functioning) having the -BL extension and events representing failure of preventative or mitigative control having a numerical extension (e.g., -1, -2, -3) with higher numbers generally representing more failures. The staff found that some IE families have more than one set that represent different branches of sequential failures. Table 3.4-2 of this SE contains the IE families and the associated root identifiers, for the LBEs included in the PSAR safety analysis, grouped by the type of transient.

Table 3.4-2 of this SE shows that there are seven LBEs where only a single event from the IE family appears in the non-DBA LBEs within the PSAR. For these events, there is either a very low likelihood of the IE occurring, a very low failure frequency attributed to the PSFs involved, or a combination, which causes the event considering an additional failure to be below the frequency cutoff for consideration as an LBE. As discussed in section 3.1.3 of this SE, the use of low failure frequencies without sufficient bases was identified as a potential concern related to the maturity of the design and the PRA.

Two of these seven LBEs, RFH-OERC-BL and RFH-FDRC-1, are related as release events both occurring in the pin removal cell (PRC). Other events from these two IE families do not appear in the PSAR due to the low failure frequency attributed to the PRC. Based on information provided in response to RCI-1 and reflected in table 3.4-1 of this SE, the mean failure frequency of the PRC is 1.5×10^{-5} per plant year and it is attributed with reducing the 30-day TEDE by a factor of **[[]]**. This reliability and capability may not be achievable for a hot cell. This was the subject of an audit question and follow-on discussion between the staff and USO. Based on the staff's review of the information in the PSAR and the response to RCI-1 and the staff's confirmation of PSAR information via the audit, the staff concluded that the treatment of the PRC is potentially non-conservative but given the preliminary nature of the design and that the PRC is currently SR for radionuclide retention, the evaluation was sufficient for the CP stage.

For IE families with more than one LBE, the staff utilized the IE grouping to assess the preventative and mitigative effects of the PSFs and identify the expected SSC safety classification based on those effects. Figure 3.4-1 of this SE provides an example of this evaluation process for RFH-FDEM IE family. This family has two events involving fuel handling events occurring within the EVHM. In RFH-FDEM-1 the EVHM barrier performs its radionuclide retention function and in RFH-FDEM-2 the EVHM barrier function fails and is assumed to leak at 100 percent per day. By looking at these events together, the staff notes that the contribution of the EVHM barrier to mitigating RFH-FDEM-1 and preventing RFH-FDEM-2 can be evaluated relative to the F-C target. The staff notes that because the uncertainty band on RFH-FDEM-1 crosses into the DBE region, the EVHM barrier is SR based on both the mitigation for

RFH-FDEM-1 and the prevention for RFH-FDEM-2, a high-consequence BDBE. Additional discussion on these LBEs and PSFs are provided in section 5.4 of this SE.

The staff identified one event, the sodium leakage from the RV into the RV-GV annulus, that potentially needs to be updated to an LBE at the OL. The staff observed that this event did not screen into the LBE analyses provided for the CP; however, during audit discussions, the applicant acknowledged that the mean frequency of an RV leak should be in the BDBE range. The staff notes that this event therefore should have been identified as an LBE in the PSAR. In section 7.1.2.1.1 of this SE, the staff further discusses the GV design basis with respect to RV leakage and notes that RV leakage has not yet been fully justified as being outside the DBE region with consideration of uncertainty. PSAR table 5.2-4 identifies a PSF, DL4-RR1b, "Guard Vessel Leak Prevention Function," for DID adequacy that would be related to this event and would need to be reevaluated at the OL. The staff will review the identification and analysis of LBEs, including one associated with RV leakage into the RV-GV annulus, at the OL stage.

3.4.2 Conclusion

The staff reviewed the LBE summary information described in PSAR section 3.5 and determined that the identification and classification of LBEs was consistent with NEI 18-04 and RG 1.233, with the two exceptions noted above. The staff finds that the LBE information provided in the PSAR (1) is consistent with the guidance in RG 1.253 and NEI 21-07 and (2) provides a complete, consistent and integrated summary of the event sequences that form a large portion of the plant's design bases. Therefore, the staff finds that the LBE summary information adequately support the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1) and (4) and 10 CFR 50.35, as applicable.

Table 3.4-1: Additional BDBEs based on 95th Percentile Frequency of 5×10^{-7} /plant-year

LBE	5 th Percentile Frequency	Mean Frequency	95 th Percentile Frequency	5 th Percentile 30-day EAB TEDE [rem]	Mean 30-day EAB TEDE [rem]	95 th Percentile 30-day EAB TEDE [rem]
RFH-OERC-EX1	4.81×10^{-9}	3.30×10^{-7}	1.19×10^{-6}	1.20	3.31	5.05
IPI-IHEL-EX1	1.41×10^{-8}	1.57×10^{-7}	5.28×10^{-7}	no release		

Table 3.4-2: Initiating Event Families Reflected in the PSAR

Transient Type	Number of non-DBA LBEs	Initiating Event Family ID	Initiating Event Family Description
Loss of primary flow	5	DHP-L1PP	Loss of one primary sodium pump
	1	DHP-LAPP	Loss of all primary sodium pumps
	5	DHP-LOOP	Loss of offsite power
	2	SUD-LOOP	Loss of offsite power while at low power
	5	OTH-LMAC	Loss of a single medium voltage AC bus
Increase or decrease in heat removal	5	DHS-ISTL	Loss of heat sink
	3	DHS-RNBK	Energy island transient without reactor power runback
	3	IPI-IHEL	Intermediate heat exchanger secondary-to-primary leak
	2	SUD-IHEL	Intermediate heat exchanger secondary-to-primary leak while shutdown
	3	SUD-IACA	Loss of one train of IAC while shutdown
Reactivity	5	RPD-CW1ACS	Control rod-induced transient overpower
	5	RPD-SS	Reactor scram or spurious scram
Release from Ex-Vessel Systems	2	RRS-CGR	SCG leak
	2	SUD-CGR	SCG leak while shutdown
	1	RRS-ISPL	SPS leak at the cold trap
	1	RRS-SPLX	SPS-P leak in the RXB
	1	RRS-SPLA	SPS-P leak in the RAB

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Transient Type	Number of non-DBA LBEs	Initiating Event Family ID	Initiating Event Family Description
	2	RRS-RWG	RWG leak
Fuel Handling	5	RFH-FDIV	Fuel handling event occurs while moving fuel assembly in the reactor vessel
	2	RFH-FDSP	Fuel handling event occurs while moving fuel assembly in the SFP
	4	RFH-ESWR	Excessive sodium-water reaction in the pool immersion cell (PIC)
	2	RFH-LTCA	Loss of EVST active cooling while storing fuel assembly
	3	RFH-LMCA	Loss of EVHM active cooling while handling fuel assembly or an LTA or LDA
	1	RFH-LBCA	Loss of BLTC Active Cooling While Handling Fuel Assembly
	2	RFH-FDBL	Fuel handling event occurs while moving fuel assembly in the BLTC
	2	RFH-FDEM	Fuel handling event occurs while moving fuel assembly or an LTA or LDA in the EVHM
	2	RFH-FDET	Fuel handling event occurs while moving fuel assembly in the EVST
	2	RFH-FDPI	Fuel handling event occurs while moving fuel assembly in the PIC
	1	RFH-FDRC	Fuel handling event occurs while moving fuel assembly in the PRC
	1	RFH-OERC	Fuel damage while handling an LTA or LDA test pin in the PRC
Local Fuel Faults	3	LFF-SAO	Core blockage and local faults
Other	2	RFH-LSPC	Loss of spent fuel pool cooling

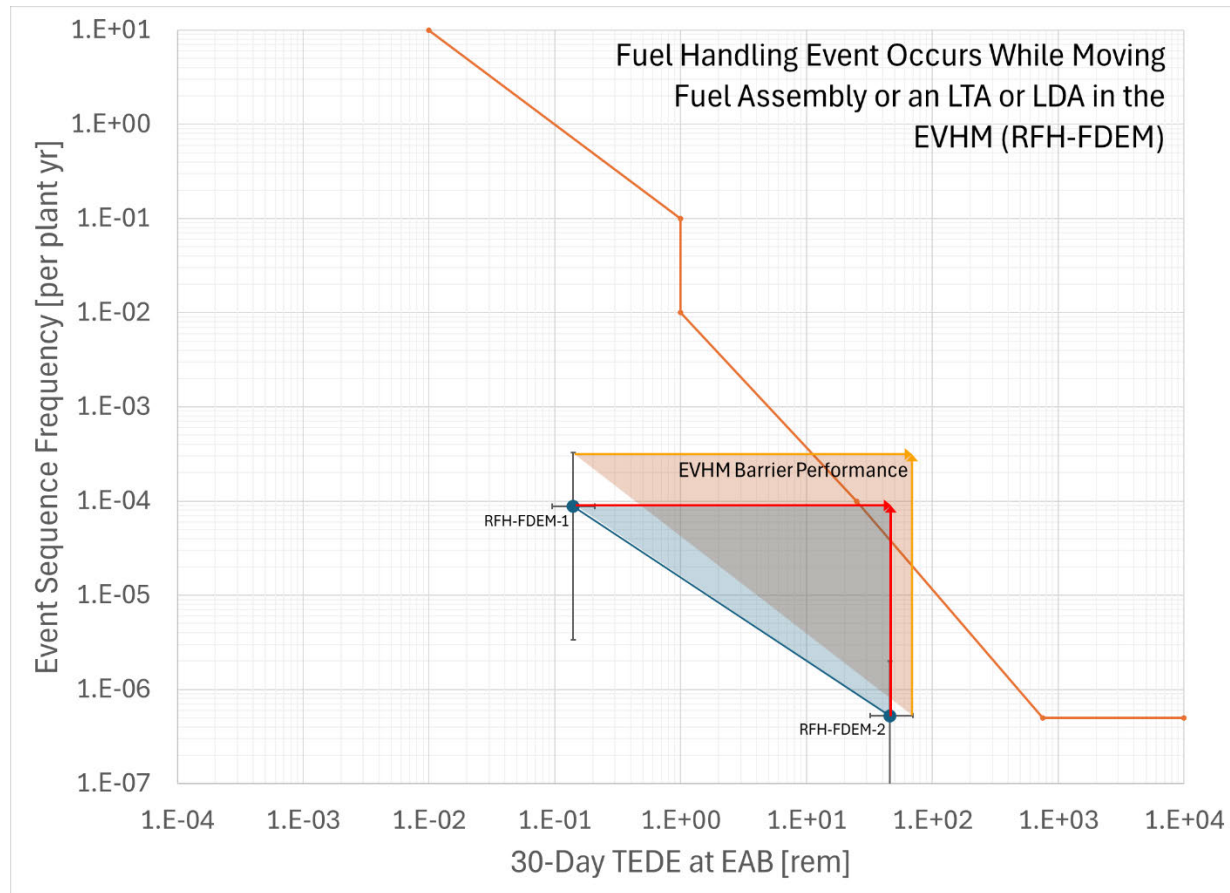


Figure 3.4-1: Example of Evaluation for PSF Contribution and SSC Safety Classification

3.5 Anticipated Operational Occurrences

PSAR section 3.6 describes the analysis of AOOs selected through use of methodologies described in PSAR sections 3.3 and 3.4 for inclusion in the plant's licensing basis.

The applicant divided the AOOs into several different categories: loss of primary flow, increase or decrease in heat removal, reactivity, and fuel handling AOOs. No ex-vessel system releases or other-type transients were identified as AOOs. Additionally, the frequency of local fuel faults was assigned at 1×10^{-3} per year as discussed in PSAR table 3.1-1, so these events are below the AOO frequency range.

The applicable regulatory requirements and guidance for the evaluation of AOOs are identical to those described in section 3.4 of this SE.

3.5.1 Technical Evaluation

The staff evaluated the PSAR section 3.6 AOO descriptions for their conformance with NEI 21-07, which states in section 3.3.1 that, for each AOO, the PSAR should contain a definition of the initial plant conditions and plant operating state, radionuclide source, IEs covered in the family, characterization of the responses of SSCs that perform PSFs, operator actions that perform PSFs, identification of whether or not there is a release, and definition of the safe, stable end state. AOOs with releases should also provide additional information describing the mechanistic source term, the mean, 5th, and 95th percentile frequency and dose, and significant factors that influence any degradation of layers of defense. For a CP, NEI 21-07 additionally notes that the assessments are preliminary, and the information provided should reflect the preliminary nature of the design information.

The staff assessed the AOOs to ensure they were appropriately categorized as AOOs based on their frequency. The staff also evaluated the credited functions to ensure they are consistent with the safety analysis assumptions, the descriptions of the SSCs in chapter 7 of the PSAR, and the safety classifications and tables in PSAR chapter 5. For AOOs with releases, the staff confirmed that the AOO remained within the F-C target curve. The staff's detailed evaluation of mechanistic source terms and functional containment performance is provided in section 3.2 of this SE.

The staff developed table 3.5-1, below, to summarize the PSAR evaluation of AOOs in terms of the MAR and the fundamental safety functions for each LBE.

3.5.1.1 Loss of Primary Flow AOOs

Loss of primary flow AOOs are described in PSAR section 3.6.1, which includes the following transients:

- Loss of one primary sodium pump with non-passive IAC (DHP-L1PP-BL)
- Loss of offsite power with non-passive IAC (DHP-LOOP-BL)
- Loss of offsite power while at low power with non-passive IAC (SUD-LOOP-BL)
- Loss of a single medium voltage AC bus with non-passive IAC (OTH-LMAC-BL)

The safety concern for these LBEs is an imbalance between heat generation in the reactor core and heat removal capability through the primary coolant. The MAR for these LBEs is the reactor core.

As described in PSAR section 3.6.1.1, the IE in DHP-L1PP-BL is a failure that causes the loss of a single primary sodium pump (PSP) while at full power. The staff confirmed in audit that the scenario represents a limiting reduction in flow (i.e., a locked rotor) for one PSP. The reduction in flow causes a scram based on high high power-to-flow ratio. This is consistent with what the staff would expect for loss of flow transients, because the flow would immediately begin to drop on the loss of forced flow. Decay heat removal is provided by IAC in non-passive cooling mode. As discussed in the footnote to SE table 3.5-1, non-passive IAC is an NST function, so no safety-significant PSF is associated with it in PSAR chapter 5, though the staff did verify through audit that a PSF was appropriately assigned. As shown in SE table 3.5-1, trips of the intermediate sodium pumps (ISPs) and remaining PSP are not credited, and a review of the analysis documentation provided in audit confirmed that the pump trips are appropriately not credited and pump heat continues to be added to the primary and intermediate systems. The PSAR also identifies that heat removal from non-passive IAC is sufficient to ensure fuel cladding integrity and there is no radionuclide release, which the staff confirmed through audit of the supporting calculations.

As described in PSAR sections 3.6.1.2 and 3.6.1.3, DHP-LOOP-BL and SUD-LOOP-BL are initiated by a LOOP while operating at full power and low power conditions, respectively. For DHP-LOOP-BL, the reduction in flow causes a reactor scram on high high power to flow ratio, and the staff considers this to be reasonable for the same reasons as discussed above for DHP-L1PP-BL. For SUD-LOOP-BL, the credited reactor scram is a loss of primary flow, and the staff considers this to be reasonable for low power conditions because the flow at the low power initial condition is higher than the power as a fraction of the full power condition, as discussed in PSAR section 3.3.4.2. In all cases the reactor shuts down successfully from the scram.

Decay heat removal for DHP-LOOP-BL and SUD-LOOP-BL is provided by non-passive IAC. The staff noted that non-passive IAC relies on electric power to the ISP and IAC blowers. On a LOOP, power is provided by the standby diesel generators, as discussed in PSAR section 1.1.4.3.14. The staff confirmed through audit that the analysis assumes an appropriate delay between the loss of offsite power and startup of the non-passive IAC, which will be further confirmed at the OL against the design and operational procedures. The LOOP causes the PSPs and ISPs to trip and coast down, and appropriately no pump trip is credited in the analysis because they occur due to the loss of offsite power. Even considering this delay, the PSAR identifies that heat removal from non-passive IAC is sufficient to ensure fuel cladding integrity and there is no radionuclide release, which the staff confirmed through audit of the supporting calculations.

As described in PSAR section 3.6.1.4, OTH-LMAC-BL is initiated by the loss of a single medium voltage alternating current (AC) bus, which disables one PSP and one ISP. The loss of flow in the primary system causes a scram on high high power-to-flow ratio, which the staff considers reasonable and consistent with the discussion above for the other LBEs in this section. Decay heat removal is provided by non-passive IAC. As discussed in PSAR section 7.2.2.3, IAC is designed such that a single train of non-passive IAC is sufficient to provide the required heat removal for the limiting decay heat load. Therefore, though only one train of IAC is operable due to the IE, the staff finds the use of non-passive IAC to be reasonable. A pump trip is not credited; the unaffected PSP avoids tripping by ramping down, which is an NST function performed automatically following a scram, as discussed briefly in PSAR section 7.1.3.1.2. The

PSAR identifies that heat removal from non-passive IAC is sufficient to ensure fuel cladding integrity and there is no radionuclide release. The staff confirmed in audit that OTH-LMAC-BL was not explicitly analyzed for the PSAR but instead was identified as bounded by a more conservative analysis. The staff determined this is reasonable for a preliminary analysis but notes that a full analysis or appropriate justification would be needed for final analysis at the OL.

As discussed in the response to RCI-3, these events have mean frequencies greater than 1×10^{-2} per year and are appropriately categorized as AOOs. However, DHP-LOOP-BL, SUD-LOOP-BL, and OTH-LMAC-BL have 5th percentile frequencies that are below 1×10^{-2} and should have DBAs identified per footnote 37 of Appendix A to NEI 21-07. DHP-LOOP-BL and SUD-LOOP-BL map to DBAs DHP-LOOP-CN and SUD-LOOP-CN, respectively, but OTH-LMAC-BL does not have an associated DBA at the CP stage. Through its review of the PSAR, the staff identified that OTH-LMAC-BL does not assume the failure of any SR functions. A related BDBE, OTH-LMAC-2, credits only SR SSCs and demonstrates that fuel integrity is maintained. Also, OTH-LMAC-2 is not identified in PSAR section 3.3.2.1 as one of the LBEs using the BEPU approach and uses a conservative evaluation consistent with the DBAs, as discussed in section 3.3.1.6 of this SE. The staff therefore determined that there is an adequate representation in the PSAR analyses of the needed SSC performance and potential consequences associated with a DBA developed from OTH-LMAC-BL. If OTH-LMAC-BL remains as an AOO with uncertainty bands that extend into the DBE range at the OL, a DBA should be appropriately assigned.

Based on the staff's review described above, the staff concluded that the information provided relative to the loss of primary flow AOOs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each AOO covering the initial condition, radionuclide source, IE, and characterization of the responses of SSCs performing PSFs. The staff also determined that the AOOs were appropriately categorized as AOOs, that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.5.1.2 Increase or Decrease in Heat Removal AOOs

The staff reviewed PSAR section 3.6.2, which describes increase or decrease in heat removal AOOs including the following transients:

- Loss of heat sink with non-passive IAC (DHS-ISTL-BL)
- Loss of one train of IAC while shutdown with non-passive IAC (SUD-IACA-BL)

The safety concern for these LBEs is an imbalance between heat generation in the core and heat removal to the ultimate heat sink.

As described in PSAR section 3.6.2.1, DHS-ISTL-BL is initiated by a loss of heat removal from one intermediate sodium loop at full power. The reactor scrams on high high cold pool temperature. This is consistent with staff expectations for a decrease in heat removal through the IHT, which would result in a smaller temperature drop on the primary side of the intermediate heat exchanger (IHx) and thus increasing cold pool temperature. Heat removal is provided by non-passive IAC, which, as discussed in section 3.5.1.1 of this SE is adequate even though only one train of IAC is operable due to the IE. The staff therefore finds the use of non-passive IAC to be reasonable. The PSAR identifies that heat removal from non-passive IAC is

sufficient to ensure fuel cladding integrity and there is no radionuclide release, which the staff confirmed through audit of the supporting calculations.

As described in PSAR section 3.6.2.2, SUD-IACA-BL occurs only during low power or shutdown operations, during which non-passive IAC is the normal means of heat removal as discussed in PSAR section 1.1.4.1. The transient is initiated by a loss of one train of IAC. A trip function is not credited, as shown in table 3.5-1 of this SE. The staff verified through audit that the limiting low power case analyzed demonstrates the ability of the plant to stabilize at a slightly higher temperature and slightly lower power without a reactor trip. Though one train of IAC is failed, the remaining train of non-passive IAC (and any passive heat removal provided by RAC) provides acceptable heat removal to maintain fuel within acceptable limits, consistent with the discussion above, and there is no radionuclide release, as confirmed by the staff through audit of the supporting calculations.

As discussed in the response to RCI-3, these events have mean frequencies greater than 1×10^{-2} per year and are appropriately categorized as AOOs. The 5th percentile frequencies are greater than 1×10^{-2} , so there is no need to develop associated DBAs.

Based on the staff's review described above, the staff concluded that the information provided relative to the increase or decrease in heat removal AOOs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each AOO covering the initial condition, radionuclide source, IE, and characterization of the responses of SSCs performing PSFs. The staff also determined that the AOOs were appropriately categorized as AOOs, that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.5.1.3 Reactivity AOOs

The staff reviewed PSAR section 3.6.3, which describes reactivity AOOs, including the following transients:

- Control rod-induced transient overpower with non-passive IAC (RPD-CW1ACS-BL)
- Reactor scram or spurious scram with non-passive IAC (RPD-SS-BL)

These LBEs relate to unexpected changes in core reactivity.

As described in PSAR section 3.6.3.1, in RPD-CW1ACS-BL the IE is a failure that causes a single control rod to continuously withdraw from the core while operating at full power. The increase in reactor power caused by the control rod withdrawal causes a reactor scram on either high high power range neutron flux or high high power to flow ratio. The staff verified through audit that the assumed reactivity insertion curve is conservative with respect to the available control rod worth and the capacity of the motor drive system, and the reactivity insertion continues until the reactor scram. Decay heat removal is provided by non-passive IAC, which is discussed in more detail in section 3.5.1.1 of this SE. The heat removal from non-passive IAC is sufficient to ensure fuel cladding integrity and there is no radionuclide release, which the staff confirmed through audit of the supporting calculations.

As described in PSAR section 3.6.3.2, in RPD-SS-BL the IE is a reactor scram. No additional function is credited to control heat generation; because the IE results in the reactor being in a

subcritical condition, the staff determined this is reasonable. Decay heat removal is provided by non-passive IAC, which is discussed in more detail in section 3.5.1.1 of this SE. The heat removal from non-passive IAC is sufficient to ensure fuel cladding integrity and there is no radionuclide release.

As discussed in the response to RCI-3, these events have mean frequencies greater than 1×10^{-2} per year and are appropriately categorized as AOOs. The 5th percentile frequency for RPD-CW1ACS-BL is below 1×10^{-2} , so there should be an associated DBA; however, one is not assigned. The staff notes that a related BDBE, RPD-CW1ACS-2, credits only SR functions, as confirmed through audit of associated calculational notebooks, with no fuel failure identified. Also, RPD-CW1ACS-2 is not identified in PSAR section 3.3.2.1 as one of the LBEs using the BEPU approach and uses a conservative evaluation consistent with the DBAs, as discussed in section 3.3.1.6 of this SE. The staff therefore determined that there is an adequate representation in the PSAR analyses of the needed SSC performance and potential consequences associated with a DBA developed from RPD-CW1ACS-BL. If RPD-CW1ACS-BL remains as an AOO with uncertainty bands that extend into the DBE range at the OL, a DBA should be appropriately assigned.

Based on the staff's review described above, the staff concluded that the information provided relative to the reactivity AOOs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each AOO covering the initial condition, radionuclide source, IE, and characterization of the responses of SSCs performing PSFs. The staff also determined that the AOOs were appropriately categorized as AOOs, that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.5.1.4 Fuel Handling AOOs

The staff reviewed PSAR section 3.6.4 on fuel handling AOOs, which include the following transients:

- Fuel Damage While Handling an LTA or LDA Test Pin with PRC Barrier Successfully Retains Release (RFH-OERC-BL)
- Loss of Spent Fuel Pool Cooling with Cooling Restored (RFH-LSPC-BL)
- Loss of EVST Active Cooling While Storing Fuel Assemblies with Passive Cooling (RFH-LTCA-BL)

These LBEs relate to issues with fuel handling that have the potential to cause fuel damage, potentially to multiple fuel assemblies. Because these events do not involve the reactor core, reactivity control SSCs are not credited in the LBEs.

As described in PSAR section 3.6.4.1, RFH-OERC-BL represents a scenario where a fuel handling event occurs while removing a pin from a high burnup LTA or LDA in the PRC. PSAR section 3.6.4.1.2 states that the event results in damage to the pin being removed as well as the surrounding six pins, for a total of seven damaged fuel pins. The staff considers this to be a reasonable assumption based on the geometry of the fuel assembly and the mechanism for removing pins in the LTA or LDA as described in NAT-2806-A, "Fuel and Control Assembly

Qualification,” Rev. 0 (ML24354A192). The applicant does not discuss any credit for decay heat removal SSCs for the LBE, though the staff notes that decay heat removal is provided by a liquid sodium holding tank built into the floor of the PRC hot cell as described in PSAR section 7.3.2.1.7. The radionuclide retention function credited is the SR PRC cell barrier. Additional discussion on the PRC cell barrier can be found in section 5.4.3 of this SE. The source term and consequence analysis assumptions are consistent with the discussion in section 3.2.1.3 of this SE, which notes that the source term is taken from an EVHM transient but scaled to reflect fewer fuel pins assumed failed.

As described in PSAR section 3.6.4.2, RFH-LSPC-BL represents a loss of SFP cooling, but cooling is restored before the SFP boils and there is no fuel failure. The only function credited is the SR function to provide passive heat removal in the water pool fuel handling system (FHP), and as noted in PSAR section 7.3.1.2 the SFP can cool passively for more than 7 days before operator action is required. The staff finds it reasonable to expect that SFP cooling would be reestablished (either through makeup or restoration of the cooling system) after the typical LBE time of 72 hours but before 7 days, using non-safety SSCs or operator actions.

As described in PSAR section 3.6.4.3, RFH-LTCA-BL represents a loss of EVST active cooling. The EVST active cooling system is NST and has minimal description in the PSAR. The staff understands from the audit that this system is going through redesign, but since it is assumed to be failed in the LBE analysis it does not play a role in the event progression. For decay heat removal, the LBE credits heat removal using the EVST passive heat removal system. This system is described in PSAR section 7.3.2, with specific performance characteristics for the system provided in section 7.3.2.3. The heat removal from the EVST passive heat removal system is sufficient to ensure fuel cladding integrity and there is no radionuclide release, which the staff confirmed through audit of the supporting calculations.

As discussed in the response to RCI-3, these events have mean frequencies greater than 1×10^{-2} per year and are appropriately categorized as AOOs. RFH-OERC-BL and RFH-LSPC-BL have 5th percentile frequencies below 1×10^{-2} , but DBAs are not assigned.

For RFH-OERC-BL, the PRC cell barrier is SR as noted earlier. As discussed in section 3.4.1 of this SE, RFH-OERC-EX1 is another event within this IE family with a mean below 5×10^{-7} /plant-year, but a 95th percentile frequency above 5×10^{-7} /plant-year that would be classified as a BDBE according to the LMP methodology. Based on the frequency and consequence information from these two events and the LMP methodology, the PRC cell barrier is SR based on evaluation of the mean values relative to the F-C target curve for this AOO (mitigation of event with frequency band extending into the DBE region) and based on its need to mitigate the DBA event associated with the AOO crossing into the DBE region. As discussed in section 3.4.1 of this SE, both the prevention and mitigation credit provided to the PRC cell barrier are significant and may be non-conservative, but the design is still very preliminary. These values will continue to be evaluated by USO as the design develops and be reviewed in detail by the staff at the OL stage.

For RFH-LSPC-BL, the only function credited is the SR function to provide passive heat removal in the FHP, and as noted in PSAR section 7.3.1.2 the SFP can cool passively for more than 7 days before operator action is required. The staff finds it reasonable to expect that SFP cooling would be reestablished (either through makeup or restoration of the cooling system) after the typical LBE time of 72 hours but before 7 days, using non-safety SSCs like the NSRST FHP emergency makeup function (DL4-HR4) or operator actions. The staff therefore determined that RFH-LSPC-BL demonstrates the performance of the necessary SR SSCs, and is consistent

with what would be expected for a DBA analysis of this kind of event, so there is no impact of the missed DBA on the plant design and SSC safety classification for the CP. The staff expects to review an appropriate DBA analysis derived from RFH-LSPC-BL at the OL stage.

3.5.2 Conclusion

The staff reviewed PSAR section 3.6 against the guidance in RG 1.233 and NEI 18-04 regarding analysis of AOOs and against the content of application guidance provided in RG 1.253 and NEI 21-07. The staff determined that the information provided in the PSAR is consistent with the guidance documents and the information provided on AOO analysis is acceptable on the basis that the AOO analyses, which were conducted consistent with the methods described in section 3.3 of this SE, demonstrate that the specified acceptable system radionuclide release design limits (SARRDLs) are maintained by maintaining fuel integrity, except for one AOO that assumes fuel failure for which the offsite dose consequences remain within the F-C target curve. The AOOs credit appropriate SSCs to perform PSFs, consistent with the event descriptions. The analysis is preliminary, and additional work is needed to finalize the design and analysis prior to the OL, as described above.

The staff determined that the AOO analysis information provided in the PSAR gives a sufficient description of the AOOs that is consistent with design information in other sections of the PSAR and is consistent with the guidance in RG 1.253 and NEI 21-07. The methodology used to identify and evaluate LBEs in the AOO region was consistent with RG 1.233 and NEI 18-04, with the exception of the identification of DBAs for AOOs with 5th percentile frequencies in the DBE region. As discussed above, the DBAs that were not identified and evaluated within the PSAR were evaluated by the staff and are not expected to result in additional SR SSCs. These additional DBAs should be provided in the FSAR at the OL stage for review. Therefore, the staff finds that the AOO analysis information adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1) and (4) and 10 CFR 50.35, as applicable.

3.5.3 Tables

Table 3.5-1: Summary of anticipated operational occurrences from PSAR section 3.6

PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Heat Generation Control Functions	PSFs of Heat Generation Control Functions	Decay Heat Removal Functions	PSFs of Decay Heat Removal Functions	Radionuclide Retention Functions	PSFs of Radionuclide Retention Functions
3.6.1.1	Loss of One Primary Sodium Pump with Non-Passive IAC	DHP-L1PP-BL	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2
3.6.1.2	Loss of Offsite Power with Non-Passive IAC	DHP-LOOP-BL	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2
3.6.1.3	Loss of Offsite Power While at Low Power with Non-Passive IAC	SUD-LOOP-BL	Loss of Primary Flow	Reactor core	Reactor scram on loss of primary flow	DL3-RC1	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2
3.6.1.4	Loss of a Single Medium Voltage AC Bus with Non-Passive IAC	OTH-LMAC-BL	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2
3.6.2.1	Loss of Heat Sink with Non-Passive IAC	DHS-ISTL-BL	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high cold pool temperature	DL3-RC1	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2
3.6.2.2	Loss of One Train of IAC While Shutdown with Non-Passive IAC	SUD-IACA-BL	Increase or Decrease in Heat Removal	Reactor core	None	None	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2

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PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Heat Generation Control Functions	PSFs of Heat Generation Control Functions	Decay Heat Removal Functions	PSFs of Decay Heat Removal Functions	Radionuclide Retention Functions	PSFs of Radionuclide Retention Functions
3.6.3.1	Control Rod-Induced Transient Overpower with Non-Passive IAC	RPD-CW1ACS-BL	Reactivity	Reactor core	Reactor scram on high high power range neutron flux or high high power-to-flow ratio	DL3-RC1	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2
3.6.3.2	Reactor Scram or Spurious Scram with Non-Passive IAC	RPD-SS-BL	Reactivity	Reactor core	Unspecified reactor scram	DL3-RC1	IAC non-passive mode	None ¹	Fuel cladding	DL3-RR2
3.6.4.1	Fuel Damage While Handling an LTA or LDA Test Pin with PRC Barrier Successfully Retains Release	RFH-OERC-BL	Fuel Handling	High burnup LTA or LDA, 2.5 day decay	N/A	N/A	N/A	N/A	PRC cell barrier	DL3-RR6
3.6.4.2	Loss of Spent Fuel Pool Cooling with Cooling Restored	RFH-LSPC-BL	Fuel Handling	SFP contents	N/A	N/A	Passive heat removal in FHP	DL3-HR10	Fuel cladding	DL3-RC2
3.6.4.3	Loss of EVST Active Cooling While Storing Fuel Assemblies with Passive Cooling	RFH-LTCA-BL	Fuel Handling	EVST contents	N/A	N/A	Passive heat removal in EVST	DL3-HR7	Fuel cladding	DL3-RR2

¹No safety-significant PSF is listed in the tables in PSAR section 5.2 for the non-passive mode of IAC, because it is an NST function.

3.6 Design Basis Events

PSAR section 3.7 describes the analysis of DBE selected through use of methodologies described in PSAR sections 3.3 and 3.4 for inclusion in the plant's licensing basis.

The applicable regulatory requirements for the evaluation of the DBE analysis are as follows:

- 10 CFR 50.34(a)(1), (a)(3)(ii), and (4), and
- 10 CFR 50.35.

The applicable guidance for the selection of DBEs is as follows:

- RG 1.233, and
- RG 1.253.

3.6.1 Technical Evaluation

The staff evaluated the PSAR section 3.7 DBE descriptions for their conformance with NEI 21-07, which states in section 3.4.1 that, for each DBE, the PSAR should contain a definition of the initial plant conditions and plant operating state, radionuclide source, IEs covered in the family, characterization of the responses of SSCs that perform PSFs, operator actions that perform PSFs, identification of whether or not there is a release, and definition of the safe, stable end state. DBEs with releases should also provide additional information describing the mechanistic source term, the mean, 5th, and 95th percentile frequency and dose, and significant factors that influence any degradation of layers of defense. The PSAR should also identify the most limiting DBEs associated with DBAs. For a CP, NEI 21-07 additionally notes that the assessments are preliminary and the information provided should reflect the preliminary nature of the design information.

The staff assessed the DBEs to ensure they were appropriately categorized as DBEs based on their frequency (mean frequency between 1×10^{-4} and 1×10^{-2} per plant-year). The staff also evaluated the credited functions to ensure they are consistent with their descriptions in chapter 7 of the PSAR and the safety classifications and tables in PSAR chapter 5. For DBEs with releases, the staff determined that the DBEs remained within the F-C target curve and assessed the credited functional containment barriers and their assumed performance to ensure they appropriately reflected the plant design, operating state, available isolation functions, filters, etc. The staff's detailed evaluation of mechanistic source terms is provided in section 3.2 of this SE.

The staff developed table 3.5-1, below, to summarize the PSAR evaluation of DBEs in terms of the MAR and the fundamental safety functions for each LBE.

3.6.1.1 Loss of Primary Flow DBEs

The staff reviewed PSAR section 3.7.1 on the loss of primary flow DBEs including:

- Loss of one primary sodium pump with RAC (DHP-L1PP-2)

- Loss of offsite power with passive IAC (DHP-LOOP-1)
- Loss of offsite power while at low power with passive IAC (SUD-LOOP-1)

The safety concern for these LBEs is an imbalance between heat generation in the reactor core and heat removal capability through the primary coolant. In all PSAR section 3.7.1 transients the IE is a failure that causes a reduction in primary coolant flow.

As described in PSAR section 3.7.1.1, DHP-L1PP-2 is initiated by a failure that causes the loss of a single PSP while at full power. The staff confirmed in audit that the scenario represents a limiting reduction in flow (i.e., a locked rotor) for one PSP. The reduction in primary flow results in a reactor scram on high high power-to-flow ratio, which is consistent with what the staff would expect for loss of flow transients as discussed in section 3.5.1.1 of this SE. Based on the assumed failure of both trains of active and passive IAC, decay heat removal is provided by the SR RAC. As discussed in PSAR section 7.2.1.3, RAC relies on natural circulation of the primary coolant in the reactor vessel. This allows the reactor vessel temperature to increase, which improves the effectiveness of RAC. In addition to the RAC heat removal function itself, the staff identified that primary coolant natural circulation is also appropriately credited. The LBE also credits primary and intermediate sodium pump trips on high high primary sodium temperature, which are intended to reduce the pump heat added to the coolant as discussed in PSAR sections 7.1.3 and 7.1.4, respectively. Heat removal from RAC is sufficient to ensure fuel cladding integrity and there is no radionuclide release, which the staff confirmed through audit of the supporting calculations.

As described in PSAR sections 3.7.1.2 and 3.7.1.3, DHP-LOOP-1 and SUD-LOOP-1 are initiated by a LOOP while operating at full power and low power conditions, respectively. For DHP-LOOP-1, the reduction in flow causes a reactor scram high high power to flow ratio, and the staff considers this to be reasonable for the same reasons as discussed above for DHP-L1PP-2. For SUD-LOOP-1, the credited reactor scram is a loss of primary flow. The staff finds this to be reasonable for low power conditions because the flow at the low power initial condition is higher than the power as a fraction of the full power condition, as discussed in PSAR section 3.3.4.2. In both cases the reactor shuts down successfully from the scram. Based on the loss of offsite power, active IAC is assumed to fail on both trains so decay heat removal for DHP-LOOP-BL and SUD-LOOP-BL is provided by passive IAC. The LOOP causes the PSPs and ISPs to trip and coast down, and appropriately no pump trip is credited in the analysis because they occur due to the loss of offsite power. The heat removal from passive IAC is sufficient to ensure fuel cladding integrity and there is no radionuclide release, which the staff confirmed through audit of the supporting calculations.

As discussed in the response to RCI-3, these events have mean frequencies between 1×10^{-2} and 1×10^{-4} per plant-year and are appropriately categorized as DBEs. DHP-L1PP-2, DHP-LOOP-1, and SUD-LOOP-1 are the most limiting DBEs that serve as the basis for DBAs DHP-L1PP-CN, DHP-LOOP-CN, and SUD-LOOP-CN, which are discussed in section 3.8.1.1 of this SE.

Based on the staff's review described above, the staff concluded that the information provided relative to the loss of primary flow DBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each DBE covering the initial condition, radionuclide source, IE, and characterization of the responses of SSCs performing PSFs. The staff also determined that the DBEs were appropriately categorized as DBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis

assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7. The staff confirmed that DBEs identified as limiting were appropriately mapped to DBAs described in PSAR section 3.9.

3.6.1.2 Increase or Decrease in Heat Removal DBEs

The staff reviewed PSAR section 3.7.2 on the increase or decrease in heat removal DBEs including:

- Loss of heat sink with passive IAC (DHS-ISTL-1)
- Loss of heat sink with RAC (DHS-ISTL-2)
- Energy island transient without reactor power runback, with non-passive IAC (DHS-RNBK-1)
- Energy island transient without reactor power runback, with RAC (DHS-RNBK-3)
- Loss of one train of IAC while shutdown with passive IAC (SUD-IACA-1)
- Intermediate heat exchanger secondary-to-primary leak with non-passive IAC (IPI-IHEL-BL)
- Intermediate heat exchanger secondary-to-primary leak while shutdown with non-passive IAC (SUD-IHEL-BL)

The safety concern for these LBEs is an imbalance between heat generation in the core and heat removal to the ultimate heat sink.

As described in PSAR sections 3.7.2.1 and 3.7.2.2, for DHS-ISTL-1 and 2, the IE is a loss of heat removal from an intermediate sodium loop at full power. The staff confirmed in the audit that both events are modeled as a limiting loss of heat removal to one of the two sodium loops. This causes cold pool temperatures to increase, which results in a reactor scram on high high cold pool temperature; the staff finds this to be reasonable and consistent with the discussion in section 3.5.1.2 of this SE for decrease in heat removal AOOs. DHS-ISTL-1 assumes non-passive IAC fails on both trains and credits a single train of passive IAC for decay heat removal. Though the LBE appropriately credits PSP and ISP trips to limit heat addition to the coolant, consistent with the discussion provided in SE section 3.6.1.1, a single loop of passive IAC is not sufficient to remove the limiting decay heat load as discussed in section 7.2.2.3 of the PSAR. Though it is not explicitly credited in the event description or PSAR table 5.2-2, RAC is available and would be expected to participate in decay heat removal for these LBEs. The staff confirmed by auditing the supporting calculations that RAC heat transfer plays an important role in heat removal for DHS-ISTL-1, and the combined single train of passive IAC, augmented by the heat transfer provided by RAC, is sufficient to ensure fuel integrity. DHS-ISTL-2 assumes both passive and non-passive IAC fail and fully credits RAC, with appropriate credit for PSP and ISP trips to limit heat addition to the coolant, consistent with the discussion provided in SE section 3.6.1.1. For DHS-ISTL-2, RAC is sufficient to ensure fuel integrity, as confirmed by the staff audit of the supporting analyses.

As described in PSAR sections 3.7.2.3 and 3.7.2.4, for DHS-RNBK-1 and DHS-RNBK-3, the IE is a runback demand initiated by an increase or decrease in heat removal from the nuclear island salt system (NSS) during full power operation. Under normal conditions, the demand would initiate a power runback which, as described in PSAR section 1.3.2.2 and 3.3.4.1, results in insertion of control rods and reduction in PSP and ISP speed. In these LBEs, which are intended to bound events that result in a runback demand, the runback is not completed prior to an assumed reactor scram, which shuts down the reactor. The scram signal is not specified in the PSAR, which the staff finds to be reasonable because the scram is assumed to occur as part of the analysis. Decay heat removal is provided by non-passive IAC for DHS-RNBK-1, while RAC provides decay heat removal for DHS-RNBK-3, with appropriate credit for SR PSP and ISP trips, PSP coastdown, and primary system natural circulation. For both transients, fuel cladding integrity is maintained and there is no radionuclide release. The staff audited the LBE analyses performed in support of the KU1 CP application and observed some inconsistencies compared to the event description provided in the PSAR. However, the staff identified that the analyses that were performed are sufficiently conservative to support the PSAR's conclusions regarding the outcome of the event and the adequacy of the credited SSCs. The staff determined this was adequate for preliminary analyses and notes the OL application is anticipated to provide final analyses that correspond to the event description or appropriate justification that a different analysis is bounding.

As described in PSAR section 3.7.2.5, the IE for SUD-IACA-1 is a loss of a single train of IAC during low power or shutdown operating conditions, which, as discussed in SE section 3.5.1.2, rely on IAC for normal decay heat removal. As with SUD-IACA-BL, a reactor trip is not credited. The LBE credits the remaining train of IAC in passive mode, which provides sufficient heat removal, and fuel cladding integrity is maintained. The staff audited the supporting analysis and identified some inconsistencies compared to the event description in the PSAR, in that both trains of IAC continue to be available in the passive mode. However, given that this event is bounded by SUD-IACA-CN, which uses conservative assumptions and demonstrates acceptable performance with decay heat removal provided solely by RAC, the staff determined the preliminary analysis was appropriate for the CP stage. If SUD-IACA-1 remains a DBE in the OL application, it is anticipated that the event description or analysis would be revised for consistency.

As described in PSAR sections 3.7.2.6 and 3.7.2.7, the IE for IPI-IHEL-BL and SUD-IHEL-BL is a leak in the IHX that causes intermediate sodium to flow into the primary system. The immediate safety concern is that the quantity of sodium that leaks into the primary system would be enough to overfill the reactor vessel and pose a challenge to the primary coolant boundary. PSAR section 3.7.2.7.1 describes that the vessel overfill is prevented by the IHT physical arrangement as discussed in PSAR section 7.1.4. PSAR section 7.1.4 states that the physical arrangement that limits the overfill is based on a high point downstream of the primary sodium pump with a siphon break provided by the expansion tank; in the event of drain down of the IHT loop, the high point fills with inert gas and prevents the entire IHT inventory from being siphoned into the reactor. The staff identified that the anti-siphon feature is reliant on a trip of the ISP on the faulted IHT train, which is provided as a SR ISP trip on high high primary sodium level (DL3-HR12). IPI-IHEL-BL and SUD-IHEL-BL appropriately credit this trip to prevent the overfill as discussed in PSAR section 3.7.2.7.1 and table 5.2-2, and indicated in SE table 3.6-1, below.

However, because IPI-IHEL-BL and SUD-IHEL-BL result in the loss of heat removal from a single train of IHT even when the leakage is terminated, both LBEs must be evaluated to ensure there is adequate decay heat removal. Both events credit non-passive IAC for decay heat removal. Though non-passive IAC only provides heat removal in the intact IHT loop, it is

sufficient to remove decay heat as discussed in section 3.5.3.1 of the PSAR, and fuel cladding integrity is maintained.

As discussed in the response to RCI-3, these events have mean frequencies between 1×10^{-2} and 1×10^{-4} per plant-year and are appropriately categorized as DBEs. IPI-IHEL-BL is identified as the most limiting DBE that serves as the basis for DBA IPI-IHEL-CN, and SUD-IHEL-BL is identified as the most limiting DBE that serves as the basis for DBA SUD-IHEL-CN, which are discussed in section 3.8.1.2 of this SE.

Based on the staff's review described above, the staff concluded that the information provided relative to the increase or decrease in heat removal DBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each DBE covering the initial condition, radionuclide source, IE, and characterization of the responses of SSCs performing PSFs functions. The staff also determined that the DBEs were appropriately categorized as DBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7. The staff determined that DBEs identified as limiting were appropriately mapped to DBAs described in PSAR section 3.9.

3.6.1.3 Release from Ex-Vessel Systems DBEs

The staff reviewed PSAR section 3.7.3 on the DBEs that involve release from ex-vessel systems, including:

- SCG leak in the SCG primary coolant boundary where RXB substructure successfully retains release (RRS-CGR-BL)
- SCG leak downstream of the SCG vapor trap cell where RXB substructure successfully retains release (RRS-CGR-1)
- SPS leak at the cold trap with RAB barrier successfully retains release (RRS-ISPL-BL)
- SPS-P leak in the RXB (RRS-SPLX-BL)
- SPS-P leak in the RAB (RRS-SPLA-BL)
- RWG leak from the holdup tank with FHB barrier fails to retain release (RRS-RWG-1)
- RWG leak to the stack to the environment (RRS-RWG-2)

The PSAR describes that LBEs are postulated for systems that contain radionuclides, either from circulating radionuclides from failed fuel during operation or activated primary sodium, which have the potential to be released to the environment. In the short term these LBEs do not involve the potential for releases from the core, and no reactor scram functions are credited. The staff considers this to be reasonable for the PSAR, because the LBEs evaluate the immediate risk of release resulting from the leak. Longer-term issues (for example, loss of inerting for the reactor core following an SCG leak) will be evaluated at the OL for any manual actions (e.g., reactor shutdown, manual isolation) needed to bring the plant to safe condition following an ex-vessel release

As described in PSAR sections 3.7.3.1 and 3.7.3.2, RRS-CGR-BL and RRS-CGR-1 involve leaks from the sodium cover gas system (SCG), the difference being where the leak takes place. In RRS-CGR-BL, the leak is from the SCG primary coolant boundary into the HAA, while in RRS-CGR-1 the leak is into the vapor trap cell (VTC). The staff's evaluation of the source term and functional containment performance is described in section 3.2.1.4 of this SE. As shown in table 3.6-1, the credited functional containment functions are the ex-RES functional containment barrier, the HAA barrier, and HAA HVAC operations following release, all of which are NSRST functions. These are consistent with the source term discussion in section 3.2.1.4 of this SE and as such the staff determined that they are reasonable. PSAR section 3.7.3.2 notes that the vapor trap cell has a lower leakage rate than the HAA, so the consequence results from RRS-CGR-BL are applied to RRS-CGR-1. Though the VTC leakage rate is not specified in the PSAR, the staff finds the application of the RRS-CGR-BL results to RRS-CGR-1 to be acceptable, because, as stated in PSAR section 7.2.3.2.1, the VTC is sealed, inerted, and maintained at negative pressure relative to its surroundings and would be expected to have lower leakage than the HAA.

PSAR section 3.7.3.1.2 describes that the HAA HVAC operation is modeled differently in RRS-CGR-BL (and by application of the same source term, RRS-CGR-1) from other events in the PSAR. Other events involving the HAA assume a design leak rate of 10 percent per day and a degraded leak rate of 100 percent per day throughout the event. For RRS-CGR-BL and -1, the HAA HVAC operates at 400 cubic feet per minute (cfm) for 30 minutes with filtration, prior to isolation, after which the design leak rate is applied. The staff notes that the HVAC flow rate and filtration prior to the isolation reflects normal initial conditions, but no justification was provided in the PSAR for the timing of the isolation function. The staff also noted that RRS-CGR-CN, the associated DBA, has a 30-day LPZ dose of 7.77 rem TEDE, with conservative assumptions in the consequence analysis and without credit for HVAC filtration or isolation. The staff will review further justification for the isolation timing at the OL stage.

PSAR section 3.7.3.3 describes that RRS-ISPL-BL relates to a leak from the SPS-I that releases precipitated tritium from the intermediate cold trap. As discussed in PSAR section 7.2.4.3, tritium is produced in the core primarily by ternary fission in the fuel and by a threshold reaction with boron in the control rods. Driven by concentration gradients, tritium will migrate through the barriers from the primary sodium to sodium in the intermediate loop and then to a lesser extent from the intermediate loop to the salt loop. Most tritium in the primary and intermediate loops is held in cold traps, minimizing the concentration of circulating tritium. RRS-ISPL-BL postulates the release of one of two cold traps in the intermediate loop as discussed in section 3.2.1.4 of this SE. The staff audited information on expected quantities of tritium throughout the systems, performed independent calculations, and determined that even with bounding assumptions and conservative analysis methods, consequences from an unmitigated release of all tritium in the primary or intermediate system would not exceed dose targets at the EAB. The staff will review additional information on the inventory of the SPS-P cesium trap and the potential for combined releases involving the cold trap at the OL stage.

As described in PSAR sections 3.7.3.4 and 3.7.3.5, RRS-SPLX-BL and RRS-SPLA-BL involve leaks in the SPS-P, with RRS-SPLX-BL leaking into the RXB and RRS-SPLA-BL leaking into the SPS cell in the RAB. For these events, if the sodium processing system pump is not tripped, there is a risk of excessively draining the primary system. However, both events credit a SR SPS-P pump trip, and the SPS-P line is designed with a siphon break to limit the available volume for leakage.

Aside from the potential to drain the primary coolant, the staff's main concern in terms of radionuclide release for these events is activated sodium, primarily in the form of sodium-24. For RRS-SPLA-BL, 20 gallons of aerosol is assumed to be leaked into the SPS cell. This is leaked from the SPS cell at 10 percent volume per day to the RAB, which exhausts to the environment every two hours. While the basis for the 20 gallons of aerosol was not identified, the staff notes that it is likely conservative based on the amount of sodium that would need to leak to produce such a quantity of sodium aerosol based on staff calculations of leakage into an inert cell. The staff also noted that the SPS cell has associated ventilation and inerting systems; the 10 percent leakage rate assumed from the SPS cell appears to assume these are isolated. Isolation for these systems is included as a subfunction of DL4-RR3, the SPS cell barrier function, as shown in PSAR table 5.2-4 where it is noted as NSRST for defense in depth purposes. This function is not explicitly credited for this LBE. The staff understands from audit that this is largely a function of the design and PRA maturity at the time the analysis was conducted and will review the OL-stage analysis to ensure the appropriate functions are analyzed and credited.

As discussed in section 5.5.3 of this SE, both the nominal and degraded leakage rates for the SPS cell have significant impacts on the final consequence results for these events due to the short half-lives of the dominant radionuclides. The staff expects to review additional information on the bases for these leak rates at the OL stage. In addition, the availability of inerting systems to achieve the initial conditions for these safety analysis evaluations and prevent additional vaporization or failure of the SPS cell by sodium fires will need to be addressed either through inclusion in the TSs or explicit modeling of the inerting function in the PRA.

The PSAR describes that RRS-SPLX-BL assumes the same consequences as RRS-SPLA-BL, with the basis that the HAA leakage rate is similar to the SPS cell leakage rate, but the volume in the SPS cell is smaller, which would lead to larger releases. The staff determined this is reasonable from a radionuclide release consequence perspective, but the staff notes that leakage of sodium from SPS-P into the HAA should be considered to account for possible sodium fires. As noted in PSAR table 3.1-3, the fire hazard analysis will be addressed at the OL, consistent with RG 1.253. However, the staff notes that there is an NSRST enclosure around the SPS-P piping in the HAA as mentioned in PSAR sections 7.2.4 and 8.2, which limits interaction with air. As described in PSAR sections 3.7.3.6 and 3.7.3.7, RRS-RWG-1 and RRS-RWG-2 both involve leaks from the RWG holdup tank, which is discussed in more detail in PSAR section 7.4.1.3.1. RRS-RWG-1 involves a leak from the holdup tank into the tank vault, which provides some confinement with filtered exhaust provided by NHV. The NHV filtration is credited but, as shown in table 3.6-1 of this SE, does not have a safety-significant PSF assigned because it is NST. RRS-RWG-2 involves a leak of the holdup tank directly out to the vent stack over a 10 minute period with filtration in the RWG system downstream of the holdup tank. The staff's evaluation of the source term and functional containment performance is described in section 3.2.1.4 of this SE.

As discussed in the response to RCI-3, these events have mean frequencies between 1×10^{-2} and 1×10^{-4} per year and are appropriately categorized as DBEs. The staff noted that RRS-SPLA-BL and RRS-RWG-1 have 95th percentile frequencies in the AOO region. The staff determined that further evaluation of these LBEs as AOOs is not needed because they have substantial margin to the F-C target curve based on the doses reported in the PSAR. The bounding DBEs in this section that serve as the basis for DBAs are:

- RRS-CGR-1, which is the basis for RRS-CGR-CN

- RRS-ISPL-BL, which is the basis for RRS-ISPL-CN
- RRS-SPLX-BL, which is the basis for RRS-SPLX-CN
- RRS-SPLA-BL, which is the basis for RRS-SPLA-CN
- RRS-RWG-2, which is the basis for RRS-RWG-CN

The DBAs are discussed in more detail in section 3.8.1.3 of this SE.

Based on the staff's review described above, the staff concluded that the information provided relative to the release from ex-vessel systems DBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each DBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also determined that the DBEs were appropriately categorized as DBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7. The staff confirmed that DBEs identified as limiting were appropriately mapped to DBAs described in PSAR section 3.9.

3.6.1.4 Fuel Handling DBEs

The staff reviewed PSAR section 3.7.4 that describes the DBEs that are initiated by fuel handling events, including:

- Fuel handling event occurs while moving fuel assembly in the reactor vessel with no damage (RFH-FDIV-BL)
- Fuel handling event occurs while moving fuel assembly in the reactor vessel with a single assembly failed and functional containment barriers successfully retain release (RFH-FDIV-1)
- Fuel handling event occurs while moving fuel assembly in the reactor vessel with two assemblies failed and functional containment barriers successfully retain release (RFH-FDIV-3)
- Fuel handling event occurs while moving fuel assembly in the SFP with a single assembly failed and FHB barrier fails to retain release (RFH-FDSP-1)
- Excessive sodium-water reaction in the PIC (RFH-ESWR-BL)
- Excessive sodium-water reaction in the PIC with PIC and BLTC barrier successfully retains release (RFH-ESWR-1)
- Loss of EVST active cooling while storing fuel assembly without passive cooling (RFH-LTCA-1)
- Loss of BLTC active cooling while handling fuel assembly with passive cooling (RFH-LBCA-BL)

- Loss of EVHM active cooling while handling fuel assembly or an LTA or LDA with passive cooling (RFH-LMCA-BL)

These LBEs address the potential for releases of radionuclides to the environment caused by fuel handling events.

As described in PSAR sections 3.7.4.1, 3.7.4.2, and 3.7.4.3, RFH-FDIV-BL, -1, and -3 all relate to fuel handling events that occur while moving fuel in the reactor. RFH-FDIV-BL is the LBE where the event does not result in fuel damage, and consequently there is no release. RFH-FDIV-1 and -3 are for events with one and two high burnup fuel assemblies damaged by the fuel handling event, respectively. The staff evaluated the source terms and assumptions regarding functional containment performance in section 3.2.1.2 of this SE, where they were found to reasonably represent the in-vessel fuel handling events. Consistent with the functional containment performance assumptions, the LBEs that result in release appropriately credit the primary system boundary and the EVHM barrier. As discussed in detail in section 5.4.3 of this SE, an SR SCG isolation function was added by USO during the review. This function is credited to mitigate releases in these LBEs and is necessary for the primary functional containment barrier to achieve its safety function.

As described in PSAR section 3.7.4.4, RFH-FDSP-1 involves a fuel handling event in the SFP. When the fuel is damaged in the SFP, the fuel bond sodium mixes with the water and reacts, which increases the temperature. The applicant stated in PSAR section 3.7.4.4.2 that the effect of this phenomenon on radionuclide retention in the spent fuel pool is still being studied but assumed “relatively low” decontamination factors for the PSAR analysis of SFP fuel handling events. As stated in the letter submitted by USO dated September 17, 2025, (ML25260A002) (PIC supplement), an analysis methodology specific to events involving the sodium-water reaction with considerations for the impact of pressure and temperature effects driving radionuclide releases from fuel will be developed by USO. The staff will review this methodology at the OL stage. The staff evaluated the source terms and assumptions regarding functional containment performance in section 3.2.1.3 of this SE, where they were found to reasonably represent the SFP fuel handling events. Radionuclides that get through the SFP to the FHB atmosphere are assumed to be exhausted to the environment with no filtration. Consistent with the functional containment performance assumptions, no functional containment barriers are credited.

As described in PSAR sections 3.7.4.5 and 3.7.4.6, RFH-ESWR-BL and -1 represent excessive sodium-water reactions in the PIC. The PIC is used to clean sodium off spent fuel assemblies prior to their insertion in the SFP, as described in PSAR section 7.3.1.1. Because water is used in the cleaning process and the PIC is used to transport core assemblies underwater, there is a possibility that an abnormal quantity of water could interact with the residual sodium on a fuel assembly, causing a sodium-water reaction that could damage a fuel assembly in the PIC. RFH-ESWR-BL represents a case where there is an excessive sodium-water reaction that does not cause fuel damage, so there is no radionuclide release. RFH-ESWR-1 assumes the reaction is sufficient to cause failure of all pins in a single high-burnup assembly. As discussed in section 3.2.1.3 of this SE, the source term used for the PIC was generated by applying scaling to other source terms and did not model the filtered pathway that is not isolated during the event. In addition to the further development of the methodology for ESWR, USO will also be developing event specific source term models that represent the detailed system configuration of the PIC, as discussed in the PIC supplement.

The safety analysis and SSC safety classification for the CP stage assumed that the BLTC was mated to the PIC throughout the PIC operations needed to transition an assembly into the SFP. This resulted in the identification of two different safety functions involving the same PIC radionuclide retention barrier. For the function where the PIC is mated to the BLTC, this is SR function DL3-RR5a. For the PIC on its own, this is NSRST function DL2-RR2. The tables within section 5.2 of the PSAR lists RFH-ESWR-1 as an associated LBE for both functions. Through audit discussions, the staff gained a better understanding of what was assumed and modeled in the PSAR analysis, how the design has continued to evolve, and what additional controls have been put in place that may impact these events. The PIC supplement provides a summary of this information. From this information and details confirmed in audit, the staff determined that:

- The BLTC is not expected to be mated with the PIC during the washing operation. Therefore, the mated BLTC/PIC (DL3-RR5a) is the appropriate radionuclide barrier for the drop events (RFH-FDPI), but not for the washing events (RFH-ESWR).
- The methods and assumptions used to calculate source terms for the ESWR events were not consistent, making it difficult to determine the contribution of each function to the LBE mitigation.
- The current design includes multiple preventative controls that were not modeled within the PRA when the CP application analysis was performed.

While uncertainty in the design remains, the staff determined that the information provided was sufficient to support the CP based on:

- The PIC radionuclide retention barrier SSCs are currently classified as SR through the DL3-RR5a SR function.
- The additional preventative controls are expected to reduce the frequencies for these LBEs farther from the F-C target curve, as described in the PIC supplement.
- While there is still development needed in the source term methodology for modeling these LBEs, the analysis used assumed that all pins fail and release their fission products to the functional containment and no credit is taken for settling of particulates during holdup. These are conservative assumptions that could be adjusted as the event specific mechanistic source term is developed at the OL stage.

As described in PSAR section 3.7.4.7, RFH-LTCA-1 relates to a loss of all cooling in the EVST, which is used to store fuel assemblies after they are discharged from the reactor but before they are cleaned and put into the SFP. The PSAR analysis shows that the heat capacity of the sodium is sufficient to absorb a limiting decay heat load for at least 72 hours without fuel damage. The staff confirmed through audit of the analysis performed that the limiting fuel assembly in the EVST would remain below a temperature at which fuel damage would be expected. Because 72 hours is the typical time before recovery actions are assumed for LBEs, the staff finds the result to be reasonable.

As described in PSAR sections 3.7.4.8 and 3.7.4.9, RFH-LBCA-BL and RFH-LMCA-BL relate to loss of active cooling on the BLTC and EVHM, respectively. However, passive cooling systems of the BLTC and EVHM are credited to function. As described in PSAR sections 7.3.2.1.2 and 7.3.2.1.1, the BLTC passive cooling system is designed to remove 1.2 kW of decay heat,

which the staff determined represents about 1 year decay (considering both in-vessel and EVST storage), and the EVHM passive cooling system is designed to handle core assemblies 7 days after shutdown with a decay heat of 15.2 kilowatts. The staff confirmed that the passive heat removal rates are reasonable for the specified fuel temperature and arrangement based on simplified calculations of radiative heat transfer and that the minimum decay times have a high probability of being satisfied, as discussed in section 7.3.2.1.7 of this SE. Therefore, the staff finds the result of fuel cladding integrity being maintained and no radiological release reasonable for these LBEs.

As discussed in the response to RCI-3, these events have mean frequencies between 1×10^{-2} and 1×10^{-4} per year and are appropriately categorized as DBEs. The staff noted that RFH-LMCA-BL has its 95th percentile frequency in the AOO region. This LBEs does not result in a release, so no further evaluation is needed. All the events discussed in this section of the SE are mapped to DBAs, including:

- RFH-FDIV-3, which maps to RFH-FDIV-CN
- RFH-FDSP-1, which maps to RFH-FDSP-CN
- RFH-ESWR-1, which maps to RFH-ESWR-CN
- RFH-LTCA-1, which maps to RFH-LTCA-CN
- RFH-LBCA-BL, which maps to RFH-LBCA-CN
- RFH-LMCA-BL, which maps to RFH-LMCA-CN

The DBAs are discussed in more detail in section 3.8.1.4 of this SE.

Based on the staff's review described above, the staff concluded that the information provided relative to fuel handling DBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each DBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the DBEs were appropriately categorized as DBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7. The staff confirmed that DBEs identified as limiting were appropriately mapped to DBAs described in PSAR section 3.9.

3.6.1.5 Local Fuel Fault DBE

The staff reviewed PSAR section 3.7.5 on the DBE initiated by local fuel faults, which includes partial flow blockage and fuel misloads as discussed in section 2.4 of NAT-9394-A. The one DBE in this section is:

- Core blockage and local faults with non-passive IAC and functional containment barrier successfully retains release (LFF-SAO-BL)

The initiating event affects a single fuel assembly, which is assumed to fail in the analysis as discussed in PSAR section 3.3.4.1.5.

For LFF-SAO-BL, control of heat generation is provided by a manual scram initiated by operators in response to indications in the cover gas system that failed fuel exceeds limits. No specific function is associated with this event in the tables in PSAR chapter 5, but a manual scram function is listed as DL4-RC1. The staff expects operator actions to be further developed and incorporated into the design and safety analysis prior to the applicant's OL application. Decay heat removal is provided by non-passive IAC. The staff determined this is an appropriate assumption since there is no core-wide transient.

Radionuclides released from the failed fuel assembly travel into the sodium pool and the cover gas volume, then leak through the primary coolant boundary, to the HAA, and to the environment. The staff's evaluation of the source terms and assumptions regarding functional containment performance for this event in section 3.2.1.1 of this SE, where they were found to reasonably represent at-power in-vessel events. Consistent with the source term discussion, credited functions include the primary coolant boundary, the ex-RES functional containment barrier, and HAA. As discussed above, the SR SCG isolation is needed for the primary coolant boundary to achieve the performance level credited in the analysis.

The staff notes that the assumption of a failed fuel assembly for LFF-SAO-BL is conservative given preliminary analyses conducted using the partial flow blockage methodology described in NAT-9395-A demonstrate small numbers of failed fuel pins are expected for blockages that are expected in the DBE frequency range and misloads are generally not expected to be consequential as discussed in NAT-9394-A. However, it is the staff's understanding that these IEs cannot be detected until fuel failure occurs due to [[

]]. Because of this, the staff considers this conservative assumption to be necessary to bound the effects of a local fault.

As discussed in the response to RCI-3, the event has a mean frequency between 1×10^{-2} and 1×10^{-4} per year and is appropriately categorized as a DBE. The staff notes that the initiating event frequency for this LBE was assumed to be 1×10^{-3} with an error factor of 10 as discussed in PSAR table 3.1-1. The staff audited the applicant's documentation related to this assumption, and while the initial basis for the 1×10^{-3} initiating event frequency was based on generic data that did not appear to be fully applicable, subsequent design-specific analyses audited by the staff confirmed that the frequency of partial flow blockages is expected to be bounded by the frequency assumed in the KU1 PRA. LFF-SAO-BL is the most limiting DBE that serves as the basis for DBA LFF-SAO-CN, which is discussed in more detail in section 3.8.1.6 of this SE.

Based on the staff's review described above, the staff concluded that the information provided relative to the local fault DBE was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the DBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the DBE was appropriately categorized as a DBE, and that it credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7. The staff confirmed that DBE was appropriately mapped to a DBA described in PSAR section 3.9.

3.6.2 Conclusion

The staff finds that the DBE results meet the requirements in 10 CFR 50.34 and is consistent with the guidance in NEI 18-04, as endorsed with clarifications in RG 1.233. The level of detail provided in the applicant's PSAR is consistent with the guidance in NEI 21-07, as endorsed with clarifications in RG 1.253, considering the preliminary nature of the design at the CP stage.

The staff finds the DBE results to be acceptable and support the safety conclusions. Therefore, the staff finds that the DBE analyses adequately support the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1), (a)(3)(ii), and (4) and 10 CFR 50.35, as applicable.

3.6.3 Tables

Table 3.6-1: Summary of design basis events from PSAR section 3.7

PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.7.1.1	Loss of One Primary Sodium Pump with RAC	DHP-L1PP-2	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1	RAC, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature, natural circulation of sodium in primary system	DL3-HR4, DL3-HR1, DL3-HR2, DL3-HR3, DL3-HR5	Fuel cladding	DL3-RR2
3.7.1.2	Loss of Offsite Power with Passive IAC	DHP-LOOP-1	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1	IAC passive mode	DL4-HR1	Fuel cladding	DL3-RR2
3.7.1.3	Loss of Offsite Power While at Low Power with Passive IAC	SUD-LOOP-1	Loss of Primary Flow	Reactor core	Reactor scram on loss of primary flow	DL3-RC1	IAC passive mode	DL4-HR1	Fuel cladding	DL3-RR2

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PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.7.2.1	Loss of Heat Sink with Passive IAC	DHS-ISTL-1	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high cold pool temperature	DL3-RC1	IAC passive mode, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature	DL4-HR1, DL3-HR1, DL3-HR2, DL3-HR3	Fuel cladding	DL3-RR2
3.7.2.2	Loss of Heat Sink with RAC	DHS-ISTL-2	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high cold pool temperature	DL3-RC1	RAC, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature, natural circulation of sodium in primary system	DL3-HR4, DL3-HR1, DL3-HR2, DL3-HR3, DL3-HR5	Fuel cladding	DL3-RR2
3.7.2.3	Energy Island Transient without Reactor Power Runback, with Non-Passive IAC	DHS-RNBK-1	Increase or Decrease in Heat Removal	Reactor core	Unspecified reactor scram	DL3-RC1	IAC non-passive mode	None*	Fuel cladding	DL3-RR2

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PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.7.2.4	Energy Island Transient without Reactor Power Runback, with RAC	DHS-RNBK-3	Increase or Decrease in Heat Removal	Reactor core	Unspecified reactor scram	DL3-RC1	RAC, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature, natural circulation of sodium in primary system	DL3-HR4, DL3-HR1, DL3-HR2, DL3-HR3, DL3-HR5	Fuel cladding	DL3-RR2
3.7.2.5	Loss of One Train of IAC While Shutdown with Passive IAC	SUD-IACA-1	Increase or Decrease in Heat Removal	Reactor core	None	None	IAC passive mode	DL4-HR1	Fuel cladding	DL3-RR2
3.7.2.6	Intermediate Heat Exchanger Secondary-to-Primary Leak with Non-Passive IAC	IPI-IHEL-BL	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high primary sodium level	DL3-RC1	IAC non-passive mode, ISP trip on high high primary sodium level	None*, DL3-HR12	Fuel cladding	DL3-RR2
3.7.2.7	Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown with Non-Passive IAC	SUD-IHEL-BL	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high primary sodium level (for low power condition)	DL3-RC1	IAC non-passive mode, ISP trip on high high primary sodium level	None*, DL3-HR12	Fuel cladding	DL3-RR2

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PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.7.3.1	SCG Leak Inside the HAA where RXB Substructure Successfully Retains Release	RRS-CGR-BL	Release from Ex-Vessel Systems	Cover gas radionuclide inventory	N/A	N/A	N/A	N/A	Ex-RES functional containment barrier, HAA barrier, HAA HVAC operations following postulated release	DL4-RR1, DL4-RR1a, DL4-RR1c
3.7.3.2	SCG Leak Downstream the SCG Cell where RXB Substructure Successfully Retains Release	RRS-CGR-1	Release from Ex-Vessel Systems	Cover gas radionuclide inventory	N/A	N/A	N/A	N/A	Ex-RES functional containment barrier, HAA barrier, HAA HVAC operations following postulated release	DL4-RR1, DL4-RR1a, DL4-RR1c
3.7.3.3	SPS Leak at the Cold Trap with RAB Barrier Successfully Retains Release	RRS-ISPL-BL	Release from Ex-Vessel Systems	SPS-I cold trap tritium inventory	N/A	N/A	N/A	N/A	None	None
3.7.3.4	SPS-P Leak in the RXB	RRS-SPLX-BL	Release from Ex-Vessel Systems	SPS-P inventory	N/A	N/A	SPS pump trip on low low primary sodium level	DL3-HR11	Ex-RES functional containment barrier	DL4-RR1
3.7.3.5	SPS-P Leak in the RAB	RRS-SPLA-BL	Release from Ex-Vessel Systems	SPS-P inventory	N/A	N/A	SPS pump trip on low low primary sodium level	DL3-HR11	SPS cells barrier	DL4-RR3

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PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.7.3.6	RWG Leak from the Holdup Tank with FHB Barrier Fails to Retain Release	RRS-RWG-1	Release from Ex-Vessel Systems	RWG holdup tank inventory	N/A	N/A	N/A	N/A	NHV filtration	None*
3.7.3.7	RWG Leak to the Stack to the Environment	RRS-RWG-2	Release from Ex-Vessel Systems	RWG holdup tank inventory	N/A	N/A	N/A	N/A	NHV filtration	None*
3.7.4.1	Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with No Damage	RFH-FDIV-BL	Fuel Handling	Single fuel assembly	N/A	N/A	N/A	N/A	Fuel cladding	DL3-RR2
3.7.4.2	Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with a Single Assembly Failed and Functional Containment Barriers Successfully Retain Release	RFH-FDIV-1	Fuel Handling	High burnup fuel assembly	N/A	N/A	N/A	N/A	Primary system boundary, EVHM cask barrier	DL3-RR1, DL3-RR3

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PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.7.4.3	Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with Two Assemblies Failed and Functional Containment Barriers Successfully Retain Release	RFH-FDIV-3	Fuel Handling	Two high burnup fuel assemblies	N/A	N/A	N/A	N/A	Primary system boundary, EVHM cask barrier	DL3-RR1, DL3-RR3
3.7.4.4	Fuel Handling Event Occurs While Moving Fuel Assembly in the SFP with a Single Assembly Failed and FHB Barrier Fails to Retain Release	RFH-FDSP-1	Fuel Handling	High burnup fuel assembly, 310 day decay	N/A	N/A	N/A	N/A	None	None
3.7.4.5	Excessive Sodium-Water Reaction in the PIC	RFH-ESWR-BL	Fuel Handling	Single fuel assembly	N/A	N/A	N/A	N/A	Fuel cladding	DL3-RR2
3.7.4.6	Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Successfully Retains Release	RFH-ESWR-1	Fuel Handling	High burnup fuel assembly, 310 day decay	N/A	N/A	N/A	N/A	PIC radionuclide boundary, BLTC barrier, BLTC transfer barrier	DL2-RR2, DL3-RR5, DL3-RR5a

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PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.7.4.7	Loss of EVST Active Cooling While Storing Fuel Assembly without Passive Cooling	RFH-LTCA-1	Fuel Handling	EVST contents	N/A	N/A	None	None	Fuel cladding	DL3-RR2
3.7.4.8	Loss of BLTC Active Cooling While Handling Fuel Assembly with Passive Cooling	RFH-LBCA-BL	Fuel Handling	Single fuel assembly	N/A	N/A	Passive heat removal in BLTC	DL3-HR8	Fuel cladding	DL3-RR2
3.7.4.9	Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA with Passive Cooling	RFH-LMCA-BL	Fuel Handling	Single fuel assembly	N/A	N/A	Passive heat removal in EVHM	DL3-HR6	Fuel cladding	DL3-RR2
3.7.5.1	Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Successfully Retains Release	LFF-SAO-BL	Local Fuel Fault	Reactor core	Manual reactor scram	None*	IAC non-passive mode	None*	Primary coolant boundary, ex-RES functional containment barrier, HAA barrier	DL3-RR1, DL4-RR1, DL4-RR1a

* No safety-significant PSF is identified for these functions in the tables in PSAR chapter 5 because they are NST.

3.7 Beyond Design Basis Events

PSAR section 3.8 describes the analysis of BDBEs selected through use of methodologies described in PSAR sections 3.3 and 3.4 for inclusion in the plant's licensing basis.

The applicable regulatory requirements and guidance for the evaluation of the BDBE analysis are identical to those described in section 3.6 of this SE.

3.7.1 Technical Evaluation

As discussed in section 3.4.1 of this SE and shown in table 3.4-1, the staff noted that there are two events that the LMP methodology as outlined in NEI 18-04 and RG 1.233 would have categorized as BDBEs based on the 95th percentile frequency that USO did not include in the PSAR. This is because USO used the mean frequency when categorizing events as BDBEs against the 5×10^{-7} lower bound. USO confirmed in RCI-1 that it intends to continue using the mean at the OL stage for event categorization but evaluates LBEs based on the 95th percentile frequency. Because USO still performs the LMP evaluations and SSC classification steps on events that fall below the BDBE frequency cutoff, the staff determined that USO's use of the 50th percentile did not impact the conclusions in the PSAR. The staff intends to continue to evaluate BDBEs based on the 95th percentile frequency when reviewing the OL application.

The results for RFH-OERC-EX1 provide insight into the preventive and mitigative credit being attributed to the PRC barrier, though, as discussed in section 3.4.1 and 3.5.1 of this SE, the reliability and capability reflected by the safety analysis may not be achievable. Considering the initiating event family relative to the F-C target (see figure 3.7-1), this may shift RFH-OERC-EX1 to higher frequency and consequence, but the addition of a high consequence BDBE that would result in classifying the PRC as SR does not have affect SSC safety classifications because the PRC barrier is already SR based on the AOO and DBA evaluations as discussed in section 3.5.1 of this SE.

The staff evaluated the PSAR section 3.8 BDBE descriptions for their conformance with NEI 21-07, which states in section 3.4.1 that, for each BDBE, the PSAR should contain a definition of the initial plant conditions and plant operating state, radionuclide source, IEs covered in the family, characterization of the responses of SSCs that perform PSFs, operator actions that perform PSFs, identification of whether or not there is a release, and definition of the safe, stable end state. NEI 21-07 also states BDBEs with releases should provide additional information describing the mechanistic source term, the mean, 5th, and 95th percentile frequency and dose, and significant factors that influence any degradation of layers of defense. For a CP, NEI 21-07 additionally notes that the assessments are preliminary and the information provided should reflect the preliminary nature of the design information.

The staff assessed the BDBEs to ensure they were appropriately categorized as BDBEs based on their frequency (upper 95th percentile frequency above 5×10^{-7} and mean frequency less than 1×10^{-4}). The staff also evaluated the credited functions to ensure they are consistent with their descriptions in chapter 7 of the PSAR and the safety classifications and tables in PSAR chapter 5. For BDBEs with releases, the staff confirmed that the BDBEs remained within the F-C target curve and assessed the credited functional containment barriers and their assumed performance to ensure they appropriately reflected the plant design, operating state, available isolation functions, filters, etc. The staff's detailed evaluation of mechanistic source terms is provided in section 3.2 of this SE.

3.7.1.1 Loss of Primary Flow BDBEs

The staff reviewed PSAR section 3.8.1 on the loss of primary flow BDBEs, including:

- Loss of one primary sodium pump with passive IAC (DHP-L1PP-1)
- Loss of one primary sodium pump with scram motor drive-in (DHP-L1PP-3)
- Loss of one primary sodium pump with alternative shunt trip (DHP-L1PP-4)
- Loss of all primary sodium pumps with non-passive IAC (DHP-LAPP-BL)
- Loss of offsite power with RAC (DHP-LOOP-2)
- Loss of offsite power with scram motor drive-in (DHP-LOOP-3)
- Loss of offsite power with alternative shunt trip (DHP-LOOP-4)
- Loss of a single medium voltage AC bus with passive IAC (OTH-LMAC-1)
- Loss of a single medium voltage AC bus with RAC (OTH-LMAC-2)
- Loss of a single medium voltage AC bus with scram motor drive-in (OTH-LMAC-3)
- Loss of a single medium voltage AC bus with alternative shunt trip (OTH-LMAC-4)

The safety concern for these LBEs is an imbalance between heat generation in the reactor core and heat removal capability through the primary coolant. In all PSAR section 3.8.1 transients the initiating event is a failure that causes a loss of primary coolant flow. All the events in this section of the PSAR are initiated from full power conditions.

As described in PSAR sections 3.8.1.1, 3.8.1.2, and 3.8.1.3, the initiating event for DHP-L1PP-1, -3, and -4, is a failure that causes the loss of a single PSP while at full power. The staff confirmed in audit that the scenario represents a limiting reduction in flow (i.e., a locked rotor) for one PSP. The reduction in primary flow results in a reactor scram on high high power-to-flow ratio, which is consistent with what the staff would expect for loss of flow transients as discussed in section 3.5.1.1 of this SE. In DHP-L1PP-3, the control rod insertion by gravity fails and the rods are pushed in with the NSRST control rod drive follow function. In DHP-L1PP-4, the normal scram system fails to function and the scram breakers are tripped by the anticipatory automatic seismic trip system (AST), causing the control rods to fall into the core by gravity. DHP-L1PP-1 relies on passive IAC to remove decay heat, which means that passive IAC, PSP coastdown, and PSP and ISP pump trips on high high primary sodium temperature are appropriately credited, as first discussed in sections 3.6.1.1 and 3.6.1.2 of this SE. DHP-L1PP-3 and -4 rely on non-passive IAC for decay heat removal, which as first discussed in section 3.6.1.1 of this SE is an NST function not described in detail in the PSAR. Decay heat removal is sufficient to ensure fuel cladding integrity is maintained for all three LBEs, which the staff verified through audit of the supporting calculations.

As described in PSAR section 3.8.1.4, DHP-LAPP-BL is a total loss of all forced primary flow. The staff identified through audit of the analysis notebooks that this scenario does not represent

a dual locked rotor scenario, but rather that the PSPs are assumed to trip and coast down. The reduction in primary flow results in a reactor scram on high high power-to-flow ratio, which the staff finds reasonable as discussed above. Decay heat removal is provided by non-passive IAC, similar to DHP-L1PP-3 and -4 discussed above. The staff notes that the PSPs are designed such that a natural flow path through them exists when the shaft is stopped, as discussed in section 7.1.3.2.3 of the PSAR, enabling natural circulation to continue to remove heat from the core after the pumps coast down to zero forced flow. Decay heat removal is sufficient to ensure fuel cladding integrity is maintained, which the staff verified through audit of the supporting calculations.

As described in PSAR sections 3.8.1.5, 3.8.1.6, and 3.8.1.7, DHP-LOOP-2, -3, and -4 result from a loss of offsite power, which causes the PSPs and ISPs to trip and coast down. For all three events, the reduction in primary flow results in a reactor scram on high high power-to-flow ratio, which the staff finds reasonable as discussed above. DHP-LOOP-2 relies on RAC and appropriately credits the RAC and natural circulation in the primary system; pump trips are not credited because they occur due to the loss of offsite power, and the staff confirmed in audit that the pumps begin to coast down at the initiation of the transient. DHP-LOOP-3 and -4 rely on and appropriately credit non-passive IAC, similar to DHP-L1PP-3 and -4 discussed above. As discussed in section 3.5.1.1 of this SE, the LOOP cases relying on non-passive IAC also credit restoration of electric power to IAC provided by the diesel generators. The staff confirmed through audit that an appropriate delay to account for generator startup was modeled. Fuel cladding integrity is maintained for DHP-LOOP-2, but DHP-LOOP-3 and -4 result in fuel failures. The mechanistic source terms for these LBEs rely on the primary functional containment barrier, which requires the SR SCG isolation function discussed in section 3.6.1 and 5.4 of this SE. As noted in the PSAR sections describing these LBEs, the SCG isolation function has not been fully incorporated into the PSAR and will be evaluated at the OL stage; the staff will review additional information about the SCG isolation function at that time. The staff's evaluation of the source terms and assumptions regarding functional containment performance for this event is discussed in section 3.2.1.1 of this SE, where they were found to reasonably represent at-power in-vessel events.

As described in PSAR sections 3.8.1.8, 3.8.1.9, 3.8.1.10, and 3.8.1.11, OTH-LMAC-1, -2, -3, and -4 result from a loss of power to a medium voltage bus, which disables one PSP and one ISP. OTH-LMAC-1 relies on passive IAC, and appropriately credits the IAC system, PSP coastdown, and PSP and ISP trips. OTH-LMAC-2 relies on RAC, and appropriately credits the RAC system, PSP and ISP trips, and primary system natural circulation. OTH-LMAC-3 and -4 both rely on non-passive IAC. The staff identified in the audit that OTH-LMAC-3 and -4 were not explicitly analyzed for the PSAR but instead were identified as bounded by analysis that credited only RAC. Because RAC is less capable of removing heat quickly than non-passive IAC, the staff determined this is reasonable for a preliminary analysis but notes that a full analysis or appropriate justification would be needed for final analysis at the OL.

As discussed in the response to RCI-3, these events have mean frequencies less than 1×10^{-4} per year and upper 95th percentile frequencies greater than 5×10^{-7} per year and are appropriately categorized as BDBEs. Several of the events have 95th percentile frequencies that extend into the DBE range and should have DBAs identified per footnote 37 of Appendix A to NEI 21-07, including DHP-L1PP-1, DHP-L1PP-4, and OTH-LMAC-1.

While DHP-L1PP-1 is not assigned to the DBA DHP-L1PP-CN, DHP-L1PP-1 does not assume the failure of any SR equipment and is appropriately bounded by DHP-L1PP-CN. If

DHP-L1PP-1 remains as a BDBE with uncertainty bands that extend into the DBE range at the OL, a DBA should be appropriately assigned.

DHP-L1PP-4 assumes failure of the scram on high high power-to-flow ratio and as a result, DHP-L1PP-CN is not bounding for this event. If DHP-L1PP-4 remains as a BDBE with uncertainty bands that extend into the DBE range at the OL, a new DBA will be needed. With the frequencies and consequences included in the PSAR analyses, evaluation of this DBA is anticipated to elevate the alternative shunt trip credited in this LBE from NSRST to SR. The staff confirmed that an appropriate level of design control is in place to accommodate the potential for DBA and credited SSCs, which are already safety-significant SSCs, while this event is being re-evaluated for the OL. Additionally, the staff confirmed USO has awareness of the potential effects of this evaluation. Based on these considerations, the staff determined that review of this evaluation can reasonably be deferred to the OL stage based on the preliminary nature of the design.

With respect to OTH-LMAC-1, the staff identified that it does not assume the failure of any SR functions and another event in the family, OTH-LMAC-2, credits only SR SSCs and demonstrates that fuel integrity is maintained. Also, OTH-LMAC-2 is not identified in PSAR section 3.3.2.1 as one of the LBEs using the BEPU approach and uses a conservative evaluation consistent with the DBAs, as discussed in section 3.3.1.6 of this SE. The staff determined that OTH-LMAC-2 provides an adequate representation in the PSAR of the needed SSC performance and potential consequences associated with a DBA developed from OTH-LMAC-1. If OTH-LMAC-1 remains as a BDBE with uncertainty bands that extend into the DBE range at the OL stage, a DBA should be appropriately assigned.

Based on the staff's review described above, the staff concluded that the information provided relative to the loss of primary flow BDBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each BDBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the BDBEs were appropriately categorized as BDBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.7.1.2 Increase or Decrease in Heat Removal BDBEs

The staff reviewed PSAR section 3.8.2 on the loss of primary flow BDBEs including:

- Loss of heat sink with scram motor drive-in (DHS-ISTL-3)
- Loss of heat sink with alternative shunt trip (DHS-ISTL-4)
- Energy island transient without reactor power runback, with passive IAC (DHS-RNBK-2)
- Loss of one train of IAC while shutdown with RAC (SUD-IACA-2)
- Intermediate heat exchanger secondary-to-primary leak with passive IAC (IPI-IHEL-1)
- Intermediate heat exchanger secondary-to-primary leak with RAC (IPI-IHEL-2)

- Intermediate heat exchanger secondary-to-primary leak while shutdown with passive IAC (SUD-IHEL-1)

These BDBEs address an imbalance between heat generation in the core and heat removal to the ultimate heat sink.

As discussed in PSAR sections 3.8.2.1 and 3.8.2.3, DHS-ISTL-3 and -4 result from a loss of heat removal from one intermediate sodium loop. The credited reactivity control function is a scram on high high cold pool temperature, which the staff finds reasonable consistent with the discussion in SE section 3.5.1.2 for DHS-ISTL-BL. In DHS-ISTL-3 the control rod insertion fails and NSRST CRD driveline scram follow takes over, while in DHS-ISTL-4 the normal scram system fails to function and the scram breakers are tripped by AST, causing the control rods to fall into the core by gravity. Decay heat removal for both events is provided by non-passive IAC, which the staff finds is consistent with the IAC system design as discussed in SE section 3.5.1.2 even though one train of IAC is not available. Decay heat removal is sufficient to ensure fuel cladding integrity is maintained for both events, as confirmed by the staff through audit of supporting calculations.

As discussed in PSAR section 3.8.2.3, that the initiating event for DHS-RNBK-2 is a runback demand initiated by an increase or decrease in heat removal from the NSS during full power operation. Under normal conditions, the demand would initiate a power runback which, as described in PSAR section 1.3.2.2 and 3.3.4.1, results in insertion of control rods and reduction in PSP and ISP speed. In this LBE, which is intended to bound events that result in a runback demand, the runback is not completed prior to an assumed reactor scram, which shuts down the reactor. The scram signal is not specified in the PSAR, which the staff finds to be reasonable because the scram is assumed to occur as part of the analysis. Decay heat removal is provided by passive IAC for DHS-RNBK-2, with appropriate credit for PSP coastdown and safety-related PSP and ISP trips. Fuel cladding integrity is maintained. The staff noted in the audit that the specific scenario of DHS-RNBK-2, with passive IAC, was not analyzed but was considered bounded by analyses that considered only RAC heat removal and demonstrated fuel failure limits were not exceeded. Because RAC is less capable of removing heat quickly than passive IAC, the staff determined this is reasonable for a preliminary analysis but notes that a full analysis or appropriate justification would be needed for final analysis at the OL.

As discussed in PSAR section 3.8.2.4 the initiating event for SUD-IACA-2 is a loss of IAC at a low power condition as discussed in sections 3.5.1.2 and 3.6.1.2 of this SE for SUD-IACA-BL and -1. As with SUD-IACA-BL and -1, a reactor trip is not credited. The LBE relies on RAC for decay heat removal and credits the RAC system and natural circulation in the primary system. Pump trips are not credited. The supporting analysis demonstrates fuel cladding integrity is maintained, but the staff identified in audit that the event was only analyzed once as a DBA and the results were applied to both SUD-IACA-CN and SUD-IACA-2. The staff notes that the credited SSCs are identical to the SUD-IACA-CN DBA case, as discussed in section 3.8.1.2 of this SE, and the DBA results have more conservative assumptions than BDBEs, so the staff determined it is acceptable to apply the results of the DBA to the BDBE.

As discussed in PSAR sections 3.8.2.6 and 3.8.2.7, the initiating event for IPI-IHEL-1 and -2 and SUD-IHEL-1 is a leak in the IHX that causes intermediate sodium to flow into the primary system. The immediate safety concern with this kind of event is that the quantity of sodium that leaks into the primary system would be enough to overfill the reactor vessel and pose a challenge the primary coolant boundary. PSAR sections 3.8.2.5.1 and 3.8.2.6.1 describes that the vessel overfill is prevented by the IHT physical arrangement as discussed in PSAR section

7.1.4. PSAR section 7.1.4 states that the physical arrangement that limits the overfill is based on a high point downstream of the primary sodium pump with a siphon break provided by the expansion tank; in the event of drain down of the IHT loop, the high point fills with inert gas and prevents the entire IHT inventory from being siphoned into the reactor. The anti-siphon feature is reliant on a trip of the ISP on the fault IHT train, which is provided as a SR ISP trip on high high primary sodium level (DL3-HR12). IPI-IHEL-1 and 2 and SUD-IHEL-1 credit this trip to prevent the overfill as discussed in PSAR sections 3.8.2.5.1 and 3.8.2.6.1 and PSAR table 5.2-2, and indicated in SE table 3.7-1, below.

Because IPI-IHEL-1 and 2 and SUD-IHEL-1 result in the loss of heat removal from a single train of IHT even when the leakage is terminated, these LBEs must be evaluated to ensure there is adequate decay heat removal. IPI-IHEL-1 and SUD-IHEL-1 credit passive IAC for decay heat removal, with appropriate pump trips for the PSPs and PSP coastdown credited in addition to the ISP trip on high high primary sodium level. However, as discussed in PSAR section 7.2.2.3, a single loop of passive IAC is not sufficient to remove the limiting decay heat load. Though it is not explicitly credited in the event descriptions or PSAR table 5.2-2, RAC is available and would be expected to participate in decay heat removal for these LBEs. IPI-IHEL-2 credits RAC for decay heat removal, along with a PSP trip on high high primary sodium temperature, PSP coastdown, and natural circulation. In all three cases, there is sufficient decay heat removal to ensure fuel cladding integrity is maintained.

The staff audited the documentation for IPI-IHEL-1 and -2 and SUD-IHEL-1 and observed that the analyses performed in support of the PSAR are, in some instances, not consistent with the PSAR event descriptions. For these particular LBEs, the primary coolant overfill portion of the transient was addressed in one analysis and the loss of the affected train of IHT for heat removal purposes was considered bounded by another analysis crediting only RAC. The staff considers this to be adequate for preliminary analyses because the key phenomena are addressed by analyses that bound the expected behavior. However, the staff anticipates the OL application to provide final analyses that correspond to the event description or appropriate justification in the FSAR that a different analysis is bounding.

As discussed in the response to RCI-3, these events have mean frequencies less than 1×10^{-4} per year and upper 95th percentile frequencies greater than 5×10^{-7} per year and are appropriately categorized as BDBEs. One event, DHS-RNBK-2, has a 95th percentile frequency that extends into the DBE range and should have DBA identified per footnote 37 of Appendix A to NEI 21-07. While DHS-RNBK-2 is not assigned to the DBA DHS-RNBK-CN, DHS-RNBK-2 does not assume the failure of any SR equipment and is appropriately bounded by DHS-RNBK-CN. If DHS-RNBK-2 remains as a BDBE with uncertainty bands that extend into the DBE range at the OL, the staff expects that a DBA would be appropriately assigned.

Based on the staff's review described above, the staff concluded that the information provided relative to the loss of primary flow BDBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each BDBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the BDBEs were appropriately categorized as BDBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.7.1.3 Reactivity BDBEs

The staff reviewed PSAR section 3.8.3, which describes the reactivity BDBEs, including the following transients:

- Control rod-induced transient overpower with passive IAC (RPD-CW1ACS-1)
- Control rod-induced transient overpower with RAC (RPD-CW1ACS-2)
- Control rod-induced transient overpower with scram motor drive-in (RPD-CW1ACS-3)
- Control rod-induced transient overpower with alternative shunt trip (RPD-CW1ACS-4)
- Reactor scram or spurious scram with passive IAC (RPD-SS-1)
- Reactor scram or spurious scram with RAC (RPD-SS-2)
- Reactor scram or spurious scram with scram motor drive-in (RPD-SS-3)
- Reactor scram or spurious scram with alternative shunt trip (RPD-SS-4)

These LBEs relate to unexpected changes in core reactivity.

As discussed in PSAR section 3.8.3.1 through 3.8.3.4, RPD-CW1ACS-1 through -4 result from a failure that causes a single control rod to continuously withdraw from the core while operating at full power. The increase in reactor power caused by the control rod withdrawal causes a reactor scram on either high high power range neutron flux or high high power to flow ratio. Similar to RPD-CW1ACS-BL as discussed in section 3.5.1.3 of this SE, the staff verified through audit that the assumed reactivity insertion curve is conservative with respect to the available control rod worth and the capacity of the motor drive system, and the reactivity insertion continues until the reactor scram. In RPD-CW1ACS-1 and -2, the normal gravity scram is successful. In RPD-CW1ACS-3, the control rod insertion by gravity is assumed to fail and the rods are pushed in with the NSRST control rod drive follow function. The staff confirmed through audit that the control rod insertion rate for RPD-CW1ACS-3 assumes a bounding reactivity insertion curve based on the control rod drive insertion speed, consistent with other events that rely on control rod drive scram follow. In RPD-CW1ACS-4, the normal protection system is assumed to fail to function and the scram breakers are tripped by AST, causing the control rods to fall into the core by gravity. RPD-CW1ACS-1 relies on passive IAC and credits the appropriate PSP and ISP trips (on high high primary sodium temperature) and PSP coastdown, while RPD-CW1ACS-2 relies on RAC and credits the appropriate PSP and ISP trips (also on high high primary sodium temperature), PSP coastdown, and primary system natural circulation. RPD-CW1ACS-3 and -4 rely on non-passive IAC, which, as discussed previously, is an NST function. Fuel cladding integrity is maintained for all four LBEs, as verified by the staff through audit of the supporting analyses.

For RPD-SS-1 through -4, as described in PSAR sections 3.8.3.5 through 3.8.3.8, the initiating event is an assumed reactor scram. In RPD-SS-1 and -2, the scram is assumed to be successful as part of the initiating event, so no additional function is credited to control heat generation. The staff determined this is reasonable because the initiating event results in the reactor in a subcritical condition. In RPD-SS-3, the control rod insertion by gravity is assumed to

fail and the rods are pushed in with the NSRST control rod drive follow function. In RPD-SS-4, the normal scram system fails to function and the scram breakers are tripped by AST, causing the control rods to fall into the core by gravity. Similar to the RPD-CW1ACS LBEs described above, RPD-SS-1 and -2 rely on passive IAC and RAC, respectively, and credit the appropriate pump trips consistent with the discussion above, and RPD-SS-3 and -4 rely on non-passive IAC. Fuel cladding integrity is maintained for all four LBEs, as confirmed by the staff through audit of the supporting analyses.

As discussed in the response to RCI-3, these events have mean frequencies less than 1×10^{-4} per year and upper 95th percentile frequencies greater than 5×10^{-7} per year and are appropriately categorized as BDBEs. The staff noted that RPD-SS-2 has a 95th percentile frequency in the DBE region, but no DBA is assigned. No SR SSCs are assumed to fail in RPD-SS-2, as confirmed through audit of associated calculational notebooks, with no fuel failure identified. The staff found the RPD-SS-2 transient to be relatively benign because it represents a spurious scram with only one complicating failure that results in the loss of IAC. As such, because there are more severe plant transients that rely on the same SR SSCs (RAC and associated pump trips and coastdown functions), the staff determined that the performance of these SR SSCs is adequately covered by other DBAs considered in the PSAR. If RPD-SS-2 remains as a BDBE with uncertainty bands that extend into the DBE range at the OL, the staff expects that a DBA would be appropriately assigned.

Based on the staff's review described above, the staff concluded that the information provided relative to the reactivity BDBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each BDBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the BDBEs were appropriately categorized as BDBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.7.1.4 Release from Ex-Vessel Systems BDBEs

The staff reviewed PSAR section 3.8.3, which describes the BDBEs that involve a release from ex-vessel systems, including:

- SCG leak in the SCG primary coolant boundary where RXB substructure fails to retain release while shutdown (SUD-CGR-1)
- SCG leak downstream of the SCG vapor trap cell where RXB substructure fails to retain release while shutdown (SUD-CGR-2)

These LBEs are postulated for systems that contain radionuclides, either from circulating radionuclides from failed fuel during operation or activated primary sodium, which have the potential to be released to the environment. Because the LBEs do not involve the potential for releases from the core, no reactor scram functions are needed or credited.

As described in PSAR section 3.8.4.1, SUD-CGR-1 assumes the entire SCG inventory is instantaneously released to the HAA. The HAA is then exhausted to the environment with filtration by NHV at a rate of 400 cfm. SUD-CGR-2, which is described in PSAR section 3.8.4.2, is similar, except the release is into the VTC. Results from SUD-CGR-1 are applied to SUD-CGR-2; the staff determined this was acceptable consistent with the discussion for

RRS-CGR-BL and -1 in SE section 3.6.1.3. The staff evaluated the source terms and assumptions regarding functional containment performance in section 3.2.1.4 of this SE, where they were found to reasonably represent the ex-vessel release events. Both LBEs credit appropriate functional containment barriers consistent with the source term discussion.

As discussed in the response to RCI-3, these events have mean frequencies less than 1×10^{-4} per year and upper 95th percentile frequencies greater than 5×10^{-7} per year and are appropriately categorized as BDBEs.

Based on the staff's review described above, the staff concluded that the information provided relative to the release from ex-vessel systems BDBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each BDBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the BDBEs were appropriately categorized as BDBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.7.1.5 Fuel Handling BDBEs

The staff reviewed PSAR section 3.8.4, which describes the BDBEs that are initiated by fuel handling events, including:

- Fuel handling event occurs while moving fuel assembly in the reactor vessel with a single assembly failed and functional containment barriers fail to retain release (RFH-FDIV-2)
- Fuel handling event occurs while moving fuel assembly in the reactor vessel with two assemblies failed and functional containment barriers fails to retain release (RFH-FDIV-4)
- Fuel handling event occurs while moving fuel assembly in the SFP with two assemblies failed and FHB barrier fails to retain release (RFH-FDSP-2)
- Excessive sodium-water reaction in the PIC with PIC and BLTC barrier fails to retain release (FHB credited) (RFH-ESWR-2)
- Excessive sodium-water reaction in the PIC with PIC and BLTC barrier fails to retain release (FHB not credited) (RFH-ESWR-3)
- Loss of EVHM active cooling while handling fuel assembly or an LTA or LDA without passive cooling, EVHM barrier successfully retains release (RFH-LMCA-1)
- Loss of EVHM active cooling while handling fuel assembly or an LTA or LDA without passive cooling, EVHM barrier fails to retain release (RFH-LMCA-2)
- Fuel handling event occurs while moving fuel assembly in the BLTC with BLTC barrier successfully retains release (RFH-FDBL-1)

- Fuel handling event occurs while moving fuel assembly in the BLTC with BLTC barrier fails to retain release (RFH-FDBL-2)
- Fuel handling event occurs while moving fuel assembly or an LTA or LDA in the EVHM with EVHM barrier successfully retains release (RFH-FDEM-1)
- Fuel handling event occurs while moving fuel assembly or an LTA or LDA in the EVHM with EVHM barrier fails to retain release (RFH-FDEM-2)
- Fuel handling event occurs while moving fuel assembly in the EVST with no damage (RFH-FDET-BL)
- Fuel handling event occurs while moving fuel assembly in the EVST with a single assembly failed and EVST barrier successfully retains release (RFH-FDET-1)
- Fuel handling event occurs while moving fuel assembly in the PIC with PIC barrier successfully retains release (RFH-FDPI-BL)
- Fuel handling event occurs while moving fuel assembly in the PIC with PIC barrier fails to retain release (RFH-FDPI-1)
- Fuel handling event occurs while moving fuel assembly in the PRC with PRC barrier successfully retains release (RFH-FDRC-1)

The concern addressed by these LBEs is that a fuel handling event could result in fuel damage and release of radionuclides to the environment.

As described in PSAR sections 3.8.5.1 and 3.8.5.2, RFH-FDIV-2 and -4 are initiated by a fuel handling event in the reactor vessel. For RFH-FDIV-1 the event causes the failure of one high burnup assembly, while for RFH-FDIV-2 the event causes the failure of two high burnup fuel assemblies. The staff evaluated the source terms and assumptions regarding functional containment performance for this event in section 3.2.1.2 of this SE, where they were found to reasonably represent the in-vessel fuel handling events. Consistent with the source term discussion in section 3.2.1.2 of this SE and the event description, no functional containment barriers are credited, though some degraded performance is assumed. As discussed for other LBEs involving the primary functional containment, performance of this barrier is dependent on the SR SCG isolation function that was added during the review. Based on the information available in the PSAR, the staff could not determine whether this isolation function is needed for the degraded performance of 100 percent leakage per day assumed in the analyses as described in the PSAR. Additional information on this isolation and the basis for any degraded performance credited in the source term analysis will be reviewed at the OL stage.

As described in PSAR section 3.8.5.3, RFH-FDSP-2 is initiated by a fuel handling event in the SFP that results in the failure of two high burnup fuel assemblies. The analysis assumes the assemblies have decayed for 310 days. The staff evaluated the source terms and assumptions for this event in section 3.2.1.3 of this SE, where they were found to reasonably represent the release into the SFP, although as USO noted in the PIC supplement, additional investigation into appropriate pool scrubbing values and the potential for ESWR to drive additional release is needed to support analysis for these events at the OL stage. Consistent with the source term

discussion in section 3.2.1.3 of this SE and the event description, no functional containment barriers are credited.

As described in PSAR sections 3.8.5.4 and 3.8.5.6, RFH-ESWR-2 and -3 are initiated by a failure that causes an ESWR in the PIC, which is assumed to result in the failure of the fuel assembly being cleaned. The staff notes that RFH-ESWR-2 is distinguished from the DBE ESWR events by the assumed failure of the PIC and BLTC boundary and RFH-ESWR-3 is distinguished by the failure of both the PIC and BLTC boundary and the FHB boundary. The source terms for these events are evaluated in section 3.2.1.3 of this SE, and the staff's overall conclusions regarding the PIC ESWR events are discussed there and in section 3.6.1.4 of this SE.

As described in PSAR sections 3.8.5.6 and 3.8.5.7, RFH-LMCA-1 and -2 are initiated by a failure that causes a loss of both active and passive cooling in the EVHM while it is handling an LTA or LDA. The MAR considered in the analyses, a single high burnup LTA or LDA with 2.5 day decay, is consistent with the event definition and is thus acceptable. While passive EVHM cooling is designed to be sufficient to cool a fuel assembly as soon as 7 days after removal from the core, as described in PSAR section 7.3.2.1.1, the loss of both active and passive cooling would be expected to result in fuel failure. Consistent with the source term discussion in section 3.2.1.3 of this SE, RFH-LMCA-1 credits the EVHM cask barrier and EVHM to head barrier function, while RFH-LMCA-2 credits no radionuclide barriers.

As described in PSAR sections 3.8.5.8 and 3.8.5.9, RFH-FDBL-1 and -2 are initiated by a fuel handling event that damages a fuel assembly in the BLTC. The MAR considered in the analyses, a single high burnup fuel assembly with 210 day decay, is consistent with the event definition. The staff notes that the BLTC is much less likely to load a low-decay fuel assembly than the EVHM because the BLTC does not handle LTAs or LDAs, regular fuel assemblies that are accidentally unloaded too early can be detected in the EVHM, and fuel assemblies can be tracked more easily in the EVST due to its smaller inventory. As such the decay of 210 days is likely to represent an upper bound on the decay heat and radionuclide inventory for the BLTC, and the staff therefore considers it to be acceptable. Consistent with the source term discussion in section 3.2.1.3 of this SE, both LBEs are modeled as a failure of all pins within the EVHM because the EVHM is expected to have bounding consequences compared to the BDBE. For RFH-FDBL-1 the BLTC barrier and BLTC transfer barrier are appropriately credited, while for RFH-FDBL-2 no radionuclide barrier is credited, consistent with the event definition and the source term analysis.

As described in PSAR sections 3.8.5.10 and 3.8.5.11, RFH-FDEM-1 and -2 are initiated by a fuel handling event that damages a fuel assembly in the EVHM. The events are very similar to RFH-FDBL-1 and -2, but assume a high-burnup LTA or LDA with only 2.5 days of decay. The staff determined this is appropriate for the EVHM, which moves LTAs and LDAs to the PRC for disassembly and post-irradiation inspection. Consistent with the event definition and source term analysis discussed in section 3.2.1.3 of this SE, RFH-FDEM-1 credits the EVHM cask barrier and EVHM to head barrier functions, and RFH-FDEM-2 credits no radionuclide barrier.

As described in PSAR sections 3.8.5.12 and 3.8.5.13, RFH-FDET-BL and -1 are initiated by fuel handling events that occur when moving fuel in the EVST. In RFH-FDET-BL, the fuel handling event does not lead to fuel damage and there is no release. In RFH-FDET-1, the source term is based on a single high burnup assembly with 210 days of decay, and the analysis is based on the EVHM model. The staff evaluated the source term and functional containment performance assumptions for this event in section 3.2.1.3 of this SE, where they were found to reasonably

represent the events. Consistent with the source term evaluation in this SE and the event description in the PSAR, the EVST barrier is the only barrier credited.

As described in PSAR sections 3.8.5.14 and 3.8.5.15, RFH-FDPI-BL and -1 are initiated by fuel handling events that occur when moving fuel in the PIC. These events occur while the BLTC is still attached to the PIC, because that is when the fuel is being moved. The source term for both events is based on a single high burnup assembly with 210 days of decay, and the analysis is based on the EVHM model. The staff evaluated the source term and functional containment performance assumptions for these events in section 3.2.1.3 of this SE, where they were found to reasonably represent the events. Consistent with the source term evaluation in this SE and the event description in the PSAR, RFH-FDPI-BL credits the NSRST PIC radionuclide boundary, SR BLTC barrier, and SR BLTC transfer barrier (combined as DL3-RR5a), while RFH-FDPI-1 credits no barriers. See section 3.6.1.4 of this SE for additional information on the review of the LBE analyses associated with the PIC additional discussion on the PIC design.

As described in PSAR section 3.8.5.16, RFH-FDRC-1 is initiated by a fuel handling event that occurs when moving fuel in the PRC. This is distinguished from the RFH-OERC family of events because it occurs while moving the full assembly rather than during pin removal operations. A high burnup LTA or LDA is assumed to fail as a result of the fuel handling event. The staff evaluated the source term and functional containment performance assumptions for this event in section 3.2.1.3 of this SE, where they were found to reasonably represent the event. Consistent with the source term evaluation in this SE and the event description in the PSAR, the LBE credits the SR PRC cell barrier but not the FHB boundary. See section 5.4 of this SE for additional information on the PRC cell barrier.

As discussed in the response to RCI-3, these events have mean frequencies less than 1×10^{-4} per year and upper 95th percentile frequencies greater than 5×10^{-7} per year and are appropriately categorized as BDBEs. Several events have 95th percentile frequencies that cross into the DBE region, without associated DBAs discussed in the PSAR. The staff reviewed these LBEs to confirm that the consequences of a DBA associated with the LBE assumptions would be expected to meet the 10 CFR 50.34 dose criteria and the performance of the SR functions is captured by a DBA analyzed in the PSAR. These LBEs are dispositioned as follows.

- RFH-FDSP-2: There is a DBA, RFH-FDSP-CN, which models a fuel drop in the SFP and credits the same barriers as RFH-FDSP-2. If the dose from RFH-FDSP-CN were to be doubled to account for two assembly failures, it would still meet the 10 CFR 50.34 dose criteria. As such, the staff determined that not having a DBA associated with RFH-FDSP-2 is acceptable for the purposes of the PSAR preliminary analysis, because the existing DBA demonstrates the performance of the associated barriers (none) and the results can be reasonably expected to meet the 10 CFR 50.34 dose criteria.
- RFH-ESWR-2: This event credits the NSRST FHB boundary and NHV filtration. As discussed in section 3.6.1 of this SE, the design of SSCs involved in the ESWR analyses have evolved significantly over the course of the review. This includes the addition of multiple preventative controls that are not currently modeled within the PRA. The staff's review of updated analyses for these events, including any associated DBAs, is being deferred to the OL stage based on the understanding that additional preventative controls will move this event to lower frequencies, and its frequency band will no longer extend into the DBE region. If RFH-ESWR-2 remains a BDBE with uncertainty bands that extend into the DBE range at the OL, a DBA should be

appropriately assigned and additional SR controls identified if needed to maintain the consequence results within the 10 CFR 50.34 dose criteria.

- RFH-FDBL-1: This event does not credit any NSRST barriers and credits the SR BLTC barrier and BLTC transfer barrier (DL3-RR3 and DL3-RR3a). RFH-FDBL-2, which assumes these SR barriers fail, does not extend into the DBE region, so the staff has confidence that the reliability of these SSCs is sufficient. A DBA for RFH-FDBL-1 would be expected to produce results approximately consistent with those reported for RFH-ESWR-CN which credits the same barriers, with the results of RFH-FDBL-2 representing an extreme upper bound on possible doses. Both of these LBEs have margin to the 10 CFR 50.34 dose criteria. As such, the staff determined that the safety classifications and performance criteria would not be affected by the inclusion of a DBA for RFH-FDBL-1 at the CP stage. If RFH-FDBL-1 remains a BDBE with uncertainty bands that extend into the DBE range at the OL, a DBA should be appropriately assigned.
- RFH-FDEM-1: This event does not credit any NSRST barriers and credits the SR EVHM cask barrier and EVHM to head barrier (DL3-RR3 and DL3-RR3a). RFH-FDEM-2, which assumes these SR barriers fail, does not extend into the DBE region, so the staff has confidence that the reliability of these SSCs is sufficient. The 95th percentile dose is 2.09×10^{-1} rem TEDE, so there is significant margin to the 10 CFR 50.34 dose criteria. As such, the staff determined that the safety classifications and performance criteria would not be affected by the inclusion of a DBA for RFH-FDEM-1 at the CP stage. If RFH-FDEM-1 remains a BDBE with uncertainty bands that extend into the DBE range at the OL, a DBA should be appropriately assigned.
- RFH-FDPI-BL: This event credits the NSRST PIC barrier as connected to the SR BLTC barrier and BLTC transfer barriers (DL3-RR5a). However, the other event in the family, RFH-FDPI-1, is a risk-significant BDBE that does not credit these barriers or the FHB and provides a 95th percentile dose of 21 rem TEDE at the EAB. As such, the staff determined that the safety classifications and performance criteria would not be affected by the inclusion of a DBA for RFH-FDPI-BL at the CP stage. If RFH-FDPI-BL remains a BDBE with uncertainty bands that extend into the DBE range at the OL, a DBA should be appropriately assigned.
- RFH-FDRC-1: The LBE only credits SR SSCs and does not assume failure of any SR SSCs. The dose has significant margin to the 10 CFR 50.34 dose criterion, so even if it were to be analyzed with more conservative DBA modeling and assumptions, it would be expected to remain below the 25 rem TEDE criterion. As such, the staff determined that the SSC safety classifications and performance criteria would not be affected by the inclusion of a DBA for RFH-FDRC-1 at the CP stage. If RFH-FDRC-1 remains a BDBE with uncertainty bands that extend into the DBE range at the OL, a DBA should be appropriately assigned.

Based on the staff's review described above, the staff concluded that the information provided relative to the fuel handling BDBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each BDBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the BDBEs were appropriately categorized as BDBEs, and that they credited appropriate safety functions consistent with the event description, safety

analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.7.1.6 Local Fuel Fault BDBEs

The staff reviewed PSAR section 3.8.6, which describes the BDBEs initiated by local fuel faults including:

- Core blockage and local faults with non-passive IAC and functional containment barrier fails to retain release (vessel head success) (LFF-SAO-1)
- Core blockage and local faults with non-passive IAC and functional containment barrier successfully retains release (vessel head failed) (LFF-SAO-2)

The initiating event is a partial flow blockage or other local fault that is assumed to result in the failure of the entire fuel assembly. The reactivity control and decay heat removal functions credited for these events are the same as for LFF-SAO-BL, and the same rationale applies as discussed in section 3.6.1.5 of this SE.

As described in PSAR section 3.8.6.1, LFF-SAO-1 assumes that the primary functional containment boundary (the primary coolant boundary) performs its function but the HAA does not and appropriately credits the primary coolant boundary. The source term for LFF-SAO-1 is discussed in more detail in section 3.2.1.1 of this SE, where the staff found it to reasonably represent the event. As discussed in section 5.4.3 of this SE, success of the primary coolant boundary as a functional containment barrier depends on the SR SCG isolation function added during the staff's review, which will be evaluated further at the OL.

As described in PSAR section 3.8.6.2, LFF-SAO-2 is intended to assume that the primary functional containment boundary fails but the HAA is successful, based on the title and description of the LBE in the PSAR, the credited safety functions in PSAR table 5.2-4, the description of the event in NAT-4770, "Natrium Demonstration DID Evaluation Report," Rev. 1 (ML25205A087), and other documentation audited by the staff. However, the description for the mechanistic source term for LFF-SAO-2 in the PSAR states that the primary coolant boundary is intact and the HAA is failed. Further, the staff confirmed this is consistent with the analysis that was performed.

During the audit (ML25302A443), USO determined that this analysis was erroneous (see audit question 3-95), and subsequently developed a preliminary assessment of the dose with the appropriately credited safety functions which indicated that the dose could be slightly higher than what is reported in the PSAR. However, the staff confirmed that, even with the correction, significant margin would still be expected to the F-C target in the BDBE region. Additionally, the staff performed confirmatory analyses which considered various barrier configurations and provided the staff with an understanding of barrier performance in scenarios similar to LFF-SAO-2. Based on the margins reported in the existing analysis and staff confirmatory analyses that staff determined that this event was appropriately categorized as a BDBE for the PSAR but anticipates that the description of the event will be corrected at the OL stage.

As discussed in the response to RCI-3, these events have mean frequencies less than 1×10^{-4} per year and upper 95th percentile frequencies greater than 5×10^{-7} per year and are appropriately categorized as BDBEs. The initiating event frequencies for these events are consistent with the discussion for LFF-SAO-BL in section 3.6.1.5 of this SE.

Based on the staff's review described above, the staff concluded that the information provided relative to the local fuel fault BDBEs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of each BDBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the BDBEs were appropriately categorized as BDBEs, and that they credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.7.1.7 Other BDBEs

The staff reviewed PSAR section 3.8.6, which describes the remaining BDBEs that do not fit into other categories. These include:

- Loss of SFP cooling with makeup restored (RFH-LSPC-1)

As described in PSAR section 3.8.7.1, RFH-LSPC-1 represents a loss of SFP cooling. The normal spent fuel pool cooling system is not restored, but emergency makeup is implemented prior to boiling the SFP. Fuel cladding integrity is maintained. As discussed in section 3.5.1.4 of this SE, this is reasonable to the staff based on the time to boil the SFP. The LBE appropriately credits the SR passive heat removal in the FHP, NSRST SFP emergency makeup, and SR fuel cladding functions.

As discussed in the response to RCI-3, this event has a mean frequency less than 1×10^{-4} per year and upper 95th percentile frequency greater than 5×10^{-7} per year and is appropriately categorized as a BDBE.

Based on the staff's review described above, the staff concluded that the information provided relative to the BDBE described in this section was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the BDBE covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The staff also confirmed that the BDBE was appropriately categorized as a BDBE, and that it credited appropriate safety functions consistent with the event description, safety analysis assumptions, the SSC safety classifications and tables in PSAR chapter 5, and the description of the SSCs in PSAR chapter 7.

3.7.2 Conclusion

The staff finds that the BDBE results meet the requirements in 10 CFR 50.34 and NEI 18-04, as endorsed with clarifications by RG 1.233. The level of detail provided in the applicant's PSAR is consistent with the guidance in NEI 21-07, as endorsed with clarifications in RG 1.253, considering the preliminary nature of the design at the CP stage.

The staff finds the BDBE results to be acceptable and support the safety conclusions. Therefore, the staff finds that the BDBE analyses adequately support the issuance of a construction permit pursuant to the regulations of 10 CFR 50.34(a)(1), (a)(3)(ii), and (4) and 10 CFR 50.35, as applicable.

3.7.3 Tables and Figures

Table 3.7-1: Summary of beyond design basis events from PSAR section 3.8

PSAR Section	Name of the Transient	Alphanumeric Identifier	Type of Transient	Material at Risk	Functions Responsible for Controlling Heat Generation	PSF of Heat Generation Control Functions	Functions Responsible for Providing Decay Heat Removal	PSF of Decay Heat Removal Functions	Functions Responsible for Providing Radionuclide Retention	PSF of Radionuclide Retention Functions
3.8.1.1	Loss of One Primary Sodium Pump with Passive IAC	DHP-L1PP-1	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1	IAC passive mode, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature	DL4-HR1, DL3-HR1, DL3-HR2, DL3-HR3	Fuel cladding	DL3-RR2
3.8.1.2	Loss of One Primary Sodium Pump with Scram Motor Drive-In	DHP-L1PP-3	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio (fails), CRD driveline scram follow	DL4-RC3	IAC non-passive mode	None*	Fuel cladding	DL3-RR2
3.8.1.3	Loss of One Primary Sodium Pump with Alternative Shunt Trip	DHP-L1PP-4	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio (fails), alternative shunt trip	DL3-RC1, DL4-RC6	IAC non-passive mode	None*	Fuel cladding	DL3-RR2
3.8.1.4	Loss of All Primary Sodium Pumps with Non-Passive IAC	DHP-LAPP-BL	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1d	IAC non-passive mode	None*	Fuel cladding	DL3-RR2
3.8.1.5	Loss of Offsite Power with RAC	DHP-LOOP-2	Loss of Offsite Power	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1d	RAC, natural circulation of sodium in primary system	DL3-HR4, DL3-HR5	Fuel cladding	DL3-RR2

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3.8.1.6	Loss of Offsite Power with Scram Motor Drive-In	DHP-LOOP-3	Loss of Offsite Power	Reactor core	Reactor scram on high high power-to-flow ratio (fails), CRD driveline scram follow	DL4-RC3	IAC non-passive mode	None*	Primary coolant boundary, ex-RES functional containment barrier	DL3-RR1, DL4-RR1
3.8.1.7	Loss of Offsite Power with Alternative Shunt Trip	DHP-LOOP-4	Loss of Offsite Power	Reactor core	Reactor scram on high high power-to-flow ratio (fails), alternative shunt trip	DL3-RC1d, DL4-RC6	IAC non-passive mode	None*	Primary coolant boundary, ex-RES functional containment barrier	DL3-RR1, DL4-RR1
3.8.1.8	Loss of a Single Medium Voltage AC Bus with Passive IAC	OTH-LMAC-1	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1, DL3-RC1d	IAC passive mode, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature	DL4-HR1, DL3-HR1, DL3-HR2, DL3-HR3	Fuel cladding	DL3-RR2
3.8.1.9	Loss of a Single Medium Voltage AC Bus with RAC	OTH-LMAC-2	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio	DL3-RC1, DL3-RC1d	RAC, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature, natural circulation of sodium in primary system	DL3-HR4, DL3-HR1, DL3-HR2, DL3-HR3, DL3-HR5	Fuel cladding	DL3-RR2
3.8.1.10	Loss of a Single Medium Voltage AC Bus with Scram Motor Drive-In	OTH-LMAC-3	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio (fails), CRD driveline scram follow	DL4-RC3	IAC non-passive mode	None*	Fuel cladding	DL3-RR2

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3.8.1.11	Loss of a Single Medium Voltage AC Bus with Alternative Shunt Trip	OTH-LMAC-4	Loss of Primary Flow	Reactor core	Reactor scram on high high power-to-flow ratio (fails), alternative shunt trip	DL3-RC1d, DL4-RC6	IAC non-passive mode	None*	Fuel cladding	DL3-RR2
3.8.2.1	Loss of Heat Sink with Scram Motor Drive-In	DHS-ISTL-3	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high cold pool temperature (fails), CRD driveline scram follow	DL4-RC3	IAC non-passive mode	None*	Fuel cladding	DL3-RR2
3.8.2.2	Loss of Heat Sink with Alternative Shunt Trip	DHS-ISTL-4	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high cold pool temperature (fails), alternative shunt trip	DL3-RC1e, DL4-RC6	IAC non-passive mode	None*	Fuel cladding	DL3-RR2
3.8.2.3	Energy Island Transient without Reactor Power Runback, with Passive IAC	DHS-RNBK-2	Increase or Decrease in Heat Removal	Reactor core	Unspecified reactor scram	DL3-RC1	IAC passive mode, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature	DL4-HR1, DL3-HR1, DL3-HR2, DL3-HR3	Fuel cladding	DL3-RR2
3.8.2.4	Loss of One Train of IAC While Shutdown with RAC	SUD-IACA-2	Increase or Decrease in Heat Removal	Reactor core	None	None	RAC, natural circulation of sodium in primary system	DL3-HR4, DL3-HR5	Fuel cladding	DL3-RR2

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3.8.2.5	Intermediate Heat Exchanger Secondary-to-Primary Leak with Passive IAC	IPI-IHEL-1	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high primary sodium level	DL3-RC1c	IAC passive mode, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium level	DL4-HR1, DL3-HR1, DL3-HR2, DL3-HR12	Fuel cladding	DL3-RR2
3.8.2.6	Intermediate Heat Exchanger Secondary-to-Primary Leak with RAC	IPI-IHEL-2	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high primary sodium level	DL3-RC1c	RAC, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium level, natural circulation of sodium in primary system	DL3-HR4, DL3-HR1, DL3-HR2, DL3-HR12, DL3-HR5	Fuel cladding	DL3-RR2
3.8.2.7	Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown with Passive IAC	SUD-IHEL-1	Increase or Decrease in Heat Removal	Reactor core	Reactor scram on high high primary sodium level	DL3-RC1c	IAC passive mode	DL4-HR1	Fuel cladding	DL3-RR2
3.8.3.1	Control Rod-Induced Transient Overpower with Passive IAC	RPD-CW1ACS-1	Reactivity	Reactor core	Reactor scram on high high power range neutron flux or high high power-to-flow ratio	DL3-RC1a or DL3-RC1d	IAC passive mode, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature	DL4-HR1, DL3-HR1, DL3-HR2, DL3-HR3	Fuel cladding	DL3-RR2

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3.8.3.2	Control Rod-Induced Transient Overpower with RAC	RPD-CW1ACS-2	Reactivity	Reactor core	Reactor scram on high high power range neutron flux or high high power-to-flow ratio	DL3-RC1a or DL3-RC1d	RAC, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature, natural circulation of sodium in primary system	DL3-HR4, DL3-HR1, DL3-HR2, DL3-HR3, DL3-HR5	Fuel cladding	DL3-RR2
3.8.3.3	Control Rod-Induced Transient Overpower with Scram Motor Drive-In	RPD-CW1ACS-3	Reactivity	Reactor core	Reactor scram on high high power range neutron flux or high high power-to-flow ratio (fails), CRD driveline scram follow	DL4-RC3	IAC non-passive mode	None**	Fuel cladding	DL3-RR2
3.8.3.4	Control Rod-Induced Transient Overpower with Alternative Shunt Trip	RPD-CW1ACS-4	Reactivity	Reactor core	Unspecified reactor scram (fails), alternative shunt trip on high core exit temperature	DL3-RC1**, DL4-RC6, DL4-RC6a	IAC non-passive mode	None*	Fuel cladding	DL3-RR2

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3.8.3.5	Reactor Scram or Spurious Scram with Passive IAC	RPD-SS-1	Reactivity	Reactor core	Unspecified reactor scram	DL3-RC1	IAC passive mode, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature	DL4-HR1, DL3-HR1, DL3-HR2, DL3-HR3	Fuel cladding	DL3-RR2
3.8.3.6	Reactor Scram or Spurious Scram with RAC	RPD-SS-2	Reactivity	Reactor core	Unspecified reactor scram	DL3-RC1	RAC, PSP coastdown, PSP trip on high high primary sodium temperature, ISP trip on high high primary sodium temperature, natural circulation of sodium in primary system	DL3-HR4, DL3-HR1, DL3-HR2, DL3-HR3, DL3-HR5	Fuel cladding	DL3-RR2
3.8.3.7	Reactor Scram or Spurious Scram with Scram Motor Drive-In	RPD-SS-3	Reactivity	Reactor core	Unspecified reactor scram (fails), CRD driveline scram follow	DL4-RC3	IAC non-passive mode	None*	Fuel cladding	DL3-RR2

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3.8.3.8	Reactor Scram or Spurious Scram with Alternative Shunt Trip	RPD-SS-4	Reactivity	Reactor core	Unspecified reactor scram (fails), alternative shunt trip	DL3-RC1**, DL4-RC6	IAC non-passive mode	None*	Fuel cladding	DL3-RR2
3.8.4.1	SCG Leak Inside the HAA where RXB Substructure Fails to Retain Release While Shutdown	SUD-CGR-1	Release from Ex-Vessel Systems	SCG inventory	N/A	N/A	N/A	N/A	None	None
3.8.4.2	SCG Leak Downstream the SCG Cell where RXB Substructure Fails to Retain Release While Shutdown	SUD-CGR-2	Release from Ex-Vessel Systems	SCG inventory	N/A	N/A	N/A	N/A	None	None
3.8.5.1	Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with a Single Assembly Failed and Functional Containment Barriers Fail to Retain Release	RFH-FDIV-2	Fuel Handling	Two high burnup fuel assemblies	N/A	N/A	N/A	N/A	None	None

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3.8.5.2	Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with Two Assemblies Failed and Functional Containment Barriers Fails to Retain Release	RFH-FDIV-4	Fuel Handling	Two high burnup fuel assemblies	N/A	N/A	N/A	N/A	None	None
3.8.5.3	Fuel Handling Event Occurs While Moving Fuel Assembly in the SFP with Two Assemblies Failed and FHB Barrier Fails to Retain Release	RFH-FDSP-2	Fuel Handling	Two high burnup fuel assemblies, 310 day decay	N/A	N/A	N/A	N/A	None	None
3.8.5.4	Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Fails to Retain Release (FHB Credited)	RFH-ESWR-2	Fuel Handling	Single high burnup fuel assembly	N/A	N/A	N/A	N/A	Fuel handling building barrier	DL4-RR7
3.8.5.5	Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Fails to Retain Release (FHB Not Credited)	RFH-ESWR-3	Fuel Handling	Single high burnup fuel assembly	N/A	N/A	N/A	N/A	N/A	N/A

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3.8.5.6	Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA without Passive Cooling, EVHM Barrier Successfully Retains Release	RFH-LMCA-1	Fuel Handling	Single high burnup LTA or LDA, 2.5 day decay	N/A	N/A	None	None	EVHM cask barrier, EVHM to head barrier function	DL3-RR2, DL3-RR3
3.8.5.7	Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA without Passive Cooling, EVHM Barrier Fails to Retain Release	RFH-LMCA-2	Fuel Handling	Single high burnup LTA or LDA, 2.5 day decay	N/A	N/A	N/A	N/A	None	None
3.8.5.8	Fuel Handling Event Occurs While Moving Fuel Assembly in the BLTC with BLTC Barrier Successfully Retains Release	RFH-FDBL-1	Fuel Handling	Single high burnup fuel assembly, 210 day decay	N/A	N/A	N/A	N/A	BLTC barrier, BLTC transfer barrier	DL3-RR5, DL3-RR5a
3.8.5.9	Fuel Handling Event Occurs While Moving Fuel Assembly in the BLTC with BLTC Barrier Fails to Retain Release	RFH-FDBL-2	Fuel Handling	Single high burnup fuel assembly, 210 day decay	N/A	N/A	N/A	N/A	None	None

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3.8.5.10	Fuel Handling Event Occurs While Moving Fuel Assembly or an LTA or LDA in the EVHM with EVHM Barrier Successfully Retains Release	RFH-FDEM-1	Fuel Handling	High burnup LTA or LDA, 2.5 day decay	N/A	N/A	N/A	N/A	EVHM cask barrier, EVHM to head barrier function	DL3-RR3, DL3-RR3a
3.8.5.11	Fuel Handling Event Occurs While Moving Fuel Assembly or an LTA or LDA in the EVHM with EVHM Barrier Fails to Retain Release	RFH-FDEM-2	Fuel Handling	High burnup LTA or LDA, 2.5 day decay	N/A	N/A	N/A	N/A	None	None
3.8.5.12	Fuel Handling Event Occurs While Moving Fuel Assembly in the EVST with No Damage	RFH-FDET-BL	Fuel Handling	Single fuel assembly	N/A	N/A	N/A	N/A	Fuel cladding	DL3-RR2
3.8.5.13	Fuel Handling Event Occurs While Moving Fuel Assembly in the EVST with a Single Assembly Failed and EVST Barrier Successfully Retains Release	RFH-FDET-1	Fuel Handling	Single high burnup fuel assembly, 210 day decay	N/A	N/A	N/A	N/A	EVST barrier	DL3-RR4

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3.8.5.14	Fuel Handling Event Occurs While Moving Fuel Assembly in the PIC with PIC Barrier Successfully Retains Release	RFH-FDPI-BL	Fuel Handling	Single high burnup fuel assembly, 210 day decay	N/A	N/A	N/A	N/A	PIC radionuclide boundary, BLTC barrier, BLTC transfer barrier	DL3-RR5a
3.8.5.15	Fuel Handling Event Occurs While Moving Fuel Assembly in the PIC with PIC Barrier Fails to Retain Release	RFH-FDPI-1	Fuel Handling	Single high burnup fuel assembly, 210 day decay	N/A	N/A	N/A	N/A	None	None
3.8.5.16	Fuel Handling Event Occurs While Moving Fuel Assembly in the PRC with PRC Barrier Successfully Retains Release	RFH-FDRC-1	Fuel Handling	Single high burnup LTA or LDA	N/A	N/A	N/A	N/A	PRC cell barrier	DL3-RR6
3.8.6.1	Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Fails to Retain Release (Vessel Head Success)	LFF-SAO-1	Local Fuel Fault	Reactor core	Manual reactor scram	None*	IAC non-passive mode	None*	Primary coolant boundary	DL3-RR1

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3.8.6.2	Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Successfully Retains Release (Vessel Head Failed)	LFF-SAO-2	Local Fuel Fault	Reactor core	Manual reactor scram	None*	IAC non-passive mode	None*	Ex-RES functional containment barrier, HAA barrier	DL4-RR1, DL4-RR1a
3.8.7.1	Loss of SFP Cooling with Makeup Restored	RFH-LSPC-1	Other	SFP contents	N/A	N/A	Passive heat removal in FHP, SFP emergency makeup	DL3-HR10, DL4-HR4	Fuel cladding	DL3-RC2

* No safety-significant PSF is identified for these functions in the tables in PSAR chapter 5 because they are NST.

** For this event, the failed function is the RPS signal to trip the breakers, but the control rod insertion by gravity succeeds.

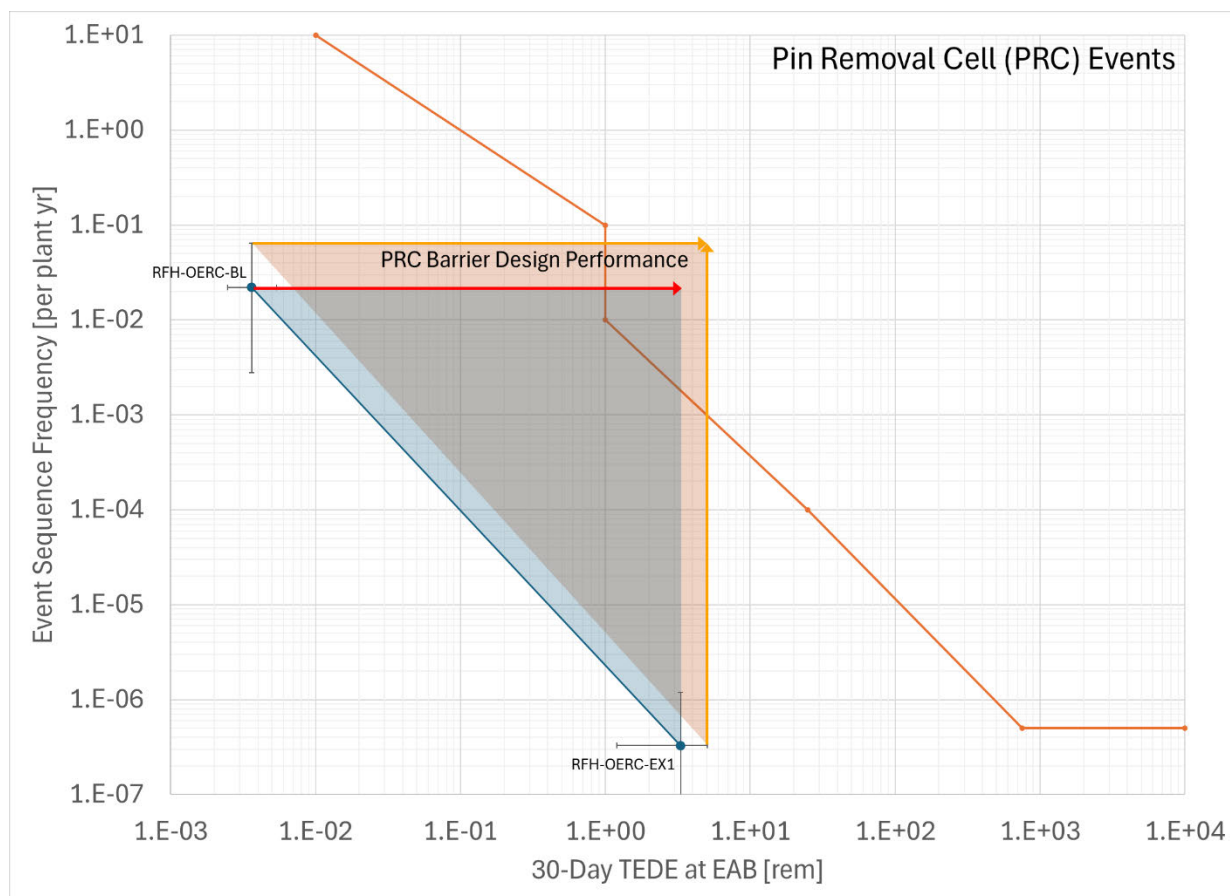


Figure 3.7-1: RFH-OERC Initiating Event Family

3.8 Design Basis Accidents

PSAR section 3.9 describes the analysis of DBAs selected through use of methodologies described in PSAR sections 3.3 and 3.4 for inclusion in the plant's licensing basis.

The applicable regulatory requirements and guidance for the selection of DBAs are identical to those described in section 3.6 of this SE.

3.8.1 Technical Evaluation

The staff reviewed PSAR section 3.9 to determine that the DBA selection is consistent with the process described in NEI 18-04 and endorsed, with certain clarifications, in RG 1.233; and described in the PSAR, consistent the guidance in NEI 21-07 as endorsed with clarifications by RG 1.253; or if deviations were used that the applicant provided a reasonable basis. The staff also reviewed the DBA initial conditions and modeling parameters to ensure conservatism and the use of acceptable analytical approaches for DBAs consistent with the licensing methodologies discussed in PSAR section 3.3.1 and section 3.3 of this SE, considering the preliminary nature of the design at the CP stage. The staff also identified the DBE associated with each DBA and identified and dispositioned any significant differences between the DBE and DBA. As discussed in section 3.3.1.6 of this SE, the DBEs typically use a similar conservative approach to analysis as the DBAs. However, as the staff confirmed in audit, while the DBE assumes nominal performance from non-SR¹⁴ SSCs based on the PRA event sequence (i.e., SSC availability due to successes and failures), the DBA generally assumes non-SR SSCs are biased in the direction that makes the event progression worse. This is consistent with the DBA methodologies described in section 3.3.1 of this SE and the staff's expectations regarding performance of non-SR SSCs in DBAs.

The staff identified seventeen AOOs and BDBEs that extend into the DBE region. Of these, four have DBAs for the LBE family in the PSAR that appropriately bound them, eight do not have relevant DBAs in the PSAR but have sufficient SR controls within the event to meet the 10 CFR 50.34 dose criteria. The AOOs and BDBEs with uncertainty ranges that extend into the DBE region are dispositioned in sections 3.5 and 3.7 of this SE, respectively.

3.8.1.1 Loss of Primary Flow DBAs

The staff reviewed PSAR section 3.9.1, which describes the loss of primary flow DBAs, which including:

- Loss of one primary sodium pump (DHP-L1PP-CN)
- Loss of offsite power (DHP-LOOP-CN)
- Loss of offsite power while at low power (SUD-LOOP-CN)

The safety concern for these DBAs is an imbalance between heat generation in the reactor core and heat removal capability through the primary coolant.

¹⁴ The term "non-SR" without any further text describing special treatments, when used in this SE, includes any SSCs that are not safety-related, including both NSRST and NST SSCs.

DHP-L1PP-CN is described in PSAR section 3.9.1.1, and is the DBA associated with DBE DHP-L1PP-2. The event sequences match and only SR SSCs are credited in both analyses. DHP-L1PP-CN also bounds DHP-L1PP-1, which is a DBE with a 95th percentile frequency extending into the DBE region. DHP-L1PP-1 has the same initiating sequence, but heat removal is provided by passive IAC. In evaluating the DBA for this event, the NSRST passive IAC is assumed to fail, resulting in the sequence evaluated for the DHP-L1PP-2 and DHP-L1PP-CN. The staff noted that the mean frequency of DHP-L1PP-1 being lower than DHP-L1PP-2, which implies that the passive IAC is more likely to fail in this IE family than it is to succeed. Since the staff did not review the fault trees, the specific reason for this is unclear to the staff, but it may be due to common cause failures that cause the loss of active IAC. Regardless, heat removal from RAC is sufficient to ensure fuel cladding integrity and there are no radionuclide releases from these events, as the staff confirmed by audit of the supporting analyses.

DHP-LOOP-CN and SUD-LOOP-CN are described in PSAR sections 3.9.1.2 and 3.9.1.3, and are the DBAs associated with DHP-LOOP-1 and SUD-LOOP-1, respectively. These DBAs also bound AOOs DHP-LOOP-BL and SUD-LOOP-BL, which have 5th percentile frequencies that extend into the DBE region. The associated AOO sequences rely on active IAC and the standby diesel generators, which are both assumed to fail in the DBA analysis. DHP-LOOP-1 and SUD-LOOP-1 rely on passive IAC which are also assumed to fail for the DBA analysis. With the failure of these NST and NSRST systems, decay heat removal is provided by RAC. RAC is credited in the analyses, as is PSP coastdown and primary coolant natural circulation. The heat removal from RAC is sufficient to ensure fuel cladding integrity and there are no radionuclide releases, as the staff confirmed by audit of the supporting analyses.

For the three scenarios described in PSAR section 3.9.1, the staff confirmed in audit that the analyses assume bounding parameters consistent with the licensing methodology as described in PSAR section 3.3.1. The staff notes that several important parameters in the PSAR analysis are preliminary. For example, the PSP coastdown curve is based on design pump characteristics. While the assumptions about the PSP appear to the staff to be conservative relative to the design, the staff notes that additional work is necessary to confirm the performance of the PSPs as the design matures. RAC performance is also significant to the plant response, and the staff confirmed in audit that RAC was modeled appropriately, consistent with TR NAT-9390-A. However, RAC performance depends on aspects that are subjects of verification and validation efforts as described in this topical report or are the subject of research and development efforts as described in PSAR chapter 13. The staff confirmed that analytical assumptions that are the subject of ongoing design, analysis, or testing were generally reasonable or conservative, and the staff thus concluded that the DBAs described in this section were reasonable for preliminary analyses.

Based on the staff's review described above, the staff concluded that the information provided relative to the loss of primary flow DBAs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the DBA covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The analysis demonstrated that the 25 rem TEDE dose criterion from 10 CFR 50.34 is met by showing no fuel is failed as a result of the DBAs discussed in this section.

The staff confirmed through the review that the DBAs credited appropriate SR safety functions consistent with the event description, safety analysis assumptions, and the description of the SSCs in PSAR chapter 7. The staff also identified that further technical or design information is needed as described above. Because reasonable or conservative assumptions were made relative to these parameters, and because further design, analysis, and testing are needed to

refine the assumptions, the staff identified that these issues can be reasonably left for later consideration in the review of the FSAR at the OL stage.

3.8.1.2 Increase or Decrease in Heat Removal DBAs

The staff reviewed PSAR section 3.9.2, which describes the increase or decrease in heat removal DBEs, including:

- Loss of heat sink (DHS-ISTL-CN)
- Energy island transient (DHS-RNBK-CN)
- Loss of one train of IAC while shutdown (SUD-IACA-CN)
- IHX tube break (IPI-IHEL-CN)
- Intermediate heat exchanger secondary to primary leak while shutdown (SUD-IHEL-CN)

The concern for these LBEs is an imbalance between heat generation in the core and heat removal to the ultimate heat sink.

DHS-ISTL-CN is described in PSAR section 3.9.2.1 and is the DBA associated with DBEs DHS-ISTL-1 and DHS-ISTL-2. DHS-ISTL-1 relies on one train of passive IAC, which is assumed to fail in the DBA since it is a non-SR system. However, the thermal capacity of the remaining loop is assumed to be available, as noted in the PSAR. The DBA event sequence matches DHS-ISTL-2, only crediting heat removal from the SR RAC system, with credit for SR PSP and ISP trips to limit heat addition to the coolant, consistent with the discussion provided in SE section 3.6.1.1. RAC is sufficient to ensure fuel integrity, as confirmed by the staff through audit of the supporting calculations.

DHS-RNBK-CN is described in PSAR section 3.9.2.2 and is the DBA associated with DBEs DHS-RNBK-1 and DHS-RNBK-3. DHS-RNBK-CN also bounds DHS-RNBK-2, which is a DBE with a 95th percentile frequency extending into the DBE region. The event sequence for DHS-RNBK-CN and DHS-RNBK-3 are the same. DHS-RNBK-1 and DHS-RNBK-2 have the same initial sequence, but heat removal is provided by non-passive and passive IAC, respectively. In the applicant's evaluation of the DBA, the IAC functions are assumed to fail because they are non-SR functions, resulting in the same event sequence evaluated for DHS-RNBK-3. Note, similar to the DHP-L1PP IE family as described in section 3.8.1.1 of this SE, the mean frequency of DHS-RNBK-2 is lower than DHS-RNBK-3, which implies to the staff that the passive IAC is more likely to fail in this IE family than it is to succeed. Decay heat removal is provided by the SR RAC function, with appropriate credit for SR PSP and ISP trips, PSP coastdown, and primary system natural circulation. RAC heat removal is sufficient to ensure fuel cladding integrity is maintained and there is no radionuclide release, as confirmed by the staff through audit of the supporting calculations.

SUD-IACA-CN is described in PSAR section 3.9.2.3 and is the DBA associated with DBE SUD-IACA-1. SUD-IACA-1 relies on IAC in passive mode for heat removal, which is assumed to fail in the DBA analysis because it is a non-SR function. SUD-IACA-CN credits heat removal from RAC, but the SR PSP trip is not credited. This is consistent with the analysis audited by the staff, but the staff notes that []

]] RAC heat removal is sufficient to ensure fuel cladding integrity is maintained and there is no radionuclide release, as confirmed by the staff through audit of the supporting calculations.

IPI-IHEL-CN and SUD-IHEL-CN are described in PSAR sections 3.9.2.4 and 3.9.2.5, and are the DBAs associated with IPI-IHEL-BL and SUD-IHEL-BL, respectively. Both DBAs assume failure of the non-SR passive IAC relied on for heat removal in DBEs, though the thermal capacity of IHT in the intact loop is assumed to be available, as discussed in the PSAR. RAC is credited for sufficient decay heat removal to ensure fuel cladding integrity is maintained.

For the scenarios described in PSAR section 3.9.2, the staff confirmed in audit that the analyses assume bounding parameters consistent with the licensing methodology as described in PSAR section 3.3.1. Consistent with the discussion in section 3.8.1.1 of this SE, the staff noted the analysis is preliminary and thus contains important assumptions that will be verified as the design matures, analytical methods are verified and validated, and research and development activities are conducted. The staff determined that the analytical assumptions that are the subject of ongoing design, analysis, or testing were generally reasonable or conservative, and the staff thus concluded that the DBAs described in this section were reasonable for preliminary analyses.

Based on the staff's review described above, the staff concluded that the information provided relative to the increase or decrease in heat removal DBAs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the DBA covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The analysis demonstrated that the 25 rem TEDE dose criterion from 10 CFR 50.34 is met by showing no fuel is failed as a result of the DBAs discussed in this section.

The staff determined through the review that the DBAs credited appropriate SR safety functions consistent with the event description, safety analysis assumptions, and the description of the SSCs in PSAR chapter 7. The staff also identified that further technical or design information is needed as described above. Because the staff determined that reasonable or conservative assumptions were made relative to these parameters, and because further design, analysis, and testing are needed to refine the assumptions, the staff identified that these issues can be reasonably left for later consideration in the review of the FSAR.

3.8.1.3 Reactivity DBAs

The staff reviewed PSAR section 3.9.3, which includes the one reactivity DBA (SEV-SSE-CN) intended to represent a seismic event. There are no DBEs related to this DBA, which was added to address design basis hazards associated with a seismic event at the CP stage as discussed in PSAR section 3.4.

SEV-SSE-CN, discussed in PSAR section 3.9.3.1 is initiated by a safe shutdown earthquake (SSE) during full power operation. Core compaction caused by the SSE results in a reactivity insertion. The applicant indicated in PSAR section 3.9.3.1.1 that a range of reactivity profiles were examined to identify a limiting reactivity insertion for the event, which was found to be approximately 30 cents. This is a significant reactivity insertion, but small enough that the reactor kinetics would still be expected to behave like a light water reactor (Waltar and Reynolds, 1980) and within the range of the models used. This reactivity insertion leads to a reactor scram on high high neutron flux rate or high high power-to-flow ratio, both of which the

staff would expect to actuate in this scenario due to the increase in power and neutron flux because of the reactivity insertion.

After the scram, control rods are assumed to drop at a degraded rate. This is because the control rods move up and down within their own ducts within the core, as described in section 7.1.1.3.1 of the PSAR, and relative motion of the control rod assembly and duct can cause friction or binding. The purpose of the secondary control rods, also described in section 7.1.1.3.1 of the PSAR and in NAT-2806-A, is to mitigate the failure of control rods to insert in such a scenario. The staff identified based on the design of the secondary control rods that though they may not fail to insert, they could insert more slowly. As such, the staff considers the use of a degraded control rod insertion rate in the analysis to be acceptable.

The PSPs and ISPs are assumed to trip in the event due to either a loss of offsite power or high high primary sodium temperature caused by the power excursion. Because of a loss of offsite power would be expected to occur during an SSE and sodium temperature is calculated to rise substantially as part of the event, the staff found this assumption to be reasonable. The PSPs coast down after the trip consistent with their SR function. Decay heat removal is provided by RAC and is sufficient to ensure fuel cladding integrity is maintained.

The staff confirmed in audit that the SEV-SSE-CN analyses assume bounding parameters consistent with the licensing methodology as described in PSAR section 3.3.1. In particular, the staff identified that a range of reactivity profiles were examined to identify a limiting reactivity insertion. These profiles were based on the vertical and horizontal motions consistent with the SSE (assuming seismic isolators for the reactor perform their safety functions), and the resulting motions of the core assemblies. The staff confirmed in audit that a reactivity insertion bounding the examined profiles was included in the analysis. The staff also confirmed that an appropriate degraded control rod insertion curve was assumed.

Consistent with the discussion in section 3.8.1.1 of this SE, the staff noted the preliminary analysis contains important assumptions that will be verified as the design matures, analytical methods are verified and validated, and research and development activities are conducted. The staff determined that the analytical assumptions that are the subject of ongoing design, analysis, or testing were generally reasonable or conservative, and the staff thus concluded that the DBAs described in this section were reasonable for preliminary analyses.

Based on the staff's review described above, the staff concluded that the information provided relative to the reactivity DBA was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the DBA covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The analysis demonstrated that the 25 rem TEDE dose criterion from 10 CFR 50.34 is met by showing no fuel is failed as a result of the DBA discussed in this section.

The staff determined through the review that the DBA credited appropriate SR safety functions consistent with the event description, safety analysis assumptions, and the description of the SSCs in PSAR chapter 7. The staff also identified that further technical or design information is needed as described above. Because the staff determined that reasonable or conservative assumptions were made relative to these parameters, and because further design, analysis, and testing are needed to refine the assumptions, the staff identified that these issues can be reasonably left for later consideration in the review of the FSAR.

3.8.1.4 Release from Ex-Vessel Systems DBAs

The staff reviewed PSAR section 3.9.4, which of the PSAR describes the DBAs that involve a release from ex-vessel systems, including:

- SCG leak downstream the SCG cell (RRS-CGR-CN)
- SPS-I leak at the cold trap (RRS-ISPL-CN)
- SPS-P leak in the RXB (RRS-SPLX-CN)
- SPS-P leak in the RAB (RRS-SPLA-CN)
- RWG leak (RRS-RWG-CN)

These DBAs are postulated for systems that contain radionuclides, either from circulating radionuclides from failed fuel during operation or activated primary sodium, which have the potential to be released to the environment. Because the DBAs do not involve the potential for releases from the core, no reactor scram functions are needed or credited.

RRS-CGR-CN, which is described in PSAR section 3.9.4.1, is the DBA associated with DBEs RRS-CGR-BL and RRS-CGR-1. These DBEs, as discussed in section 3.6.1.3 of this SE, are releases of the CGR inventory into different functional containments (HAA and VTC), but for the CP stage were represented with a single mechanistic source term analysis that evaluated release into the HAA. The HAA and the VTC are NSRST confinements. For the DBA analysis, it was assumed that the HAA was not isolated (NSRST function) and that the NHV continued to operate. The full SCG inventory is released to the HAA and then exhausted in 2 hours from the HAA by NHV with no credit for filtration. The staff's evaluation of the source term and functional containment performance is described in section 3.2.1.4 of this SE. As discussed in section 5.1.3 of this SE, the staff expects to review justification for the level of degraded performance at the OL stage. In addition, because these DBEs are releasing material into different functional containments, mechanistic source term analyses specific to each DBE and associated DBA should be provided at the OL or justification for representing both with a single analysis should be provided.

RRS-ISPL-CN, which is described in PSAR section 3.9.4.2, is the DBA associated with DBE RRS-ISPL-BL. As discussed in section 3.6.1.3 of this SE, even with bounding assumptions and conservative analysis methods, consequences from an unmitigated release of all tritium in the primary system would not exceed dose targets at the EAB. With these LBEs postulating a smaller inventory of tritium being released from the intermediate cold trap, no SR SSCs are needed for mitigation to remain below the 10 CFR 50.34 dose criteria.

RRS-SPLX-CN and RRS-SPLA-CN, which are described in PSAR sections 3.9.4.3 and 3.9.4.4, are the DBAs associated with DBEs RRS-SPLX-BL and RRS-SPLA-BL, respectively. The staff found that the event assumptions on the volume of leakage and location of the leaks are consistent between the DBE and DBA pairs. Like the DBEs, both DBAs are analyzed based on a leakage into the SPS cell. The NSRST SPS cell is assumed to leak at a degraded rate of 50 percent per day versus 10 percent assumed for the DBEs. Additional discussion on the staff's evaluation of the source term associated with these events is provided in section 3.2.1.4 of this SE. The staff expects these events to be very sensitive to assumptions regarding holdup in the

SPS cell because of the dominant contribution of sodium-24, as shown in PSAR table 3.2-11, which has a short half-life. Consistent with the discussion in section 5.1 of this SE, it may be reasonable in the context of the NEI 18-04 process to assume some degraded performance of passive, non-SR SSCs that are not affected by the initiating event. However, the applicant provided no basis for the assumption of 50 percent leakage per day, and as such the staff cannot reach a determination as to its acceptability. As discussed in section 5.1.3 and 5.5.3 of this SE, USO the staff expects to review bases for the nominal and degraded performances of radionuclide retention barriers used in the source term analyses at the OL stage. However, given the level of design information available at the CP stage and the relatively low consequences of the event (even in consideration of potentially non-conservative assumptions regarding SPS cell leakage), the staff finds the preliminary analysis to be reasonable and will address detailed performance in the final analyses at the OL. In addition, because these DBAs are releasing material into different functional containments, mechanistic source term analyses specific to each DBE and associated DBA should be provided at the OL or justification for representing both with a single analysis should be provided.

RRS-RWG-CN, which is described in PSAR section 3.9.4.5, is the DBA associated with DBEs RRS-RWG-1 and RRS-RWG-2. RRS-RWG-CN involves a leak from the RWG holdup tank directly out to the vent stack over a 10 minute period, which matches the timing assumption in RRS-RWG-2 and bounds the timing assumed for RRS-RWG-1. RRS-RWG-1 and RRS-RWG-2 both relied on non-SR filtration that is not credited in RRS-RWG-CN. The staff's evaluation of the source term and functional containment performance is described in section 3.2.1.4 of this SE. Because DBEs associated with this DBA at the CP are releasing material into different functional containments, mechanistic source term analyses specific to each DBE and associated DBA should be provided at the OL or justification for representing both with a single analysis should be provided.

Based on the staff's review described above, the staff concluded that the information provided relative to the release from ex-vessel systems DBAs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the DBAs covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The PSAR also described the mechanistic source term and associated assumptions, and demonstrated that the 25 rem TEDE dose criterion from 10 CFR 50.34 is met for all the DBAs described in this section.

The staff determined through the review that the DBAs credited appropriate SR safety functions consistent with the event description, safety analysis assumptions, and the description of the SSCs in PSAR chapter 7. Degraded performance of NSRST SSCs is credited for RRS-SPLX-CN and RRS-SPLA-CN, as discussed above. Further technical or design information is needed as described to justify the performance of these SSCs, but the staff identified above that these issues can be reasonably left for later consideration in the review of the FSAR.

3.8.1.5 Fuel Handling DBAs

The staff reviewed PSAR section 3.9.5, which describes the DBAs that are initiated by fuel handling events, including:

- Fuel handling event in reactor vessel (RFH-FDIV-CN)
- Fuel handling event in spent fuel pool (RFH-FDSP-CN)

- Excessive sodium-water reaction in the PIC (RFH-ESWR-CN)
- Loss of EVST active cooling (RFH-LTCA-CN)
- Loss of BLTC active cooling (RFH-LBCA-CN)
- Loss of EVHM active cooling (RFH-LMCA-CN)

The safety concern addressed by these LBEs is that a fuel handling event could result in fuel damage and release of radionuclides to the environment.

RFH-FDIV-CN, which is described in PSAR section 3.9.5.1, is the DBA associated with DBEs RFH-FDIV-BL, -1, and -3. RFH-FDIV-CN matches the event sequence of RFH-FDIV-3, which bounds RFH-FDIV-BL and RFH-FDIV-1 by assuming all pins in two high burnup assemblies are released rather than none or one, respectively. All three LBEs with releases credit the SR radionuclide retention barrier made up of the RV, the core assembly transfer tube, and the EVHM for a leakage rate of 1 percent per day, which is then released directly to the environment. As discussed in the DBE evaluation in section 3.6.1.4 of this SE, performance of the primary functional containment is dependent on the SR SCG isolation function, which would also be needed to mitigate this DBA. The staff's review of the source term associated with this event is discussed in section 3.2.1.2 of this SE, where the staff found it to be reasonable. The staff noted based on the mean frequencies of the associated DBEs that the drop event is expected to result in a single assembly releasing half the time and has a one in four chance of either no release or two assemblies releasing. The staff determined that these are reasonable assumptions to make at the CP stage, but additional justification should be provided at the OL.

RFH-FDSP-CN, which is described in PSAR section 3.9.5.2, is the DBA associated with DBE RFH-FDSP-1. The source term credits none of the NSRST FHB barriers, consistent with RFH-FDSP-1. The main difference between RFH-FDSP-CN and RFH-FDSP-1 is that the DBA analysis uses a different code with slightly different modeling assumptions and enveloping atmospheric dispersion factors, as described in PSAR section 3.3.1.4. The staff's review of the source term associated with this event is discussed in section 3.2.1.2 of this SE, where the staff found it to be reasonable. RFH-FDSP-CN also provides insight into what the DBA dose for RFH-FDSP-2, a BDBE with a 95th percentile frequency extending into the DBA region, would be. RFH-FDSP-2 has the same assumptions, but with two assemblies releasing, rather than one. Doubling the dose from RFH-FDSP-CN results in a TEDE of 14.66 rem for both the 2-hour EAB dose and the 30-day LPZ dose.

RFH-ESWR-CN, which is described in PSAR section 3.9.5.3, is the DBA associated with DBEs RFH-ESWR-BL and RFH-ESWR-1. As discussed in section 3.2.1.2 of this SE, the source terms for the ESWR cases were generated by scaling EVHM and SFP source terms. The SFP source term was used as the source term for RFH-ESWR-1, treating the PIC leakage as equivalent to the SFP scrubbing. For RFH-ESWR-2, a BDBE with a 95th percentile frequency extending into the DBE region, the PIC barrier is assumed to have failed, but the FHB filters are credited with a factor of **[[]]** reduction in source term. As can be seen by comparing the RFH-FDSP-1 and the RFH-ESWR-2 consequences, the RFH-ESWR-2 source term differs from the SFP source term by a factor of 10, which is consistent with credit for the FHB filters and a factor of **[[]]** times reduction from pool scrubbing that is not present in RFH-ESWR-2. Unlike the DBEs, the source term for RFH-ESWR-CN is based off of the EVHM source term which appears to the

staff to have resulted in a lower release fraction from the fuel. The staff estimated that use of the RFH-ESWR-1 source term with the DBA consequence methodology would result in a TEDE of approximately 7.3 rem versus the 0.609 rem reported for this DBA, which is higher but still within the 25 rem TEDE acceptance criterion and thus acceptable. As discussed in sections 3.2.1.2 and 3.6.1.4 of this SE and the PIC supplement, the source term methodology for ESWR events will be further developed prior to the OL and mechanistic source terms will be developed for each of the LBEs. Based on the statements from the applicant in the PIC supplement and the design development to reduce the frequency of these events, the staff determined that further review can be reasonably left for the OL stage. Note, because RFH-ESWR-2 assumes failure of the radionuclide barrier, it is not bounded by this DBA.

RFH-LTCA-CN, which is described in PSAR section 3.9.5.5, is the DBA associated with DBE RFH-LTCA-1. The event sequences for these LBEs match and relate to a loss of all cooling in the EVST. No fuel damage is expected for at least 72 hours following a loss of cooling. Because 72 hours is the typical time before recovery actions are assumed for LBEs, the staff finds the result to be reasonable.

RFH-LBCA-CN and RFH-LMCA-CN, which are described in PSAR sections 3.9.5.6 and 3.9.5.7, are the DBAs associated with DBEs RFH-LBCA-BL and RFH-LMCA-BL, respectively. These LBEs relate to loss of active cooling on BLTC and EVHM, respectively. The event sequences for the DBAs match the associated DBEs, and the associated passive cooling systems are credited to function, providing sufficient heat removal to maintain fuel cladding integrity. Consistent with the discussion in section 3.6.1.4 of this SE, the passive heat removal rate is sufficient to provide adequate decay heat removal for the minimum decay times that are reasonably expected (as discussed in section 7.3.2.1.7 of this SE). Therefore, the staff finds the result to be reasonable for these DBAs.

Based on the staff's review described above, the staff concluded that the information provided relative to the fuel handling DBAs was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the DBAs covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The PSAR also described the mechanistic source term and associated assumptions for events in this section that result in releases, and demonstrated that the 25 rem TEDE dose criterion from 10 CFR 50.34 is met for all the DBAs described in this section.

The staff determined through the review that the DBAs credited appropriate SR safety functions consistent with the event description, safety analysis assumptions, and the description of the SSCs in PSAR chapter 7. The staff also identified that further technical or design information is needed as described above. Because of the preventative controls added to reduce the frequency of ESWR events, statements from USO in the PIC supplement that event specific mechanistic source terms that incorporate methodology updates addressing ESWR, SFP scrubbing, and sodium scrubbing will be developed, and with the addition of the SR SCG isolation function to provide a means of achieving the primary functional containment barrier performance credited, the staff determined that these issues can be reasonably left for later consideration in the review of the FSAR.

3.8.1.6 Local Fuel Fault DBAs

The staff reviewed PSAR section 3.9.6, which describes the DBE initiated by local fuel faults. Local faults include partial flow blockage and fuel misloads as discussed in section 2.4 of NAT-9394-A. The one DBE in this section is:

- Core blockage and local faults (LFF-SAO-CN)

The initiating event affects a single fuel assembly, which is assumed to fail in the analysis as discussed in PSAR section 3.3.4.1.5.

The discussion in section 3.6.1.5 of this SE for the DBE LFF-SAO-BL is also relevant to the DBA analysis. The main difference is in the source term analysis, where the DBA assumes the HAA leaks at a rate of 100 percent per day directly to the environment with no credit for holdup or decay in the RXB. The staff evaluated the source term and functional containment performance assumptions in section 3.2.1.1 of this SE, where they were found to be reasonable for this event when the SR SCG isolation is added to ensure the primary functional containment barrier can achieve its design performance level.

Based on the staff's review described above, the staff concluded that the information provided relative to the local fault DBA was consistent with the guidance provided in NEI 21-07 in that the PSAR provided an adequate description of the DBA covering the initial condition, radionuclide source, initiating event, and characterization of the responses of SSCs performing PSFs. The PSAR also described the mechanistic source term and associated assumptions, and demonstrated that the 25 rem TEDE dose criterion from 10 CFR 50.34 is met.

The staff determined through the review that the DBA credited appropriate SR safety functions consistent with the event description, safety analysis assumptions, and the description of the SSCs in PSAR chapter 7. The staff expects to review further information related to the validation of the partial flow blockage methodology, as discussed in section 3.3.1.3 of this SE. Because full validation of methodologies is not expected at the CP stage, and because the analysis conservatively bounds the consequences of local faults by assuming the full assembly fails, the staff identified that these issues can be reasonably left for later consideration in the review of the FSAR.

3.8.2 Conclusion

The staff determined that the DBAs were properly identified and analyzed using conservative deterministic safety analysis. The staff finds that the DBA results meet the requirements in 10 CFR 50.34. The level of detail provided in the applicant's PSAR is consistent with the guidance in NEI 21-07, as endorsed with clarifications in RG 1.253, considering the preliminary nature of the design at the CP stage.

The staff finds the DBA results to be acceptable and support the safety conclusions. Therefore, the staff finds that the DBA analyses adequately support the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1), (a)(3)(ii), and (4) and 10 CFR 50.35, as applicable.

3.9 Mitigation of Beyond-Design-Basis Events under 10 CFR 50.155

As a result of lessons learned from the Fukushima Dai-ichi accident, the staff amended its regulations to establish regulatory requirements for nuclear power reactor applicants and

licenses for mitigating beyond-design-basis events in 10 CFR 50.155, “Mitigation of beyond-design-basis events.”

The applicable regulatory guidance for mitigation of beyond-design-basis events is as follows:

- RG 1.253, and
- DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap” (ML23277A139).

3.9.1 Technical Evaluation

The staff reviewed PSAR section 3.10, which states that mitigation of beyond design basis events will be addressed at the OL stage in accordance with 10 CFR 50.155. Table 1 of Appendix B of DANU-ISG-2022-01 states that 10 CFR 50.155 requirements only apply to 10 CFR Part 50 OLs. On the basis of the regulations in 10 CFR 50.155 and the guidance in DANU-ISG-2022-01, the staff finds that it is acceptable to defer addressing the requirements in 10 CFR 50.155 to the operating licensing phase of the KU1 review.

3.9.2 Conclusion

The staff finds that the review of the requirements found in 10 CFR 50.155 can be deferred to the operating licensing phase of the KU1 review consistent with the regulation and guidance.

3.10 Fuel System Design

PSAR section 3.11 incorporates by reference TR NAT-2806-A and states that it identifies the preliminary fuel design, acceptance criteria for the fuel design, the fuel qualification plan, and preliminary fuel qualification results. The PSAR also indicates that final fuel qualification results, including radionuclide retention requirements and associated fuel criteria to ensure conformance with principal design criterion (PDC) 10, will be provided at the OL stage. NAT-2806-A was reviewed and approved by the NRC, as documented by the staff’s SE included in the approved version of the TR.

The staff evaluated the KU1 fuel system design with respect to the following regulations:

- 10 CFR 50.34(a)(1)(ii)(D),
- 10 CFR 50.34(a)(3)(i), and
- 10 CFR 50.34(a)(4).

The applicable guidance to the Sodium reactor’s fuel system design includes:

- RG 1.233,
- RG 1.253, and
- DANU-ISG-2022-01.

3.10.1 Technical Evaluation

DANU-ISG-2022-01 states that the staff's review at the CP stage should focus on the role of the fuel in the safety analysis and the adequacy of the fuel qualification plan. Both issues are discussed in NAT-2806-A, which was incorporated by reference in the PSAR. The staff's evaluation of the preliminary fuel design and the associated acceptance criteria and qualification plan may be found in the safety evaluation included in NAT-2806-A.

3.10.1.1 NAT-2806-A Limitations and Conditions

The staff's conclusions in the SE on NAT-2806-A were subject to five L&Cs.

1. This TR represents an acceptable approach for qualifying Sodium Type 1 fuel and control assemblies for use in a reactor but does not in and of itself demonstrate that the fuel and control assemblies are qualified. Additional activities, including those discussed in the NRC staff's SE, must be completed to execute this plan and appropriately justify that the fuel and control assemblies are qualified.
2. This TR addresses the material properties and performance of U-10Zr and HT9 in fuel. If other materials are used in the fuel system in licensing applications, the applicant or licensee must demonstrate that they are manufactured according to standard specifications and used consistent with their qualification under relevant NRC-accepted codes and standards, or otherwise appropriately justified.
3. This TR does not provide a means for demonstrating that proposed SARRDLs are satisfied during normal operations and AOOs for the Sodium plant. The role of the fuel acceptance criteria is to demonstrate that the fuel system is not damaged as a result of normal operations and AOOs; if these criteria are satisfied, then the fuel system need not be further assessed against the SARRDLs. However, the SARRDLs must still be evaluated against other sources of radionuclides, including circulating radionuclides resulting from an appropriate number of random fuel failures.
4. The **[[]]** have not been subject to previous NRC review or approval. If they are to be used to develop design criteria and associated limits that support fuel assembly acceptance criteria, these design criteria and associated limits must be appropriately justified.
5. This TR does not address the extent to which the fuel system is expected to retain radionuclides following a cladding breach. If an applicant or licensee wishes to qualify Sodium Type 1 fuel with an expectation that radionuclides are expected to remain within the fuel following a cladding breach, models for fuel system radionuclide retention and release must be proposed and appropriately justified by comparison to experimental data.

These L&Cs are dispositioned as follows.

L&C 1 states that, while the fuel qualification plan presented in NAT-2806-A is acceptable, further work is needed to qualify fuel assemblies. As discussed in DANU-ISG-2022-01, fuel qualification activities do not need to be complete prior to NRC issuance of a CP. Because of this, the staff does not consider L&C 1 applicable to a CP application.

L&C 2 states that the fuel qualification plan is based on a uranium-zirconium alloy fuel in HT9 cladding. As discussed in PSAR section 7.1.1.1, this is consistent with fuel planned for use at KU1. Because of this, the staff determined L&C 2 is satisfied for the KU1 application.

L&C 3 states that the fuel qualification TR does not establish SARRDLs and does not demonstrate that SARRDLs are not exceeded. Section 1.3.2 of the PSAR provides the basis for the SARRDLs, which are established to ensure offsite dose criteria are not exceeded, and states that transient and accident analyses are used to demonstrate that SARRDLs are maintained for AOOs. The staff determined this is sufficient for the purposes of compliance with L&C 3 and therefore considers it to be satisfied for the KU1 application.

L&C 4 noted that the bases for certain fuel assembly acceptance criteria relied on documents that had not been reviewed or accepted by the NRC. Because the acceptance criteria are preliminary at the CP stage and are still subject to assessment through the qualification plan, the staff considers this L&C to not be applicable to a CP application.

L&C 5 states that the topical report does not address the extent to which the fuel system would be expected to retain radionuclides following a cladding breach, and that models for radionuclide retention following cladding failure must be proposed and appropriately justified. Radionuclide release from the fuel matrix is modeled using the release fractions assumed in the mechanistic source term. See section 3.2 of this SE for an evaluation of the mechanistic source term methodology and analyses.

Based on the conclusions above that each of the L&Cs associated with NAT-2806-A are either not applicable at the CP stage or are addressed appropriately by the CP application, the staff concluded that the incorporation by reference of NAT-2806-A into the PSAR is acceptable.

3.10.1.2 Role of Fuel in Safety Analysis

The fuel cladding presents the first SR barrier to radionuclide release in the functional containment strategy as discussed in PSAR section 1.3.2.1. The fuel cladding acceptance criteria and safety analysis approach are described in several topical reports incorporated by reference into the PSAR and discussed previously in chapter 3 of this safety evaluation. Fuel cladding acceptance criteria are provided in NAT-2806-A in terms of temperature, stress, strain, and other parameters. A time-at-temperature-no-failure (TATNF) criterion, which is discussed in more detail in NAT-9394-A and NAT-9390-A, integrates several of these parameters into a single acceptance criterion below which fuel cladding failure is precluded. As described in NAT-9394-A, transient analysis is conducted comparing the fuel temperature to the TATNF criterion. If the TATNF criterion is exceeded, the fuel cladding acceptance criteria are analyzed in more detail to determine whether fuel is damaged and to what extent.

As stated in section 3.10.1.1 of this SE, for LBEs that results in fuel cladding failure, radionuclide release (and retention in the fuel matrix) is modeled as discussed in NAT-9392-A. Dropped fuel assemblies and fuel assemblies subject to local faults (e.g., fuel misloads, partial flow blockages) are assumed to fail in their entirety as discussed in NAT-9394-A.

Because the role of the fuel in the safety analysis is described in detail in the topical reports incorporated by reference as noted above, the staff concluded that the role of the fuel in the safety is adequately described, consistent with DANU-ISG-2022-01.

3.10.1.3 Fuel Qualification

The staff concluded in the SE for NAT-2806-A that the fuel system qualification plan provided in the TR is an acceptable approach to fuel qualification and can be used to support compliance with the Sodium PDC. As such, the staff concluded that the fuel qualification plan for the PSAR is adequate, consistent with DANU-ISG-2022-01.

3.10.2 Conclusion

Because of the acceptability of the fuel qualification plan based on the review provided in NAT-2806-A, and because the staff understands the role of the fuel in the safety analysis as discussed above, the staff concluded that information provided relative to the fuel system design is acceptable for a CP application. The staff notes that additional work is needed to finalize the fuel qualification, as discussed in the safety evaluation to NAT-2806-A, and the staff will review this work in support of the OL.

3.11 Nuclear Design

PSAR section 3.12 provides an overview of the nuclear design of the KU1 reactor core. The discussion in the PSAR focuses on the determination of important core parameters, like reactivity coefficients, power distribution, and shutdown margin, but also includes nuclear design related topics like reactor stability and neutron fluence.

Applicable regulatory requirements for the evaluation of the nuclear design include:

- 10 CFR 50.34(a)(1),
- 10 CFR 50.34(a)(3)(ii),
- 10 CFR 50.34(a)(3)(iii),
- 10 CFR 50.34(a)(4), and
- 10 CFR 50.35.

This section incorporates by reference one TR, which has been reviewed and approved by the NRC:

- NAT-9393-A, "Reactor Stability Methodology," Rev. 0 (ML25211A276).

This section also incorporates by reference one technical report, whose content will be reviewed in the context of how it supports content in section 3.12 of the PSAR:

- TP-LIC-RPT-0011, "Core Design and Thermal Hydraulic Technical Report," Rev. 0 (ML24088A085)

The applicable regulatory requirements for the evaluation of SR SSCs are as follows:

- 10 CFR 50.34(a)(1), (4), and (5), and

- 10 CFR 50.35.

The following PDC apply to the nuclear design:

- PDC 10 – Reactor Design
- PDC 11 – Reactor Inherent Protection
- PDC 12 – Suppression of Reactor Power Oscillations
- PDC 26 – Reactivity Control Systems

3.11.1 Technical Evaluation

The content of application guidance in RG 1.253, which endorses NEI 21-07, provides limited guidance on information related to nuclear design. RG 1.70 and RG 1.206 provide guidance to large LWR OL and COL applicants on nuclear design contents of application, while NUREG-0800, section 4.3, Rev. 3 (ML070740003), provides guidance to staff. This guidance is not directly applicable, since KU1 is not an LWR, but the staff found NUREG-0800, section 4.3 provided a useful overall framework for conducting the review. Similarly, NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors Standard Review Plan and Acceptance Criteria” (ML042430048), provides guidance for reviewing the nuclear design of non-power reactors. This guidance is also not fully applicable because KU1 is a power reactor, but it provides guidance relevant to a wide variety of reactor designs. In consideration of these and other relevant guidance documents, the staff adopted the review approach used in the precedent Kairos Hermes SE (ML23158A268), which references NUREG-1537 but makes use of the overall structure from the LWR guidance.

The staff’s focus for the review of PSAR section 3.12 was to ensure that the nuclear design adequately supports the relevant PDC, specifically:

- PDC 10, which requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that SARRDLs are not exceeded during normal operation or AOOs. The nuclear design of the core, and in particular the power distribution, reactivity coefficients, and reactor kinetic parameters play a significant role in providing this assurance.
- PDC 11, which requires the reactor core and associated systems that contribute to reactivity feedback to be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.
- PDC 12, which requires the reactor core; associated structures; and associated coolant, control, and protection systems to be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.
- PDC 26, which provides requirements related to reactivity control, including specifying independent means of controlling reactivity and the functional capability that must be accomplished by those means.

The staff also reviewed the analytical methods used to perform the KU1 nuclear design and analysis to ensure that they are appropriate for demonstrating that the core design supports the PDC listed above.

3.11.1.1 Analytical Methods

The analytical methods used for neutronic analysis for KU1 are described in detail in sections 3, 4, and 5 of TP-LIC-RPT-0011. Section 3 describes the codes used, section 4 describes the applications of the codes in the steady state core design analysis methodology, and section 5 describes verification and validation of the methodology.

3.11.1.1.1 Tools and Methodology

The applicant used DIF3D and MC²-3, both of which are part of the (ANL) suite of simulation tools, but used them in conjunction with TerraPower-specific codes, including [[

]]. The ANL codes are also used with specific modules and options that are distinct from how they are applied at ANL. The staff found that the TP-LIC-RPT-0011 provides generally sufficient information to understand the methodology used for neutronic analyses of Sodium.

With respect to the depletion methodology, the staff confirmed that the neutronic modeling workflow accounts for [[

]], which are important metal-fueled SFR phenomena. The depletion workflow also accounts for [[

]]. [[

]]. The depletion methodology is expected by the staff to provide a relatively high level of fidelity, beyond what is typically performed at ANL using REBUS. While the staff found the approach to be reasonable for the CP stage, a more detailed report describing the depletion modeling workflow would be expected in the OL submission, or as part of the code documentation and verification and validation (V&V), since only limited information was provided on [[

]]. The depletion methodology is expected by the staff to provide a relatively high level of fidelity, beyond what is typically performed at ANL using REBUS. While the staff found the approach to be reasonable for the CP stage, a more detailed report describing the depletion modeling workflow would be expected in the OL submission, or as part of the code documentation and verification and validation (V&V), since only limited information was provided on [[

The staff also reviewed the neutronic deterministic options used. While TP-LIC-RPT-0011 identifies the use of DIF3D with [[

]], it does not specify the order of the solution; staff confirmed that a [[

]] was used, which should be appropriate for this type of SFR core. TP-LIC-RPT-0011 also does not specify the fine group structure, but the staff identified in audit that TerraPower uses a [[

]]. Limited detail is provided on the [[

]], but the staff identified in audit that it differs slightly from the approach recommended by ANL (Lee and Stauff, 2015). [[

]]. While this is correct, [[

]].

While the methodology identified by the applicant differs from the approach recommended by ANL in some key ways, the modeling choices are adequate for a preliminary analysis. Further work should be performed to quantify the impact of these deterministic options (number of fine energy group cross-section, heterogeneous cross-section treatment in fuel cells, scattering orders, etc.) and benchmark the code to higher-order methods (i.e., Monte Carlo). Based on audit discussions, the staff expects that additional information will be available as part of the OL application.

3.11.1.1.2 Treatment of uncertainties

As discussed in the technical report, uncertainties are [[

]].

A preliminary [[] was provided in tables 8-13 through 8-16 of TP-LIC-RPT-0011. The staff reviewed the supporting calculations for [[] in audit, and while there were a number of open items and assumptions, the staff found the overall approach to be reasonable for preliminary analyses with further refinement expected by the OL application. The staff expects to perform a more detailed review at the OL stage of uncertainty analyses at beginning of equilibrium cycle and end of equilibrium cycle (with higher fraction of plutonium bred into the core), together with an uncertainty evaluation of reactivity coefficients and with inserted rods.

Uncertainty quantification is also particularly important for the for the shutdown margin calculation and the decay heat calculation, described in section 4.1.1.5 and 4.1.1.6 of TP-LIC-RPT-0011, respectively. The shutdown margin calculation includes a significant allowance for uncertainty, as shown in table 8-6 of TP-LIC-RPT-0011. The staff audited the documentation of this uncertainty calculation and verified that it accounted for key biases and uncertainties, but noted [[

]].

The staff considers the overall approach to uncertainty quantification to be reasonable for this kind of analysis, but the analyses are preliminary and subject to refinement and further confirmation at the OL.

3.11.1.1.3 Benchmarking, verification, and validation

Section 5.1 of TP-LIC-RPT-0011 discusses V&V and includes a significant number of highly relevant experiments (based on [[] to validate neutronics simulations tools applied to Sodium. The staff noted one missing neutronic validation analysis with regards to decay heat modeling, and additional justification will need to be provided at the OL as to the experimental basis for the decay heat calculations. Benchmark activities using historic experimental data will be complemented by code-to-code benchmark exercises. Validation activities have yet to be completed and will be documented for the OL. Consistent with staff reviews of transient analysis methodologies, this is acceptable at the CP stage.

3.11.1.2 Power Distribution

Technical report section 3.12.1.3 discusses the power distribution in terms of the peaking factors, $F_{\Delta H}$, F_{XY} , and F_Z , all of which are consistent with the common definitions. Power distribution calculations are described in section 4.1.1.2 of TP-LIC-RPT-0011, which notes that [[

]]. Preliminary radial and axial power distributions and peaking factors are shown in sections 8.1.2 to 8.1.5 of TP-LIC-RPT-0011 and appear reasonable to the staff. The staff also audited [[

]]; this is consistent with observations for other high-assay low-enriched uranium (HALEU) fueled fast reactors (Stauff et al., 2025).

3.11.1.3 Reactivity Control and Shutdown Margin

Section 4.1.1.4 of TP-LIC-RPT-0011 describes the process for determining control rod patterns and rod worth, which is not described in the PSAR. TP-LIC-RPT-0011 describes how [[

]]; this is consistent with the discussion on the control rod function in PSAR section 7.1.1.3.1. The rod worth is determined at [[

]]. The staff determined that rod worth calculation method was discussed in sufficient detail and appropriately accounted for the key phenomena and was thus reasonable to evaluate the control system as needed for PDC 10, 11, 12, and 26.

The shutdown margin associated with the primary and secondary control rods is a significant portion of how the applicant demonstrates compliance with PDC 26 as described in PSAR section 5.3.3. PSAR section 3.12.1.4 discusses the shutdown margin calculation, which is described in more detail in section 4.1.1.5 of TP-LIC-RPT-0011. The shutdown margin analysis method described in TP-LIC-RPT-0011 and parameters accounted for in the shutdown margin calculation described in the PSAR are consistent with the staff's expectations for shutdown margin calculations and are thus acceptable. Sample calculations are given in section 8.1.10 of TP-LIC-RPT-0011. The staff noted that in table 8-6 of TP-LIC-RPT-0011, [[

]]. Discussion in the footnote of the table indicates that the applicant thinks this can be addressed by the OL by [[

]]. The staff determined that this can be accommodated (either by the means referenced by the applicant or by changes in core design) and expects to review this information at the OL.

3.11.1.4 Kinetic Parameters and Reactivity Coefficients

Section 4.1.1.3 of TP-LIC-RPT-0011 describes how kinetic parameters and reactivity coefficients are determined. For kinetic parameters, the staff verified that [[

]] are consistent with fast reactor theory and are acceptable. Reactivity coefficients are determined by [[

]]. The staff expects the overall approach would give reasonable results for [[]], but the results would be dependent on [[]]. This is reasonable provided [[]], which appears to be the case based on the sample results provided in section 8 of TP-LIC-RPT-0011.

The staff noted that the reactivity coefficient approach included specific provisions for ensuring the coefficients [[]]

]]. This is reasonable to the staff for the preliminary analysis because the [[]] of the safety analysis by the OL.

The core restraint system plays an important role in the reactivity feedback of the core and analysis of its effect is discussed separately in section 4.2 of TP-LIC-RPT-0011. The reactor uses a core restraint system where fuel is constrained vertically and laterally at the assembly inlet, but assemblies laterally support each other at an above core load pad and top load pad, and ultimately the core restraint system. Loads on the core assemblies caused by thermal effects result in assembly bowing as shown in figure 4-2 of TP-LIC-RPT-0011. Assembly bowing is calculated using a variety of tools.

Thermal hydraulic analyses supporting the assembly bowing analysis are performed using [[]]. This code's role in thermal-hydraulic analysis is discussed in more detail in section 3.12 of this SE. In section 4.2.1 of TP-LIC-RPT-0011, the applicant indicated that [[]]. Though [[]]

]] and thus expects to review it at the OL.

Mechanical analysis supporting the assembly bowing calculations are performed using [[]], which is described in slightly more detail in NAT-2806-A. [[]]. While [[]] has not been fully verified and validated, NAT-2806-A also identified a number of tests to support validation of the mechanical analysis of fuel assemblies. The use of [[]] for the assembly bowing calculations is reasonable to the staff because [[]]

]]. The staff will review [[]] in more detail at the OL stage when V&V activities are complete.

Neutronic analyses associated with assembly bowing are performed using [[]], which is described in section 3.1.6 of TP-LIC-RPT-0011. [[]]

]]. The code is well-documented in the open literature as referenced in TP-LIC-RPT-0011, where good agreement is reported between [[]]. As such, the staff finds the use of [[]] to be reasonable for preliminary analyses, but expects to review further benchmarking, verification, and validation as part of the OL.

PSAR section 3.12.1.2 describes the reactivity coefficients the applicant considers important for the Sodium reactor core design and evaluation, which include Doppler, fuel density, coolant density, cladding density, structure density, and radial expansion. Other parameters are important for transient analyses, including axial expansion (which is a function of coolant and structure density and thermal expansion of the fuel system materials). Reactivity coefficients for transient analyses are calculated using the core design methods described in TP-LIC-RPT-0011 and incorporated into the transient analyses described in PSAR chapter 3.

As shown in table 13.11-1, below, the preliminary kinetics parameters and reactivity coefficients are reasonable when compared to publicly available data on other reactors (both operated and designed). While the core-wide reactivity coefficients are negative, consistent with PDC 11, the staff notes that local [REDACTED]

[REDACTED], as shown in figure 8-10 of TP-LIC-RPT-0011. This is expected for an SFR core of this size, and is generally compensated for by other, more negative coefficients resulting in the overall negative core-wide coefficients. However, for highly localized overheating scenarios (e.g., core assembly blockage) local neutronic effects should be closely scrutinized. The staff notes that this analysis was done for the local blockages considered in NAT-9395-A where it was found to be acceptable. The staff will continue to verify that positive local reactivity coefficients are adequately addressed at the OL stage.

Finally, the staff notes that while section 4.1.1.3.4 of TP-LIC-RPT-0011 provides a methodology for calculating [REDACTED], it was not explicitly provided in the sample calculations in TP-LIC-RPT-0011 section 8. This is reasonable for the CP given that [REDACTED], but should be discussed for completeness at the OL.

3.11.1.5 Nuclear Stability

PSAR section 3.12.1.6 states that reactor stability is evaluated using the stability methodology documented in NAT-9393-A. This methodology was found to be acceptable to the staff subject to two L&Cs:

1. Inputs provided to the methodology calculated by other methodologies are to capture the higher-fidelity behavior of the identified important phenomena in a manner consistent with their incorporation into this methodology.
2. The topical report develops a [REDACTED] for the purpose of describing how such a [REDACTED] may be obtained and for the purpose of describing how such a [REDACTED] is subsequently applied as part of the methodology's calculation steps. In application, a [REDACTED] must be developed and appropriately justified for the use described in this methodology. Any applied [REDACTED] must be reviewed and approved by the NRC.

With respect to L&C 1, TP-LIC-RPT-0011 describes in several sections how the reactivity coefficients are computed for the stability methodology. The staff confirmed that the method for determining these coefficients appropriately captures the higher-fidelity behavior of the phenomena sufficient for preliminary analysis and thus determined L&C 1 was satisfied.

With respect to L&C 2, the staff audited the detailed nuclear stability analyses and found that the calculations and associated [REDACTED] were the same as the demonstration application and [REDACTED] developed based on comparison to

Fermi 1 benchmarks provided in the TR. While the Fermi 1 benchmark had challenges due to differences in design and phenomena, as noted in the staff's SE for NAT-9393-A, the staff ultimately found that it produced "an acceptable evaluation of a benchmarking exercise comparing its analytic methods to stability experiments in the Fermi 1 reactor." As such, the staff considers the [] employed to be "reviewed and approved" by the NRC consistent with L&C 2 and thus determined L&C 2 was satisfied.

PSAR section 3.12.1.6 states that a preliminary analysis indicated no instabilities were identified over the range of power evaluated, and that full results demonstrating compliance with PDC 12 would be provided at the OL stage. The staff audited the analysis and confirmed that no instabilities were identified. As discussed above, the staff found the analysis to be consistent with the demonstration analysis provided in the TR. The staff determined this is adequate for a CP, because no instabilities are expected for a reactor of this type as confirmed by the preliminary analysis, but will review more detailed analyses supporting compliance with PDC 12 at the OL.

3.11.1.6 Structural Material Irradiation

Section 4.3 of TP-LIC-RPT-0011 describes the methodology used for fluence and [] calculations, which uses []. The method is consistent with other similar analyses available in the open literature (Fei et al., 2021) and as such the staff considers it to be reasonable for preliminary analyses. The staff audited preliminary calculations using this method and found them to be consistent with staff expectations regarding fluence on structural components for this kind of reactor. PSAR section 3.12.1.7 confirms that the final analysis will be provided at the OL, and based on the adequacy of the method and the preliminary analysis the staff determined this to be reasonable.

3.11.1.7 Nuclear Core Design Limits

No nuclear core design limits were provided in the PSAR, though TP-LIC-RPT-0011 refers to []. The fuel temperature limits and other limits related to fuel thermal and mechanical performance are discussed in detail in NAT-2806-A. PSAR table 11.5-1 further indicates that appropriate core design limits are being considered for TSs either through a cycle-specific core operating limits report (COLR) or through their inclusion as safety limits or limiting conditions for operation. Because appropriate limits are identified as probable subjects of plant TS pursuant to 10 CFR 50.34(a)(5), and because the staff has some confidence in the nuclear design methodology used to determine these limits as discussed above, the staff determined the information provided is sufficient for a CP and final core design limits were reasonable to address at the OL.

3.11.1.8 Monitoring

The nuclear instrumentation system (XIS) provides instrumentation to sense neutron flux, as described in sections 1.1.4.1.12 and 7.6.4 of the PSAR. Sensing is provided by four high temperature fission chambers on the perimeter of the core barrel structure, with power range, wide range, and source range signals. For the preliminary design, this configuration of neutron detection equipment is consistent with prior SFR experience (e.g., Bennett et al., 1980) and is thus acceptable. The XIS is evaluated in more detail from an instrumentation and control perspective in section 7.6.4 of this SE.

3.11.2 Conclusion

Based on the discussion above, the staff determined that the nuclear design analytical methodology provided in TP-LIC-RPT-0011 is adequate to support preliminary analyses demonstrating the KU1 reactor is consistent with PDC 10, 11, 12, and 26. The PSAR and TP-LIC-RPT-0011 provide sufficient results for the staff to confirm that the preliminary nuclear design supports these PDC. The staff determined that the preliminary evaluation included in TP-LIC-RPT-0011 provides an adequate preliminary analysis and evaluation of the design and performance of SSCs at the facility per 10 CFR 50.34(a)(4), and the nuclear design information provided in the PSAR is sufficient for the staff to determine that the applicant adequately described the design bases and the relationship of the design bases to the PDC per 10 CFR 50.34(a)(3)(ii). Those areas that require further development prior to the issuance of an OL, per 10 CFR 50.35, are discussed above.

3.11.3 Tables

Table 3.11-1: Comparison of KU1 neutronic parameters with other SFR designs

Parameter	KU1 (TP-LIC-RPT-0011 Tables 2-4 and 8-7)	Comparison with Other SFR Designs
Core Maximum Neutron Flux [10^{15} n/cm ² -s]	[[]]	~7 in FFTF (Boyd et al., 1967)
Core Maximum Power Density at BOL [W/cm ³]	[[]]	~375-400 in ABR1000 (Grandy, et al., 2007)
Peak Fuel Pin Linear Power (BOL) [kW/m]	[[]]	42 in FFTF (Waltar et al., 2012)
Peak Discharge Burnup [MWd/kg]	[[]]	Up to 200 in (Crawford et al., 2007)
Burn-up Reactivity Swing [pcm]	[[]]	-2200 in ABR1000 (Kim, et al., 2009) and VTR (Heidet and Fei, 2022)
Total Delayed Neutron Fraction (BOL) [pcm]	[[]]	300 in FFTF (Boyd et al., 1967); 369 in VTR (Heidet and Fei, 2022); 671 in VTR (HALEU) (Stauff and Heidet, 2019)
Prompt Neutron Generation Time [ns]	[[]]	350 in FFTF (Boyd et al., 1967); 387 in VTR (Heidet and Fei, 2022)
Doppler Coefficient [pcm/K]	[[]]	-0.31 in VTR (Heidet and Fei, 2022); -0.44 in FFTF (Usman, Lartaud, and Stauff, 2019)
Doppler Coefficient (Voided) [pcm/K]	[[]]	-0.30 in VTR (Heidet and Fei, 2022)
Na (Volume) Expansion [pcm/K]	[[]]	-0.4 in FFTF (Boyd et al., 1967); -0.71 in VTR (Heidet and Fei, 2022)

3.12 Thermal Hydraulic Design

PSAR section 3.13 describes the thermal-hydraulic design of the core, specifically, the heat transport from the reactor core to the primary sodium coolant. It also discusses the design bases relevant to core thermal-hydraulic design. To provide additional detail on the analytical methods and preliminary core thermal-hydraulic results, PSAR section 3.13 also incorporates by reference TP-LIC-RPT-0011, which was submitted as part of the CP application. The content of this report will be reviewed in the context of how it supports content in PSAR section 3.13.

Applicable regulatory requirements for the evaluation of the core thermal-hydraulic design include:

- 10 CFR 50.34(a)(1),
- 10 CFR 50.34(a)(3)(ii),
- 10 CFR 50.34(a)(3)(iii),
- 10 CFR 50.34(a)(4), and
- 10 CFR 50.35.

This section also incorporates by reference one technical report, whose content will be reviewed in the context of how it supports content in section 3.13 of the PSAR:

- TP-LIC-RPT-0011

The following PDC apply to the thermal-hydraulic design:

- PDC 10 – Reactor Design
- PDC 15 – Primary Coolant System Design
- PDC 34 – Residual Heat Removal

3.12.1 Technical Evaluation

Similar to the discussion relative to the nuclear design in section 3.11 of this SE, the content of application guidance in RG 1.253, which endorses NEI 21-07, provides limited guidance on information related to core thermal-hydraulic design. Consequently, the staff's focus for the review of PSAR section 3.13 was to ensure that the nuclear design adequately supports the relevant PDC, specifically:

- PDC 10, which requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that SARRDLs are not exceeded during normal operation or AOOs. The core thermal-hydraulic design is necessary to ensure adequate cooling, which plays a key role in maintaining core and primary coolant system thermal limits.

- PDC 15, which requires the primary coolant system and associated auxiliary, control, and protection systems to be designed with sufficient margin to ensure that the design conditions of the safety-significant elements of the primary coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. The core thermal-hydraulic design and associated analyses are used to develop the limits and setpoints that protect the primary coolant system boundary.
- PDC 34, which requires a system to remove residual heat to be provided. The system function during normal operations and AOOs is to remove decay heat and other residual heat from the reactor at a rate such that SARRDLs and design conditions of the primary coolant boundary are not exceeded. The thermal-hydraulic design of the core is important to ensuring the adequacy of the residual heat removal system.

The staff also reviewed the analytical methods used to perform the KU1 thermal-hydraulic design and analysis to ensure that they are appropriate for demonstrating that the PDC listed above are supported by the core thermal-hydraulic design.

3.12.1.1 Analytical Methods

The analytical methods used for thermal-hydraulic analysis for KU1 are described in detail in sections 3, 4, and 5 of TP-LIC-RPT-0011. Section 3.2 of TP-LIC-RPT-0011 describes the codes used, sections 4.1.2 and 4.2.1 of TP-LIC-RPT-0011 describe the applications of the codes in the steady state core design analysis methodology, and section 5 of TP-LIC-RPT-0011 describes V&V of the tools.

3.12.1.1.1 Tools and Methodology

Section 3.2 of TP-LIC-RPT-0011 describes three codes used for thermal-hydraulic analysis of the core, with different levels of fidelity, ranging from a simplified, one-dimensional core model to a fully three-dimensional computational fluid dynamics (CFD) model.

The simplified core thermal-hydraulic method is discussed in TP-LIC-RPT-0011 section 3.2.1. There are simple equations for calculating core flow distribution, core-wide pressure drop, and assembly pressure drop; and a more complex model for fluid and cladding temperatures referred to as Nascent. Nascent was previously described to the staff in white paper as NAT-3049, "Nascent Thermal Hydraulic Model White Paper" (ML23080A013). As described in this white paper and TP-LIC-RPT-0011, the methods in the Nascent model are based on

[[

]] The staff's feedback (ML23229A541)

on NAT-3049 generally identified it as reasonable for simplified analyses, though there were questions regarding the calibration and V&V of the model. Based on this, the staff determined that the use of the simplified model is reasonable for preliminary analyses. V&V of the thermal-hydraulic approach is discussed in more detail below.

Mongoose++ is discussed in section 3.2.2 of TP-LIC-RPT-0011. It is a subchannel code developed internally by TerraPower for core thermal-hydraulics analysis of sodium-cooled reactors and is used to estimate [[

]]. Mongoose++ was also discussed in more detail as part of NAT-9395-A. The staff determined it to be generally consistent with accepted modeling approaches for subchannel analyses used in LWRs, with SFR-specific models for [[

]]. Ultimately, Mongoose++
[[
to be completed (e.g., [[

]]. Due to its similarity with LWR subchannel codes the staff is familiar with and the adequacy of the preliminary results provided in section 8 of TP-LIC-RPT-0011, the staff determined that the use of Mongoose++ is reasonable for preliminary analyses. More detailed review of the code, including future work and V&V, will be conducted at the OL.

The CFD code discussed in section 3.2.3 of TP-LIC-RPT-0011 is [[]]. While the staff is familiar with this code and its capabilities, the staff notes that it is impractical to review a CFD code outside of its use in a specific licensing application due to the wide range of possible models that could be used, many of which may be inappropriate for specific applications, and the complexity of the interplay of these models with specific meshing choices. The applications for CFD highlighted in TP-LIC-RPT-0011 do not directly involve licensing activities but focus on supporting the understanding of the results of experiments and support for determining whether licensing calculations adequately account for higher-dimensional phenomena. The staff determined the use of [[]] is reasonable in these applications, but use in direct licensing analyses requires more detailed review by the staff.

3.12.1.1.2 Treatment of uncertainties

Uncertainties in the thermal-hydraulic analysis are treated with [[]], as discussed in section 3.11.1.1.2 above for the neutronic uncertainties. The [[

]]. The staff determined this was reasonable for a preliminary analysis where some detailed design information may not be available, but a complete justification for [[]] should be made available at the OL.

3.12.1.1.3 Benchmarking, verification, and validation

Section 5.2 of TP-LIC-RPT-0011 describes validation of the thermal hydraulic methodologies using the codes and methods discussed above. Section 5.2.1 of TP-LIC-RPT-0011 describes a variety of legacy tests for basic thermal hydraulic phenomena in [[

]]. Section 5.2.2 of TP-LIC-RPT-0011 describes new thermal-hydraulic testing being performed at [[]]. This testing includes [[

]]. Section 5.2.3 of TP-LIC-RPT-0011 describes the use of high-fidelity CFD for [[

]]. The staff stresses the importance of [[

]]. In audit, the staff was able to examine benchmarks of [[]] and [[]]. These benchmarks showed reasonably good agreement, but further benchmarking and validation is necessary. The validation activities described in TP-LIC-RPT-0011 have yet to be completed and will be documented for an OL application. Consistent with staff reviews of transient analysis methodologies, this is acceptable at the CP stage.

3.12.1.2 Thermal-Hydraulic Design

Section 4.1.2.1 of TP-LIC-RPT-0011 describes the core flow distribution and pressure drop analyses. These analyses currently assume [[]]. This is a reasonable starting point for preliminary analyses, and the applicant noted that [[]] as discussed above in section 3.12.1.1.3 of this SE. As shown in TP-LIC-RPT-0011 figure 2-4, [[]]

]].

Section 4.1.2.2 of TP-LIC-RPT-0011 describes the assembly pressure drop analysis. [[]]

]].

Section 4.1.2.3 of TP-LIC-RPT-0011 describes the peak fluid and cladding temperature analysis. During normal operation, the core operates with coolant inlet temperature of 360°C and bulk outlet temperature of 510°C, as indicated in table 2-4 of TP-LIC-RPT-0011. The temperature rise varies between assemblies based on heat generated in individual assemblies. Considering uncertainties, the peak fuel and cladding temperatures are [[]], respectively. The former demonstrates margin to fuel melt (established at [[]] in NAT-9390-A). The latter demonstrates margin to

[[]] and provides margin to [[]]. proposed for steady-state operation in NUREG/CR-7305, the purpose of which is to minimize thermal creep of the cladding and fuel-cladding chemical interaction. As such, these results support PDC 10 and are reasonable for steady-state operation.

Preliminary thermal hydraulic results are shown in section 8 of TP-LIC-RPT-0011. Table 8-1 of TP-LIC-RPT-0011 provides [[]]

]]. The flow distributions tabulated in table 8-10 of TP-LIC-RPT-0011 appear reasonable and consistent with the [[]]. The core pressure drops shown in table 8-11 of TP-LIC-RPT-0011 are also reasonable, and in particular demonstrate that [[]]

]]. Table 8-12 of TP-LIC-RPT-0011 shows the peak cladding inner diameter temperature, which [[]]

II.

3.12.2 Conclusion

Based on the discussion above, the staff determined that the thermal-hydraulic design analytical methodology provided in TP-LIC-RPT-0011 is adequate to support preliminary analyses demonstrating the KU1 reactor is consistent with PDC 10, 15, and 34. The PSAR and TP-LIC-RPT-0011 provide sufficient results for the staff to confirm that the preliminary thermal-hydraulic design supports these PDC. The staff determined that the preliminary evaluation included in TP-LIC-RPT-0011 provides an adequate preliminary analysis and evaluation of the design and performance of SSCs at the facility per 10 CFR 50.34(a)(4), and the thermal-hydraulic design information provided in the PSAR is sufficient for the staff to determine that the applicant adequately described the design bases and the relationship of the design bases to the principal design criteria per 10 CFR 50.34(a)(3)(ii). Those areas that require further development prior to the issuance of an OL, per 10 CFR 50.35, are discussed above.

3.13 Criticality Safety of Fresh and Spent Fuel

PSAR section 3.14 addresses conformance with PDC 62 for prevention of criticality in fuel storage and handling systems as well as 10 CFR 50.68, "Criticality accident requirements," except for the U-235 enrichment limit defined in 10 CFR 50.68(b)(7). PSAR section 3.14 is focused on the methodologies used to determine the upper subcritical limit (USL) and perform criticality analyses for each area where new and spent fuel is stored and handled. The preliminary designs for these areas and their SSCs are provided in sections 7.1.2, 7.3.1, 7.3.2, and 7.3.3 of the PSAR and evaluated in the associated sections of this SE.

Exemption to 10 CFR 50.68(b)(7) is addressed in NAT-9403, "Regulatory Exemptions" (ML24088A084), and evaluated in appendix B of this SE.

The applicable regulatory requirements for the evaluation of safety-related SSCs are as follows:

- 10 CFR 50.34(a)(1) through (5),
- 10 CFR 50.35, and
- 10 CFR 50.68, which provides requirements related to the design of storage for special nuclear material and associated criticality safety analyses.

The following PDC applies to the criticality safety of fresh and spent fuel:

- PDC 62 – Prevention of Criticality in Fuel Storage and Handling

The applicable guidance for criticality safety of fresh and spent fuel is as follows:

- RG 1.240, "Fresh and Spent Fuel Pool Criticality Analysis," Rev. 0 (ML20356A127),
- NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (ML050250061),

- NUREG/CR-7308, “Sensitivity/Uncertainty Methods for Nuclear Criticality Safety Validation” (ML25099A002),
- NUREG/CR-7311, “Determination of Bias and Bias Uncertainty for Criticality Safety Computational Methods” (ML25120A337).

3.13.1 Technical Evaluation

The staff reviewed PSAR section 3.14 in accordance with applicable portions of the guidance listed above to assess compliance with the regulatory requirements also listed in that section. The staff focused its review on the applicant’s criticality safety analysis methodology and audited calculations to better understand application of the methodology and the subcriticality margins in the preliminary design. The staff also reviewed design information pertinent to criticality safety. In addition, the staff evaluated the proposed variables and conditions for TSs as they relate to PSAR section 3.14.

3.13.1.1 Analytical Methods

The applicant used Monte Carlo N-Particle Transport Code System Version 6.2 (MCNP 6.2) (Los Alamos National Laboratory (LANL), 2017) to perform criticality safety calculations. MCNP 6.2 is a radiation transport code developed by LANL. It has a broad spectrum of applications in nuclear science and engineering, including criticality safety analyses. MCNP 6.2 is capable of modeling the necessary physics and uses the same neutron transport method as confirmatory analysis methods used by the staff. For these reasons, the staff finds the use of MCNP 6.2 acceptable for this application.

In addition, the applicant used the Whisper-1.1 program (Kiedrowski, et al., 2015), also developed by LANL and distributed with MCNP 6.2, to help compute USL. The use of Whisper-1.1 is evaluated below.

The staff noted that USO’s quality assurance program as described in TP-QA-PD-0001, “TerraPower Quality Assurance Program Description,” Rev. 14-A (ML23213A199) includes software design control requirements. The staff confirmed in audit that MCNP 6.2 and Whisper-1.1 will be installed and maintained in accordance with these requirements.

3.13.1.2 Code Validation and Determination of USL

To have confidence in the results of criticality safety calculations, the code used must be validated by modeling benchmark experiments with the code and comparing the results against the experimental data. It is important for the benchmark experiments selected to adequately represent the system to be analyzed. The comparison provides the bias and bias uncertainty associated with the code and methods. The USL is an upper limit on the calculated effective neutron multiplication factor, or k_{eff} , that is conservatively adjusted to account for this bias and bias uncertainty, along with additional subcritical margin.

PSAR section 3.14.2 describes two approaches used to calculate the USL. The first is the method outlined in NUREG/CR-6698. The second uses the Whisper-1.1 program. With either approach, the USL will be calculated using PSAR equation 3.14-2, which includes the bias, bias uncertainty, administrative margin, and additional margin that may be needed to account for extensions to the area of applicability (AoA).

PSAR section 3.14.2 states that an administrative margin of 0.05 is typically applied, which is consistent with the margin in 10 CFR 50.68(b)(2) and (b)(4). PSAR section 3.14.2 also states that the bias is set to zero when the calculated average k_{eff} is greater than 1.0, meaning that the code is overpredicting k_{eff} on average. Setting the positive bias to zero is conservative and aligns with the methodology outlined in NUREG/CR-6698.

3.13.1.2.1 NUREG/CR-6698 Method

PSAR section 3.14.2 states that applicable benchmarks are selected in accordance with the methodology outlined in NUREG/CR-6698. The key parameters USO uses to determine applicability include geometry, U-235 enrichment, fissile material composition, neutron energy spectrum, coolant type, reflector material, fuel density, and temperature. The staff notes that a limited number of benchmarks is available for metal fueled systems that match the enrichment of the Natrium reactor. An even smaller subset is available when the use of sodium as a moderator and neutron absorbers are considered. USO has added a handful of experiments from the NEA/NSC/DOC(2006)01, "International Handbook of Evaluated Reactor Physics Benchmark Experiments," to supplement those from the NEA/NSC/DOC(95)03, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," to try to address this.

Using the methods outlined in NUREG/CR-6698, the benchmark experiments are modeled in MCNP 6.2 and the resulting k_{eff} and associated uncertainty are used along with the experimental k_{eff} and associated uncertainty to develop a lower tolerance limit (K_L) with 95 percent confidence that 95 percent of data will be above the limit. The additional administrative margin and AoA margin are subtracted from K_L to get the USL. Confirming that the results from a criticality safety calculation ($k_{\text{eff}} + 2\sigma$) are below the USL provides a greater than 95 percent probability that the actual k_{eff} of the system will be subcritical with greater than 95 percent confidence.

The staff reviewed preliminary criticality code validation calculations using the NUREG/CR-6698 method in audit. The calculations appeared to conform with the NUREG guidance. Selection of applicable benchmarks appeared reasonable and consistent with the Natrium design to the extent that applicable benchmarks were available. In these preliminary calculations, additional margin was not added to the USL to address limitations on the AoA. This should be justified at the OL stage, given the limited number of benchmarks available with key parameters matching the Natrium systems. In addition, consideration may need to be given to smaller sets of data during trend analysis at the OL stage to ensure trends are not inadvertently obscured based on data that are not highly representative of the system of interest.

3.13.1.2.2 Whisper Method

Whisper-1.1 calculates a baseline USL based on sensitivity/uncertainty (S/U) methods. MCNP calculates sensitivity coefficients to determine how changes in nuclear data affect k_{eff} for both benchmark experiments and the system of interest (e.g., the Natrium spent fuel pool). These coefficients are combined with nuclear data covariances to compute the correlation coefficient c_k , which measures the neutronic similarity between benchmarks and the system. Whisper-1.1 weights benchmarks by their c_k values to determine which will be used to calculate the USL, emphasizing those most like the system. If too few relevant benchmarks exist, a noncoverage penalty may apply. Whisper then uses extreme value theory to calculate the USL.

The staff audited preliminary USL calculations using Whisper-1.1 and notes that Whisper-1.1 provides a justifiable quantitative method for USL determination for this application, considering

the few directly applicable benchmarks and the amount of margin available in the preliminary calculations. In addition, previous studies have shown that the Whisper-calculated USL is generally conservative compared to traditional criticality safety validation methods (Abdel-Khalik, Hany, et al., 2023) although it should be noted the studies were performed on models with little similarity to the Sodium design. The staff notes that users must exercise caution when using Whisper-1.1 and other S/U methods to not over-rely on the automated process. The staff expects the benchmark models and sensitivity matrices to be carefully reviewed and justified by applicants, as they are generated by a third party. In addition, the AoA of the validation suite still needs to account for the key attributes of the system of interest. The staff expects the Whisper USL to be treated as a baseline, with additional margin added by an applicant to ensure conservatism, if needed. Finally, applicants should evaluate new directly applicable benchmarks, such as those provided by the Department of Energy/NRC Collaboration for Criticality Safety Support for Commercial-Scale HALEU Fuel Cycles and Transportation program, and identify whether it is necessary to repeat the validation study using the new data to confirm the conservatism of the USL.

3.13.1.3 Criticality Safety Analyses

The methodology used for the preliminary criticality safety analyses is documented in PSAR section 3.14.2.1 through 3.14.2.4. Design bases and key controls are provided in PSAR sections 7.1.2, 7.3.1, 7.3.2, and 7.3.3 and evaluated in the associated sections of this SE. Key aspects of the methodology include:

- Conservative fuel assembly model
- Conservative storage location models (i.e., SFP rack, EVST, IVS, and NAPS)
- Cases that bound credible abnormal events
- Fuel is modeled as fresh at the maximum enrichment
- A maximum of 90 percent of the boron in the SFP borated stainless steel neutron absorber material will be credited
- Moderator exclusion will be credited within the failed fuel canisters in the SFP
- The IVS is modeled with white or reflective boundary conditions

The SFP analysis is performed in accordance with RG 1.240 as stated in PSAR table 1.4-1 and PSAR sections 3.14.1, 3.14.2.1, and 7.3.1.2. The staff notes that RG 1.240 was developed for LWRs; however, much of the guidance in it is applicable to the Sodium design.

The methodology described in PSAR section 3.14.2 supports PDC 62 in providing the approaches used for calculation of the USL and criticality safety evaluations of fuel storage locations to help ensure that criticality is prevented. The descriptions of the design features and SSCs that are evaluated with these methodologies and relied on to prevent criticality are evaluated in sections 7.1.2, 7.3.1, 7.3.2, and 7.3.3 for the SE.

PSAR section 7.3.1.3 states that a neutron absorber material monitoring program will be provided at the OL stage. The staff notes that such a program is essential to ensure that the credited criticality control is maintained throughout operation. In addition, PSAR section 7.3.1.3 states that the failed fuel canisters are leak tight, designed to ASME Boiler and Pressure Vessel Code (BPVC) Section III, "Rules for Construction of Nuclear Facility Components,"

requirements, and tested to the applicable leakage criteria in American National Standards Institute (ANSI) N14.5, "Leakage Tests for Radioactive Materials," which supports the moderator exclusion assumption.

Consistent with the staff expectations, the applicant identified fuel storage requirements as a probable subject of TS design features in PSAR table 11.5-1. This effectively creates a requirement for fuel storage to be consistent with the assumptions in the criticality safety analysis of record.

The staff audited preliminary criticality safety calculations for the IVS, SFP, and NAPS. The staff observed that, in general, the models used conservative assumptions, and the calculations included comprehensive sensitivity studies to ensure a conservative calculation of k_{eff} . Each of the preliminary calculations showed the maximum calculated k_{eff} remained below the applicable USL.

For the IVS calculation, the staff audited the as-modeled IVS boundary condition to determine if it was conservative relative to the neutron flux the IVS would receive from the reactor core. The staff notes that it is not possible to discern the true k_{eff} of the IVS when explicitly modeling the core because the core k_{eff} would dominate the calculation. The staff audited a sensitivity study that confirmed minimal differences in core k_{eff} when modeling the core with and without the IVS, suggesting minimal neutronic coupling. The staff also audited a calculation which confirmed that subcritical multiplication in the IVS contributes a relatively small heat load to the core. In addition, the staff notes that hazards present in traditional criticality safety problems are reduced or not present for the IVS because of its location within the reactor vessel as well as the capability to monitor and control reactivity in the reactor vessel. For these reasons, the staff determined the present IVS criticality information sufficient to satisfy 10 CFR 50.35. For the OL stage, the applicant should further demonstrate subcriticality of the combined core and IVS system with conservative methods. In addition, the applicant should ensure that the burnup of fuel assemblies while in IVS storage, though expected to be small, is accounted for in the source term methodology and analysis at the OL stage.

For the SFP calculation, the methods and preliminary analysis described in PSAR section 3.14 and reviewed through audit are sufficient for the CP stage. The staff noted that it was not fully clear how all credible abnormal conditions, such as dropped fuel assemblies and shifting due to seismic events, were bounded by the scenarios analyzed. The staff expects that the applicant will provide clarification at the OL stage.

3.13.2 Conclusion

On the basis of its review, the staff has determined that the criticality safety methodology described in PSAR section 3.14 provides a reasonable approach for the criticality safety analyses needed to support PDC 62 and ensure that criticality is prevented. The staff determined that the preliminary evaluations described in PSAR section 3.14 provide an adequate preliminary analysis and evaluation of the design and performance of SSCs at the facility per 10 CFR 50.34 and 10 CFR 50.68. Those areas that require further evaluation prior to the issuance of an OL, per 10 CFR 50.35, are discussed above.

3.14 References

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4 INTEGRATED EVALUATIONS

This chapter of the safety evaluation (SE) describes the staff's review and evaluation of Kemmerer Power Station Unit 1 (KU1) preliminary safety analysis report (PSAR) chapter 4, which provides integrated evaluations, including a summary of overall plant risk performance against the metrics contained in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision (Rev.) 1 (ML19241A472), and a defense-in-depth adequacy evaluation.

The applicable regulatory requirements for the evaluation of the control of occupational dose are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of applications; technical information," paragraph (a), "Preliminary safety analysis report,"; and
- 10 CFR 50.35, "Issuance of Construction Permits"

The applicable guidance for the evaluation of the integrated evaluations is as follows:

- Regulatory Guide (RG) 1.233, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (Agencywide Documents Access and Management System (ADAMS) Accession No.: ML20091L698) which endorsed NEI 18-04 with clarifications.
- RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (ML23269A222) which endorsed NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology," Rev. 1 (ML22060A190), with clarifications and additions.

4.1 Overall Plant Risk Performance Summary

Section 4.1 of the PSAR summarizes the preliminary evaluation of the overall plant risk to public health and safety from plant operation. The PSAR discussion compares analysis results to the cumulative risk metric targets described in NEI 18-04. PSAR section 4.1 describes the cumulative risk metric results of the preliminary integrated performance assessment performed for this purpose and compares them to the NEI 18-04 cumulative risk metric targets. As stated in PSAR section 4.1, the integrated performance assessment uses the same considerations, methodologies, and analytical tools described in PSAR section 3.1. PSAR section 4.1 includes assessment of other quantified events (OQEs) from the probabilistic risk assessment (PRA) in addition to the non-design basis accident (DBA) licensing basis events (LBEs) described and evaluated in PSAR chapter 3.

Section 4.1 of NEI 21 07 states that this section describes the integrated plant performance for the three cumulative plant performance metrics identified in NEI 18 04, section 3.2.2, Task 7b, "Evaluate Integrated Plant Risk Against QHOs and 10 CFR 20." The evaluation of overall risk

includes anticipated operational occurrences (AOOs), design basis events (DBEs), and beyond design basis events (BDBEs).

4.1.1 Technical Evaluation

RG 1.233, which endorses NEI 18-04, with clarifications, both describe the three cumulative risk metrics that need to be met to evaluate the integrated plant performance in any application based on the Licensing Modernization Project (LMP) methodology. As described in NEI 18-04, an applicant using LMP evaluates the integrated risk of all LBEs against three cumulative risk metrics. The first cumulative risk metric is based on the annual radiation dose limit for individual members of the public as specified in 10 CFR 20.1301(a)(1). The other two cumulative risk metrics ensure that the U.S. Nuclear Regulatory Commission (NRC) safety goal quantitative health objectives (QHOs)¹⁵ are satisfied. These three cumulative risk metrics as defined in NEI 18-04 are:

- The total mean frequency of exceeding a site boundary dose of 100 mrem from all LBEs should not exceed one per plant-year.
- The average individual risk of early fatality within one mile of the exclusion area boundary (EAB) from LBEs, based on mean estimates of frequencies and consequences, should not exceed 5×10^{-7} per plant-year.
- The average individual risk of latent cancer fatalities within 10 miles of the EAB from all LBEs, based on mean estimates of frequencies and consequences, should not exceed 2×10^{-6} per plant-year.

The staff compared the cumulative risk metrics in PSAR section 4.1 to the guidance in NEI 18-04 and determined that the metrics were consistent with the exception that OQEs from the PRA were included in the integrated plant performance evaluation in addition to the LBEs required in the LMP methodology. The staff determined that this deviation was acceptable because the content of information in PSAR section 4.1 is consistent with the guidance for the content of construction permit (CP) applications found in section C.4.1 of NEI 21-07 as endorsed by RG 1.253 with certain clarifications, and the inclusion of the OQEs adds additional information in the integrated plant performance evaluation. Further consideration of the inclusion of OQEs in the analysis is discussed later in this section.

As stated in RG 1.253, section C.4.1 of NEI 21-07 provides an acceptable method for developing information related to the integrated risk evaluation and describes that the safety analysis report should include an overall plant risk performance summary and describe margins between predicted plant performance and risk targets. The overall plant risk performance summary and evaluation results are provided in PSAR sections 4.1.1 through 4.1.3. The overall plant risk performance results in these sections are given as mean values without uncertainties. The staff determined this was an acceptable approach because it is consistent with the guidance in NEI 18-04 for cumulative risk metrics. Additionally, the staff determined the results reported in PSAR sections 4.1.1 through 4.1.3 meet the overall plant risk performance acceptance criteria. Based on available margin to each of the cumulative risk metrics, the staff also determined the evaluations reasonably account for uncertainties.

¹⁵ "Safety Goals for Operations of Nuclear Power Plants; Policy Statement; Republication," Federal Register, Vol. 51, No. 162, August 21, 1986, pp. 30028-30033 (51 FR 30028).

The staff also reviewed the technical basis for the calculation of overall integrated risk described in the PSAR. The staff's review considered the PSAR chapter 3 subsections that include the description of the PRA, the evaluation of LBEs, and the methods to estimate event frequencies and consequences. As stated in PSAR section 4.1, the integrated risk assessment included the AOOs, DBEs, and BDBEs, but did not include the DBAs, which are evaluated deterministically. The staff compared the selected events for the evaluation and the guidance in NEI 21-07 section C.4.1 (which is endorsed by RG 1.253) and determined the selected events were consistent. The integrated risk assessment includes only internal event LBEs, which is consistent with the state of PRA development for the CP application as described in PSAR section 3.1. The integrated risk assessment provided in the final safety assessment report (FSAR) for the OL application should be updated to include all modes, all hazards, and all sources. As described in SE section 3.1, the staff determined that the scope, level of detail, and quality of the KU1 PRA are reasonable for the CP application. Therefore, the staff determined that the preliminary integrated risk assessment acceptably uses a PRA which is consistent with the overall PRA acceptability for use in the NEI 18-04 process for the PSAR. As described in SE section 3.3 the staff's review determined that the LBE selection and categorization methodology and analysis methodologies adequately supports the issuance of a CP.

The applicant deviated from the NEI 18-04 methodology for the integrated risk assessment by including OQEs. The applicant stated OQEs are events beyond the LBE frequency cutoff (i.e., with mean frequency less than 5×10^{-7} per plant-year) with quantified consequences. The OQEs provide additional information into the integrated risk assessment regarding radiological release events with very low likelihood. PSAR section 4.1 states that for the OQEs which include sodium pool boiling, the consequence analyses used a local sodium pool boiling source term based on the key assumption that bulk sodium boiling does not occur. The discussion in PSAR section 4.1 acknowledges that this key assumption may underestimate the total calculated cumulative risk results given the analysis which includes these OQEs in the events. However, the PSAR states that the overall integrated risk assessment results are conservative as compared to what would be the results of the NEI 18-04 methodology, which does not include events below the LBE frequency cutoff. The integrated risk results for all non-DBA LBEs and OQEs reported in PSAR section 4.1 demonstrate a large margin to each of the cumulative risk metrics. The staff determined that including OQEs in the integrated risk assessment, even with the potential underestimated results for some of the OQEs, provides additional information than the NEI 18-04 integrated risk assessment envisioned. Therefore, the staff determined that the key assumption of no bulk sodium boiling in the OQEs does not negatively affect the integrated risk assessment and evaluation against the cumulative risk metrics and provides information on the overall plant risk as necessary for the LMP process. As discussed above, the staff determined that the integrated risk assessment information provided in the PSAR is consistent with the applicable guidance documents RG 1.233 and RG 1.253 on the basis that PSAR section 4.1 is supported by analyses that include the entire range of potential radiological release events (other than DBAs) for the facility.

In evaluating the PSAR basis for concluding that the overall integrated risk methodology and analysis are acceptable, the staff confirmed the PSAR description of the integrated risk evaluation through audit (ML25302A443) of supporting documentation of the event-specific radiological source terms and consequence analyses. For event quantification, the event-specific source terms are developed through use of an approved methodology, then each source term is input to the separate approved consequence analysis methodology to estimate the event-specific consequence. The event-specific LBE radiological source terms are the same as used in the analyses supporting PSAR Chapter 3, while the OQE radiological source terms were developed using the same approved radiological source term methodology, NAT-9392-A,

“Radiological Source Term Methodology Topical Report” Rev. 0 (ML25211A271), as was used to develop source terms for the LBEs. The staff’s evaluation of the event-specific radiological source terms and the applicant’s implementation of the NAT-9392 methodology is discussed in section 3.3 of this SE.

Through the regulatory audit, the staff confirmed that the integrated risk assessment consequence analyses were performed consistent with the NAT-9391-A, “Radiological Release Consequences Methodology Topical Report,” Rev. 0 (ML25211A267) LBE evaluation model, including modeling of the site environs and population as stated in the PSAR. The analyses modeled a uniform population with no credit for emergency response actions consistent with the approved radiological release consequences methodology. The staff determined that this modeling of the population provides consequence results which are bounding for the actual population around the KU1 site. The analyses used generic meteorological data which are based on the Electric Power Research Institute Utility Requirements Document as described in NAT-9391-A, based on an assumption that the data is conservatively representative of the KU1 site location. The use of generic meteorological data in lieu of site-specific data in the LBE evaluation model is the subject of a limitation and condition on use of NAT-9391-A. The staff’s evaluation of the PSAR implementation of NAT-9391-A, including the acceptability of use of generic meteorological data is discussed above in SE section 3.3. Therefore, the staff determined that the PSAR-stage integrated risk assessment used acceptable methods to quantify the consequences. However, the staff notes that the consequence analyses supporting the integrated risk assessment provided in the FSAR for the OL application should include a specific justification demonstrating that the generic meteorological data and modeling of atmospheric dispersion are representative of the actual conditions at the KU1 site, to fully meet the limitation and condition on the use of NAT-9391-A.

As discussed above, the staff determined that the integrated risk assessment information provided in the PSAR is consistent with the applicable guidance documents RG 1.233 and RG 1.253 on the basis that PSAR section 4.1 is supported by analyses that use approved source term and consequence analysis methods with appropriate input and assumptions to quantify the consequences and provide mean risk results for the KU1 facility to compare to the NEI 18-04 cumulative risk metrics.

4.1.2 Conclusion

The staff reviewed the overall plant risk performance information described in PSAR section 4.1 and determined that the integrated risk assessment included the range of potential radiological release events for the facility, based on internal events, as consistent with the CP stage of review; used acceptable methods to quantify the consequences; and provided mean risk results to compare to the NEI 18-04 cumulative risk metrics, in accordance with the guidance in RG 1.233. The staff finds that the overall plant risk performance information provided in the PSAR is consistent with the guidance in RG 1.253 and NEI 21-07, and that it provides a complete, consistent, and integrated summary of the plant risk to public health and safety, given the preliminary design information. This is consistent with the overall PRA acceptability for use in the NEI 18-04 process for the PSAR. Therefore, the staff concludes that the overall plant risk performance information adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1) and (4) and 10 CFR 50.35.

4.2 Defense-in-Depth

Section 4.2 of the PSAR describes the methodology that the applicant used to assess the acceptability of the plant capability and programmatic measures that provide defense-in-depth (DID), and the results of the DID adequacy evaluation.

RG 1.233 endorses, with certain clarifications, the risk-informed and performance-based methods described in NEI 18-04 for assessing the adequacy of DID. In the approach described in NEI 18-04 and RG 1.233, the applicant evaluates the LBEs, which are described in PSAR chapter 3, using an integrated, risk-informed process to ensure DID adequacy. Structures, systems, and components (SSCs) relied on for DID are identified as plant capability DID. Programmatic DID measures (e.g., special treatments, operational programs) are also evaluated through this process, though limited information is expected at the CP stage.

RG 1.253 endorses, with certain clarifications, the guidance in NEI 21-07 regarding the content of applications for non-LWR designs that follow the risk-informed process described in NEI 18-04 and provides guidance regarding implementation of the NEI 18-04 methodologies. RG 1.253, Appendix A, also contains guidance regarding the scope and level of detail in the PRA analysis that is necessary to support a CP application. The NEI 21-07 guidance discusses the level of detail that needs to be in the PSAR to adequately summarize the results of the assessment of the adequacy of DID, recognizing the preliminary nature of the design and PRA analysis.

4.2.1 Technical Evaluation

The staff evaluated the information in PSAR section 4.2 against the applicable regulations and guidance discussed in the Regulatory Evaluation section above. The staff reviewed the applicant's DID evaluation to ensure consistency with the methodology specified in NEI 18-04 (as endorsed by RG 1.233) and documentation specified in NEI 21-07 (as endorsed by RG 1.253).

Because the CP stage involves preliminary design and analyses, and because programmatic measures have not yet been fully developed, the staff's review only included DID measures associated with the reactor design and full power, low power, and shutdown operating operations, as described in detail below. DID associated with other operating states, non-reactor SSCs, and programmatic measures will be reviewed at the OL stage. In addition, the staff considered RG 1.253, which endorses NEI 21-07 with clarifications, as it provides guidance on the DID adequacy information expected to be included in the PSAR.

4.2.1.1 Plant Capability DID

The staff reviewed USO's technical basis for the plant capability DID, described in PSAR sections 4.2.1. applicant stated that the plant capability DID evaluation provides for the following, consistent with section 5 of NEI 18-04:

- Margin to the frequency-consequence (F-C) target curve is maintained for all individual LBEs
- Frequency of DBEs and BDBEs is maintained below $1 \times 10^{-2}/\text{yr}$ and $1 \times 10^{-4}/\text{yr}$, respectively
- Individual risks are below the QHOs

- There is sufficient independence between defense lines (DLs)
- There is assurance against over-reliance on any single feature across DLs
- There is sufficient balance between prevention and mitigation in the DLs

NEI 21-07 additionally establishes that the plant capability evaluation should (1) address LBE margins for risk-significant LBEs, (2) evaluate the adequacy layers of defense, (3) assess reliance on single-feature, and (4) address the balance between prevention and mitigation features across defense layers. The applicant's PSAR addressed these items and the staff's evaluation below considers both the NEI 18-04 and NEI 21-07 guidance.

4.2.1.1.1 LBE Margin

NEI 21-07 section 4.2.1.1 states that applicants should provide the baseline margins established between the frequencies and consequences of individual risk-significant AOOs, DBEs, and BDBEs, and the F-C target. The guidance states that margins to the F-C target should be demonstrated for both mean and 95th percentile dose consequences.

PSAR section 4.2.1.1 discusses the LBE margin evaluation. Table 4.2-1 of the PSAR provides the margins from the mean frequency and dose to the F-C target curve for LBEs identified as risk-significant based on the mean risk. PSAR table 4.2-2 provides the margins from the 95th percentile frequency and risk to the F-C target curve for LBEs identified as risk significant based on the 95th percentile risk. USO reported no risk-significant AOOs but identified several risk-significant DBEs and BDBEs. The tables show that the risk-significant LBEs maintain margin to the F-C target curve.

Based on this, the staff determined that margin evaluation is consistent with the guidance in NEI 21-07 by demonstrating that risk-significant LBEs maintain margin to the F-C target curve. The methods used to determine the frequencies and consequences of the LBEs are evaluated in chapter 3 of this SE.

4.2.1.1.2 Layers of Defense Evaluation

Section 4.2.1.2 of NEI 21-07 states that applicants should discuss the layers of defense relative to the guidelines provided in NEI 18-04 table 5-2 and discuss any deviations. PSAR section 4.2.1 describes the five DL to which functions are assigned:

- DL1: Plant design features that reduce or eliminate postulated initiating events.
- DL2: Functions that control AOOs and prevent DBEs.
- DL3: Functions that control DBEs and prevent BDBEs.
- DL4: Functions that control severe plant conditions and mitigate the consequences of DBEs.
- DL5: Functions associated with offsite protective measures, which currently include only post-accident monitoring.

The staff compared the defense layer definitions provided in the PSAR and found them to be consistent with the NEI 18-04, table 5-2 guideline and thus to be acceptable.

The staff identified through review of the information in the PSAR that the assignment of functions to defense layers is related, in part, to their safety classification. The PSAR presents

DL3 functions as exclusively safety related while DL2 and DL4 functions are a mix of non-safety-related with special treatment (NSRST) and non-safety-related with no special treatment (NST). DL1 functions are not evaluated for their safety significance; section 5.1 of this SE discusses this relative to the LMP guidance. However, the staff also notes that the applicant has developed a process to address the safety classification of these functions for the OL application, as discussed in SE section 5.1.

All DL1 functions are listed in table 4 of NAT-4770, "Natrium Demonstration DID Evaluation Report," Rev. 1 (ML25205A087), which was submitted as a supplement to the CP by the applicant and reviewed by the staff for context on the SSC descriptions, safety classifications, and DID evaluation. The staff reviewed the DL1 functions and features listed in NAT-4770 against the SSC descriptions provided in chapters 1 and 7 of the PSAR. The staff determined that the system descriptions provided in chapters 1 and 7 of the PSAR appropriately include these DL1 features, thereby ensuring they are part of the design basis.

NEI 21-07 also states that applicants should provide a summary of the layers of defense for each risk-significant LBE and describe the extent of independence among the layers. It also establishes that applicants should address the quantitative and qualitative guidelines for each layer provided in NEI 18-04 table 5-2.

For DL1, the quantitative and qualitative guideline is to maintain frequency of plant transients within designed cycles and meet owner requirements for plant reliability and availability. However, there is also a note that the DL1 features contribute to the protective strategies and should be addressed by design-specific targets. For DL2, the quantitative guideline is that functions should be identified for risk-significant AOOs that serve to keep the frequency of DBEs below 1×10^{-2} /plant-year. Phrased differently, if a function identified in this process had a higher failure rate, an LBE currently identified as a DBE would cross into the AOO region. Similarly, for DL3, functions should be identified for risk-significant DBEs that serve to keep the frequency of BDBEs below 1×10^{-4} /plant-year. For DL4 and 5, the quantitative guideline is to identify those functions relied on for BDBEs to maintain individual risks less than the QHOs.

The DL1 functions are discussed more above, where the staff noted that they will be further dispositioned at the OL. For DL2, no risk-significant AOOs are identified in the application. PSAR table 4.2-3 provides a list of the functions relied upon in the risk-significant DBE evaluations to maintain frequency of BDBEs less than 1×10^{-4} /plant-year. This table identifies two DL3 functions, both related to the primary functional containment boundary, that maintain the frequency of BDBEs less than 1×10^{-4} /plant-year. PSAR table 4.2-4 provides a list of the functions relied on in the evaluation of the risk-significant BDBEs to maintain individual risks less than the QHOs. This table identifies one function needed to maintain individual risks less than the QHOs. The staff determined that the information provided relative to the quantitative guidelines is consistent with the guidance in NEI 21-07 and is thus acceptable.

NEI 18-04 table 5-2 also provides qualitative guidelines for the layers. For DL2, the qualitative guideline is to minimize the frequency of challenges to SR SSCs. While no risk-significant AOOs were identified, the staff also reviewed all of the AOOs in chapter 3.5 of this SE and determined that the DL2 functions relied on act to minimize challenges to SR SSCs – for example, intermediate air cooling in the non-passive mode is relied on for decay heat removal so as not to rely on the SR reactor air cooling system. For DL3 and DL4, the qualitative guidance relates to avoiding reliance on a single feature in meeting the quantitative DID objectives. PSAR tables 4.2-3 and 4.2-4 provide a column stating that there are no single design or operational features relied on for each LBE family. The staff reviewed the LBEs identified and confirmed that multiple

functions are available to meet the quantitative objectives – for example, for RFH-FDIV-1 which relies on the primary functional containment boundary, the head access area is also available as a radionuclide retention barrier in DL4. As such, the staff determined the information provided is consistent with the guidance in NEI 21-07 and thus acceptable.

As for independence between the layers, PSAR section 4.2.1 provides information on how the applicant structured the DLs to include diverse and independent means of mitigating LBEs in separate DLs. The method used by the applicant explicitly factors in common-cause failure (CCF). The staff noted that in this approach, DL2 and DL4 are independent from DL3, but DL2 and DL4 may include common systems and may not be fully independent. The staff reviewed this approach and determined that it will result in multiple independent layers of defense available for AOOs (either DL2 and DL3 or DL3 and DL4). The staff also determined that multiple independent layers of defense will be available for DBEs (either DL2 and DL3 or DL3 and DL4), unless the DBE is initiated by a CCF. If the CCF is in DL2 or DL4, only DL3 may be available (though DL2 or DL4 functions may be available if not failed due to the CCF). If the CCF is in DL3, only DL4 may be available (though DL2 may be available if not assumed failed in the analysis). The staff determined these exceptions are reasonable because they ensure at least one independent function is available beyond the initiating CCF. DL4 alone is typically relied on for BDBEs, unless DL2 or DL3 functions are available. This is consistent with the definition of DL4 in NEI 18-04 for controlling severe plant conditions and mitigating the consequences of BDBEs, and is thus acceptable to the staff. As such, the staff considers the overall approach to DL independence to be acceptable.

The applicant performed a detailed evaluation of the layers of defense for all LBEs (not just risk-significant ones) in NAT-4770. Since different events in an LBE family credit different combinations of SSCs to prevent or mitigate the transient, the evaluation considered layers of defense across the LBE family. The staff reviewed the layers of defense credited for each LBE and found them to be consistent with the safety analysis as documented in PSAR chapter 3, as noted for each LBE in the text in SE sections 3.5 through 3.8 and the corresponding rows in the tables in each section which document the credited SSCs. Although some event sequences assume failure of some or all available functions, which is reasonable based on their modeling in the PRA, the staff confirmed that multiple, independent layers of defense remain available for each LBE with respect to all applicable fundamental safety functions. There is one exception, which is discussed below in SE section 4.2.1.1.3.

4.2.1.1.3 Single Feature Reliance

Section 4.2.1.3 of NEI 21-07 states that applicants should evaluate dependence on a single feature for risk-significant LBEs and DBAs. The applicant should also discuss both the method used to perform this evaluation as well as any special treatments applied to avoid over-reliance on a single feature across multiple layers of events.

PSAR section 4.2.1.3 summarized the DL function evaluation and identified no single feature that is relied upon for the risk significant LBEs. The staff reviewed the evaluation in NAT-4770 and determined that there is no over-reliance on a single function for the risk-significant LBEs, as discussed above with respect to the layers of defense evaluation.

The applicant's evaluation identified a DBA—RRS-ISPL-CN, a release from the intermediate sodium processing system—that is reliant on a single layer of defense to avoid a release. The function relied on is DL2-RR8, the intermediate cold trap sodium processing system barrier, which is assumed to fail as the initiating event in the DBA. However, because the DBA stays

within the 10 CFR 50.34 dose acceptance criterion, the applicant did not identify any additional layers of defense needed to provide adequate DID.

NEI 18-04 indicates in table 5-2 that “reliance” in the context of a DID adequacy evaluation is evaluated against the quantitative objective, i.e., the F-C target curve for LBEs and the dose criterion for DBAs, as well as the integrated risk measures. Because the F-C target is met for the DBE and the dose criteria are met for the DBA as discussed in PSAR sections 3.7.3.3 and 3.9.4.2 and sections 3.6.1.3 and 3.8.1.4 of this SE, the staff does not consider the function to be relied on for the purposes of the DID adequacy evaluation. Accordingly, the staff determined that USO’s single-feature reliance evaluation is acceptable.

4.2.1.1.4 Prevention-Mitigation Balance

Section 4.2.1.4 of NEI 21-07 states that applicants should evaluate the balance of preventative and mitigative features across layers of defense for all risk-significant LBEs and DBAs. The applicant should also discuss both the method used to perform this evaluation as well as capabilities or programmatic actions added as a result of the evaluation. Additionally, NEI 21-07 refers to section 5.7 of NEI 18-04 which states that the DID evaluation should confirm that a balance between event prevention and mitigation is reflected in the layers of defense for risk-significant LBEs.

PSAR section 4.2.1.4 states that the applicant performed an evaluation to ensure prevention and mitigation DL functions exist for risk-significant LBEs and DBAs, by identifying DLs that are available to prevent or mitigate each event. The defense layer evaluation discussed in PSAR section 4.2.1.3 and provided in NAT-4770 was used to identify preventative and mitigative functions available for each LBE. The applicant stated that prevention-mitigation balance was generally confirmed by ensuring DL1 features are available to prevent events, and that functions are identified for each fundamental safety function for each risk-significant LBE. The staff determined that this is an acceptable approach to preventative-mitigative balance because it ensures that features are available to prevent initiating events from occurring in the first place, and also ensures functions are available across multiple DLs to prevent LBEs from progressing to more severe accidents.

4.2.1.1.5 Conclusion Regarding Plant Capability DID

The staff determined that the information provided in the PSAR is consistent with the guidance documents and that the proposed DID methodology and plant capability DID measures are acceptable on the basis that they are consistent with NEI 18-04 and NEI 21-07, as endorsed by RG 1.233 and 1.253, respectively. As discussed in the previous sections above, the staff’s evaluation determined that the DID methodology has been properly applied and results in a reasonable level of plant capability DID, given the current available design information.

4.2.1.2 Programmatic DID

The applicant did not discuss programmatic DID in detail in the PSAR; however, PSAR section 4.2.2 identified that the guidelines for programmatic DID adequacy provided in section 5.8 of NEI 18-04 were evaluated and included in the design and plant programs in PSAR chapters 6, 7, and 8. Those programs are evaluated where applied and in chapter 8 of this SE report. PSAR section 4.2.2.1 states that guidelines for evaluating significant uncertainties were included in the integrated decision-making process (IDP). The IDP is

discussed in further detail in chapter 3 of this SE report. PSAR section 4.2.2.3 states that plant-specific programs for monitoring NSRST SSCs are identified in PSAR chapter 7. These programs are discussed in chapters 7 and 8 of this SE.

RG 1.253 section C.5 states that the NRC anticipates that the DID discussion at the CP stage may be limited to plant capabilities because programmatic capabilities may not have been established yet. As such, the staff determined that the limited detail provided in the PSAR on programmatic DID is reasonable.

4.2.2 Conclusion

The staff reviewed the USO's DID evaluation presented in PSAR section 4.2 and supporting document NAT-4770. The staff conducted its review in accordance with the guidance in RG 1.233, which endorses NEI 18-04 with clarifications, and RG 1.253, which endorses NEI 21-07 with clarifications.

As discussed in the preceding sections of this SE, the staff verified that the applicant applied the DID methodology consistent with NEI 18-04 and NEI 21-07 and addressed all relevant elements, including (1) LBE margin to the F-C target curve, (2) sufficiency of layers of defense, (3) evaluation of potential single-feature reliance, and (4) balance between preventive and mitigative functions.

Based on its review, the staff concludes that the USO's DID evaluation is consistent with the endorsed guidance and demonstrates sufficiency of DID based on the preliminary design information provided in the CP application for this stage of review. The staff will perform a more detailed evaluation of DID adequacy, including programmatic measures, at the OL stage.

4.3 Integrated Decision Process and Integrated Decision Process Panel

PSAR section 1.3.1 and 4.2.3 describe that the applicant used the LMP methodology, as described in NEI 18-04 and endorsed by RG 1.233, to establish a structured risk-informed, and performance-based framework for the selection of LBEs, classification of safety functions, and determination of the appropriate classification and performance requirements for SSCs. One of the key features of this methodology is the use of the IDP, implemented through a multidisciplinary integrated decision process panel (IDPP), to ensure that the risk-informed, performance-based approach is consistently and appropriately applied. NEI 18-04 states that the IDP should be conducted by a knowledgeable, multidisciplinary panel, and that the process should be traceable, documented, and consistent with the LMP framework.

Furthermore, the IDP plays a key role in the DID evaluation required by the LMP methodology. The DID evaluation ensures that plant design and operational strategies provide multiple, independent, and layered means to protect public health and safety, consistent with NRC safety philosophy and the Commission's safety goal policy statement.

Specifically, the IDP is a deliberative decision-making process that integrates deterministic requirements, PRA insights, and engineering judgment to make informed and traceable decisions on safety classification, performance criteria, and DID evaluation. The IDPP is the decision-making body responsible for executing the IDP. Members of the IDPP typically include subject matter experts in PRA, design engineering, systems engineering, operations, human factors, and licensing, as well as decision facilitators familiar with the LMP process.

4.3.1 Technical Evaluation

The staff reviewed the applicant's description of the IDP and IDPP implementation in the PSAR and audited supporting documentation to confirm appropriate implementation of the IDP and IDPP. For context, the audited supporting documentation is listed below:

- NAT-9034, Rev. 0, "IDPP Report Natrium License Basis Events, SSC Classification and Defense in Depth"
- NAT-5535, Rev. 0, "Natrium IDPP Charter"
- NAT-9030, Rev. 0, "IDPP Training Qualification"
- NAT-10919, Rev. 0, "Interim Process for Changes to IDPP-endorsed DID Baseline"
- NAT-10640, Rev. 0, "Inaugural IDPP Convocation Summary Report"

The staff's review focused on determining if the applicant's IDP approach is consistent with NEI 18-04 as endorsed by RG 1.233, and if the IDPP deliberations resulted in defensible, traceable decisions that support the preliminary licensing basis for the CP stage.

Selection and Confirmation of LBEs

The staff compared the PSAR description of event selection and confirmation of LBEs and the guidance in NEI 18-04 and determined that the applicant's approach was consistent with the guidance. Further, staff confirmed via the audit that USO's IDPP considered the appropriate aspects in the NEI 18-04 guidance (e.g., PRA event sequences and event categorization into AOOs, DBEs, and BDBEs consistent with the LMP process). Additionally, the staff confirmed the panel considered frequency estimates, consequence evaluations, and plant design features, and ensured that low-frequency but potentially high-consequence events were retained in the LBE set where appropriate.

Safety Function Identification and SSC Classification

By auditing the documents above, the staff confirmed that the IDPP reviewed and verified the identification of safety functions credited to prevent or mitigate LBEs. SSCs were classified as safety-related (SR), non-safety-related with special treatment (NSRST), or non-safety-related with no special treatment (NST) based on their role in supporting these functions and their risk significance from the PRA. The staff confirmed that the classification process followed the criteria in NEI 18-04 as stated in the PSAR, and the staff finds the PSAR justified for the CP stage.

Defense-in-Depth Evaluation

By auditing the documents above, the staff confirmed that USO used the IDP to perform a systematic DID evaluation in accordance with NEI 1804 section 5.9 as described in the PSAR and the staff finds the PSAR justified for the CP stage.

Documentation and Traceability

By auditing the documents above, the staff confirmed that USO documentation was maintained in accordance with NEI 18-04 as described in the PSAR.

4.3.4 Permit Condition

The staff recommends that the CP include the following permit conditions:

Prior to submittal of the operating licensing application, USO shall notify the NRC within 30 days after an integrated decision-making process panel is convened for internal approval of the safety analysis, risk assessment, or defense-in-depth adequacy of the design. The notification shall include the schedule for when the supporting documentation will be available for NRC examination.

This condition ensures that the staff has the opportunity to verify that the USO's IDP, as executed by the IDPP, has been appropriately conducted in support of the LMP application. The IDP is a key element of the LMP framework, providing a structured, multidisciplinary approach for evaluating safety analysis, risk assessment, and DID adequacy. Requiring notification and availability of supporting documentation prior to submittal of the OL application allows the staff to examine whether the IDPP deliberations were traceable, balanced, and consistent with NEI 18-04 and RG 1.233. This early visibility ensures that regulatory expectations would be met, enhances transparency in the applicant's decision-making process, and provides confidence that safety-significant issues are addressed before the OL review stage.

4.3.5 Conclusion

Based on its review of the PSAR, the staff finds that USO's IDP and IDPP implementation for the KU1 CP application is consistent with the methodology described in NEI 18-04 as endorsed by RG 1.233. The IDPP composition included appropriate multidisciplinary expertise, and the process was implemented to ensure balanced consideration of deterministic, probabilistic, and engineering judgment inputs.

Therefore, the staff concludes that USO's IDP and IDPP implementation provides an acceptable basis for supporting the CP application in accordance with 10 CFR 50.34(a)(1) and (4) and 10 CFR 50.35, consistent with NRC-endorsed LMP methodology.

5 SAFETY FUNCTIONS, DESIGN CRITERIA, AND SSC SAFETY CLASSIFICATION

This chapter of the SE describes the U.S. Nuclear Regulatory Commission staff's (the staff's) review and evaluation of Kemmerer Power Station Unit 1 (KU1) PSAR chapter 5, which describes the safety classification of structures, systems, and components (SSCs), the safety-significant probabilistic risk assessment (PRA) safety functions (PSFs), and the principal design criteria (PDC).

The applicable regulatory requirements for the staff's evaluation of the SSC safety classification, PSFs, and PDC, are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of applications; technical information," paragraph (a), "Preliminary safety analysis report,"
- 10 CFR 50.35, "Issuance of Construction Permits"
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

The applicable guidance for the staff's evaluation of the SSC safety classification, PSFs, and PDC, are as follows:

- Regulatory Guide (RG) 1.233, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (Agencywide Documents Access and Management System (ADAMS) Accession No.: ML20091L698) which endorsed NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision (Rev.) 1 (ML19241A472), with clarifications.
- RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (ML23269A222) which endorsed NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology," Rev. 1 (ML22060A190), with clarifications and additions.
- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," section 1.1.4, "Developing Proposed Principal Design Criteria (PDC) for Those Aspects of the Facility Design not Informed by the [Licensing Modernization Project] LMP Process (e.g., Normal Operations)" (ML23277A139).
- RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," (ML17325A611).

5.1 Safety Classification of SSCs

The regulations in 10 CFR 50.34(a)(1) and (4) together require that each CP applicant supply a safety assessment with a preliminary analysis and evaluation of the design and performance of facility SSCs. Specifically, 10 CFR 50.34(a)(1)(ii)(D) requires consideration be given to the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. The requirements of 10 CFR 50.34(a)(4) specify that the objective of the preliminary analysis and evaluation of the design and performance of SSCs is to assess the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and mitigation of consequences of accidents. These required assessments of the facility provide an understanding of those SSCs that are most important to radiological health and safety because they play a key role in preventing and mitigating accidents. Throughout the regulations in 10 CFR Part 50 “Domestic Licensing of Production and Utilization Facilities,” it is established that certain SSCs are safety-related (SR) (as defined in 10 CFR 50.2, “Definitions”) and these SSCs must be provided with augmented design standards, quality requirements, etc. as necessary to ensure their reliability and capability to prevent and mitigate accidents; other requirements may apply to SSCs that are safety-significant but not SR.

The guidance in NEI 18-04, as endorsed with clarifications in RG 1.233 provides an acceptable process for using the PRA and licensing basis event (LBE) analyses to classify an SSC based on the role it plays in prevention and mitigation of LBEs, as described in more detail in chapter 3 of the PSAR and this SE. Under this process, the most safety-significant SSCs are classified as SR, SSCs that perform safety-significant functions but are less safety-significant are classified as non-safety-related with special treatment (NSRST), and other SSCs are classified as non-safety-related with no special treatment (NST). These classifications are an integral part of the licensing basis development process described in NEI 18-04, because they are used to assess SSCs according to their safety- or risk-significance and identify if special treatments beyond normal industrial practices are needed to ensure SSC performance of safety functions in the prevention and mitigation of LBEs.

Depending on the design, the SSCs classified as SR under the NEI 18-04 methodology are not necessarily the same as the SSCs that meet the definition of SR in 10 CFR 50.2. In addition, as relevant to the KU1 design, the definition of SR SSC in 10 CFR 50.2 includes components relied upon to remain functional during certain events to maintain the integrity of the reactor coolant pressure boundary, where reactor coolant pressure boundary is defined specifically for PWRs and BWRs. KU1 is neither a pressurized water reactor or a boiling water reactor. As such, Appendix B evaluates an exemption from the substantive requirements in 10 CFR Part 50 that refer to SR SSCs. This exemption, which the staff determined meets the exemption criteria in 10 CFR 50.12, “Specific exemptions,” instead applies the NEI 18-04 definition to the substantive requirements in 10 CFR Part 50. As the staff determined this exemption meets the relevant criteria, that is the definition the applicant used in its submissions relevant to this chapter and that the staff used when evaluating those submissions.

Section 5.1 of the PSAR states that the safety classification of SSCs was performed in accordance with NEI 18-04 as endorsed in RG 1.233. There are multiple steps in the SSC

classification process as described in sections 3 and 4 of NEI 18-04, and each one results in a different combination of risk importance and safety classification. Table 5.1-1 of this SE provides a staff summary of these steps, including the scope of LBEs to be evaluated, the parameters used in the evaluation, and the resulting risk importance category and classification for the functions and SSCs that perform those functions.

The staff reviewed the applicant's SSC safety classification process implementation to verify it was consistent with NEI 18-04 as endorsed in RG 1.233. As stated in section 5.1 of NEI 21-07, the applicant should provide a description of the specific way the LMP methodology was applied within the range of options specified in NEI 18-04. However, NEI 21-07 does not include guidance specific to the level of detail expected at the CP stage beyond noting that details of the analyses should be present in plant records. The description provided by USO in section 5.1 of the PSAR is very high level, confirming that the SSC safety classification process follows the LMP methodology and stating that safety significance of the PSF determines safety classification and a structured approach is used in identifying the SSCs that perform the PSFs functions. The staff found this level of detail, in combination with the description of the DID evaluation methodology in PSAR section 4.2, to be acceptable for the CP stage, and the staff was able to confirm the accuracy of these high-level statements through audit. However, the staff expects additional details on the specific way LMP was implemented to be provided in the FSAR to meet the guidance in NEI 21-07, as endorsed in RG 1.253.

The staff conducted an audit of USO's implementation of the LMP methodology as documented in the audit summary report (ML25302A443). The staff gained an understanding of the process used for SSC classification through audit questions, audit discussions, and review of USO's internal documentation. The staff audited relevant design basis documents to gain an appropriate level of understanding of the functions and the SSCs used to perform them. During the audit the staff was provided with a demonstration of the software used to evaluate the impact of each function or SSC on the LBEs. The staff also reviewed the inputs and outputs of the classification process, information that was in both the PSAR and audit documents, to evaluate whether the process was being executed as defined. The staff considered the radiological material in the various plant systems that would be available for release to understand the mitigative impact of the SSCs, which contributes to the safety classifications of the SSCs. This information also supported a risk-informed approach to the review, allowing the staff to focus on areas where failure of mitigating features could result in the highest consequences.

During its review of the PSAR, the staff identified certain areas of the NEI 18-04 guidance and corresponding endorsement in RG 1.233 that were interpreted differently by USO and the staff. Through audit discussions understanding was reached on these differing interpretations and how they may have impacted the SSC safety classifications provided in the PSAR. In areas where USO had unintentionally deviated from the LMP methodology, USO implemented changes to its process to align more closely with the LMP methodology as discussed in the staff's request for confirmation of information (RCI)-2 dated September 16, 2025 (ML25259A180). USO also provided a supplement dated September 10, 2025 (ML25253A386) entitled "Preventative Measures Classification Methodology and Preliminary Results," (the preventative controls supplement) which provides information on steps that were added to the DID evaluation to address SSC safety classification of preventative controls. Additional discussion on USO's implementation of the LMP methodology is provided below.

Identification of risk-significant PSFs

In response to the staff's RCI, USO confirmed, with one clarification as described in the RCI response, the following regarding its implementation of the LMP methodology:

The LMP methodology documented in NEI 18-04, as endorsed by RG 1.233, states that, "An SSC is classified as risk-significant if... a prevention or mitigation function of the SSC is necessary to meet the design objective of keeping all LBEs within the F-C Target." The methodology further clarifies that, "An LBE is considered within the F-C Target when a point defined by the upper 95th percentile uncertainty on both the LBE frequency and dose is within the F-C target." This evaluation is performed by assuming failure of the SSC in performing a prevention or mitigation function and checking how the resulting LBE risks compare with the F-C target.

These risk-significant functions are designated as required safety functions (RSFs) and result in safety related (SR) classifications if they are necessary to meet the F-C target for DBEs or high-consequence BDBEs when evaluated using mean risk values. The remaining risk-significant functions identified through the evaluation described above are classified as risk-significant NSRST functions. USO unintentionally deviated from the LMP methodology in two ways while performing these risk-significant function determinations for the CPA:

1. Identification of RSFs was performed using the 95th percentile versus mean risk values.
2. Identification of risk-significant NSRST functions needed to keep LBEs within the F-C target was not performed.

The first deviation is conservative and may have resulted in a small number of SSCs that should have been classified as risk-significant NSRST being classified as SR, reducing the impacts of the second deviation. Impacts of the second deviation are further reduced by an additional quantitative assessment step performed by USO as part of the defense in depth (DID) evaluation, beyond what is outlined in the LMP methodology. In this additional step, various sets of non-SR functions were evaluated, collectively with the SR functions, to identify the minimum set of additional non-SR functions that were needed to ensure that all LBEs were below the F-C target when assessed at the mean risk values. The set of additional non-SR functions were classified as NSRST for DID.

With this additional assessment step performed assuming all non-SR functions not in the evaluation set failed, this resulted in many of the SSCs that were not identified as risk-significant NSRST because of the second deviation being classified as NSRST for DID. With this and the use of the 95th percentile in identifying RSF, the staff determined that the impact of this second deviation on the overall design and facility risk at the CP stage is small. USO has modified their process and workflow to ensure that this step for identifying risk-significant NSRST functions is performed moving forward, in a manner consistent with the LMP methodology.

Safety classification for SSCs needed to reduce the likelihood of initiating events (IEs)

NEI 18-04 section 3.3.4 states that, “[a] PSF, as used in the LMP, is any function by any SSC modeled in the PRA that is responsible for preventing or mitigating a release of radioactive material from any radioactive material source within the plant.” Throughout NEI 18-04, where functions are discussed, phrases like “prevent or mitigate” or “prevention and mitigation” are used, indicating that both types of functions are important to ensuring safety and both should be evaluated at each step of the SSC classification process.

During its review of the PSAR, the staff identified that some preventative functions were not being evaluated through the SSC safety classification process. These preventative functions included interlocks on fuel handling movements, [[

]] USO noted in audit that these preventative functions were not viewed as PSFs within its process because they were incorporated into the PRA within the initiating event fault tree or not explicitly modeled within the PRA. This limited the evaluations that USO could perform in assessing the impact on risk, based on how the process was executed for the PSAR.

Following audit discussions, USO made two modifications to its process in order to ensure that preventative controls were addressed in the classification process. The first, as discussed by the staff and USO in audit discussions, was to evaluate preventative functions with a failure on demand as a PSF within the USO SSC safety classification process. This modification primarily impacted the interlock functions. The second was to add the following steps to the USO DID evaluation process, as described in the preventative controls supplement:

[[

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As identified by USO in the supplement, a preliminary assessment of the process indicated that these new steps would lead to SSCs, [

]] being classified as NSRST for DID adequacy.

This alternative process for the SSC classification of preventative functions limits the assessment of SSC classification for this subset of preventative control to the DID evaluation. The process in NEI 18-04 indicates that preventative functions serving to reduce the likelihood

of IEs are PSFs that should go through the same SSC classification process as mitigative functions. USO's alternative process has the potential to result in some preventative SSCs that could have been classified as risk-significant being classified as NSRST for DID because evaluations of whether the function is needed to keep LBEs below the F-C target are not performed. While the alternative process does have the potential to lead to less conservative safety classifications, the staff determined that it provides a reasonable method of ensuring safety based on the following:

- The threshold of **[[]]** is low relative to the F-C target in the DBE and BDBE region, which is likely to result in a conservative number of events being identified as requiring NSRST preventative controls.
- With functions that have a probability of failure on demand being fully evaluated by the SSC safety classification process, the number of potentially risk-significant SSCs impacted by this alternative approach is expected to be small.
- The integrated decision-making process (IDP) and IDP panel (IDPP) will review the roles of these preventative SSCs and evaluate where additional controls may be needed to meet DID adequacy or SSC classifications may need to be elevated for risk significance.

Based on these considerations, the staff determined that the alternative process provides a reasonable method of ensuring safety and additional review of the process. Its implementation, and the resulting impact on SSC classifications can be left to the OL stage based on the limited scope of SSCs impacted and the conclusions listed above.

NSRST SSCs credited at degraded performance level in DBA source term analyses

NEI 18-04 contains the following statements in sections 5.3 and 5.6.2:

- a. DBAs are then constructed, starting with each DBE, and then assuming that only the SR SSCs perform their associated RSFs.
- b. It is appropriate that for the inherent capabilities of passive functions, degradation of the passive function is considered, as opposed to complete failure (i.e., a physical non-existence of that function).

For SSCs with active functions, USO followed statement (a). However, based on USO's interpretation of statement (a) that NSRST SSCs may continue to function but not perform RSFs, and statement (b), USO incorporated some passive NSRST SSCs performing at a degraded level in their DBA source term analyses. The NSRST SSCs credited at a degraded level are generally large passive structures such as confinement barriers and buildings, where the ability to perform the NSRST function would not be impacted by the IE. A specific methodology for determining which NSRST functions may be credited with degraded performance in DBAs was not provided at the CP, but the staff was able to gain a preliminary understanding of the approach through audit discussion and expects a documented methodology to be available at the OL stage. Examples of these SSCs are discussed in section 3.8.1.4 of this SE for releases from ex-vessel systems. The staff determined that USO's interpretation of the guidance and approach to ascribing some amount of degraded performance to passive NSRST SSCs in DBA source term analyses was reasonable, but

acceptability is dependent on the design aspects of the SSCs and the level of degraded performance credited. Since the design is preliminary, a final determination on the acceptability of this approach to DBA source term analysis will be made at the OL stage when the methodology and complete scope of SSCs credited with degraded performance will be available for review.

Seismic interaction aspects of non-SR SSCs

NEI 18-04 states:

SR-classified SSCs are required to perform their RSFs following a Safe Shutdown Earthquake, NSRST and [non-safety-related with no special treatment] NST SSCs required to meet Seismic II/I requirements (required not to interfere with the performance of SR SSC RSFs following a Safe Shutdown Earthquake).

RG 1.233, which endorses NEI 18-04 with clarifications, states:

None of the non-SR with no special treatment SSCs are classified as safety significant, but they may have requirements to ensure that failures following a design-basis internal or external event do not adversely impact SR or NSRST SSCs in their performance of safety-significant functions.

NEI 21-07 states:

When a non-safety-related SSC is required to protect the SR SSCs in their ability to perform their RSFs, such non-safety-related SSCs are not necessarily NSRST. The NSRST classifications are based on the PRA Safety Functions these SSCs perform to prevent or mitigate event sequences and not functions that are focused on protecting the SR SSCs.

While the above statement from NEI 18-04 only speaks to the protection of SR SSCs from a seismic event, the statement from RG 1.233 notes that requirements may also be needed to protect the ability of NSRST SSCs to perform their safety-significant functions. Furthermore, RG 1.253, Item C.3(e) clarifies that NSRST SSCs credited in non-DBA LBEs or relied on to establish adequate DID may need to be specially designed to withstand or be protected from the design basis hazard levels (DBHLs) or beyond-design-basis hazards. Because the LMP methodology reduces the use of SR equipment by using NSRST items to mitigate events in the AOO, BDBE, and 95th percentile consequence regions as well as provide DID, widespread failure of NSRST from a seismic event could result in an unacceptably high consequence. For this reason, seismic interactions that may interfere with an NSRST SSC's ability to perform its safety-significant function must be evaluated and addressed appropriately based on seismic risk significance. USO describes how seismic risk significance is determined for NSRST SSCs in section 6.4.1.1 of the PSAR and addresses consideration of NSRST SSCs as seismic interaction targets in sections 6.1.1.1 and 6.4.1.5.2 of the PSAR, providing reasonable assurance that the safety-significant functions needed to ensure safety during and following a design basis seismic event can be performed. This is consistent with the guidance in RG 1.233 provided above, as determined in the staff's evaluation of USO's seismic interaction approach in SE section 6.1.1.3.

Note, while USO is identifying SSCs that are a source of safety concerns with respect to seismic interaction, USO is not including these considerations in their SSC safety classification which is consistent with the statements from RG 1.233 and NEI 21-07 provided above. However, NEI 21-07 also includes discussion for the consideration of “special safety functions” for NST SSCs. USO has expanded the scope of seismic interaction beyond support for systems or components located above or near SR and NSRST SSCs to buildings, including the reactor building (RXB) superstructure, and any component containing sodium. The staff’s evaluation of this is captured in SE sections 6.1.1.3 and 6.4.1.1.

NEI 21-07 section 6.1.3 indicates that design information for non-SR SSCs having DBHL-related design requirements to prevent adverse impact on the capability of SR SSCs in the performance of RSFs should be identified and described in the FSAR, including identification of the SSC being protected, the DBHL associated with the requirement, and the specific design requirement. To address the safety aspects of these “special safety functions” for NST SSCs, the staff requested the applicant describe the extent to which programs are applied to ensure the safety aspects are met. Details of this were confirmed during audit, and USO added statements on how programmatic controls are applied to seismic interaction sources to PSAR section 6.1.3.1. The staff also requested additional design information regarding the RXB superstructure, which was added to PSAR section 7.8. The associated staff evaluations are captured in SE sections 6.1.1.3, 7.8.1.1.10, and 7.8.1.2.

While seismic interaction aspects of NSRST and NST SSCs do not impact their safety classifications, the safety considerations that NEI 21-07 refers to as “special safety functions” and USO refers to as seismic interaction requirements are necessary to have a complete picture of how functions and SSCs are determined to be important from a safety standpoint and given appropriate design requirements. The staff determined that the seismic interaction approach evaluated in the SE sections referenced above in combination with the SSC safety classification methodology and design requirements for SR and NSRST SSCs provides reasonable assurance that SSCs will be able to perform the necessary safety-significant functions to prevent undue risk to the health and safety of the public during a seismic event, in accordance with 10 CFR 50.35 requirements. This will be verified through use of the seismic PRA at the OL stage.

5.1.1 Conclusion

Based on its review as documented above and supported by the General Audit, the staff determined USO’s SSC safety classification process meets the requirements in 10 CFR 50.34(a)(1), (4), and 10 CFR 50.35 and is acceptable for the CP application considering the following:

- Apart from the items discussed above, USO’s implementation of the LMP methodology is consistent with the guidance in NEI 18-04, as endorsed in RG 1.233. This methodology provides a process for identifying the safety-significant SSCs and the associated PSFs necessary for the prevention of accidents and the mitigation of the consequences of accidents to demonstrate an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products, as required by 10 CFR 50.34(a)(1), (4).

- The impacts of differences from the LMP methodology noted above on the preliminary design have been assessed and found to be small relative to the overall facility risk. Based on this, further design information resulting from the correction of these differences can reasonably be left for later consideration and reviewed at the OL stage consistent with 10 CFR 50.35(a)(2). In accordance with 10 CFR 50.35(a)(4) the information provided in the PSAR and through supplemental information provides reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) that the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

Table 5.1-1: SSC Classification Steps in NEI 18-04 Methodology

Task*	LBEs Evaluated	Evaluation	Criteria Compared Against	Values Used	Resulting SSC Classification and risk importance category
7a.1	DBEs and AOOs and BDBEs with uncertainty bands that extend into the DBE region	Remove individual functions and evaluate impact on individual LBEs	F-C target curve	Mean	SR RSF, risk-significant and safety-significant
7a.2	BDBEs with mean consequences that exceed 25 rem TEDE	Remove individual functions and evaluate impact on individual LBEs	F-C target curve	Mean	SR RSF, risk-significant and safety-significant
7d	DBAs	Conservative consequences evaluation in which only available SR SSCs perform their function	10 CFR 50.34 dose limits	95 th percentile or greater	SR RSF, safety-significant
7a.3	All non-DBA LBEs	Remove individual functions and evaluate impact on individual LBEs	F-C target curve	95 th percentile	NSRST, risk-significant and safety-significant
7b	All non-DBA LBEs	Remove individual functions and evaluate total frequency of all associated LBEs	1% of cumulative risk metrics	Mean	NSRST, risk-significant and safety-significant
7e	All non-DBA LBEs	Perform DID adequacy evaluation using the IDP and IDPP	Table 5-2 and other DID concepts described in Section 5 of NEI 18-04	N/A	NSRST, safety-significant

* From section 3.2.2 of NEI 18-04. Note, task 7c of determining risk significance is an assessment of the results determined in steps 7a and 7b

5.2 Safety-Significant PRA Safety Functions

As discussed in SE section 5.1, 10 CFR 50.34(a)(4) requires, in part, a preliminary analysis and evaluation of the design and performance of SSCs and their adequacy in the prevention and mitigation of accidents. USO performs this evaluation using a methodology that is consistent with NEI 18-04, with exceptions described in section 5.1 of the SE. In the NEI 18-04 process, as shown in figure 1.3-1 of this SE, the safety-significant PSFs are identified and then SSCs are classified based on the PSFs they perform. The category of the function and the SSC classification are based on which step of the process the safety-significant function was identified. See SE table 5.1-1 for a summary of these steps as described in NEI 18-04.

The information in section 5.2 of the PSAR is the output of the SSC classification process: the safety-significant functions, the SSCs performing those functions, and the classifications of those SSCs. The technical evaluation of this information has been divided in this SE to more closely align with the recommended structure in NEI 21-07, with RSFs and SR SSCs (tables 5.2-1 through 5.2-3 of the PSAR) evaluated in section 5.4 and NSRST functions and SSCs (table 5.2-4 of the PSAR) evaluated in section 5.5 of this SE. Section 5.5 of this SE also includes a summary of radionuclide retention barriers (SR and NSRST). This section of the SE focuses on the summary information provided in PSAR section 5.2 and overall conclusions.

The staff reviewed PSAR section 5.2 to evaluate if the preliminary list of safety-significant PSFs, in PSAR tables 5.2-1 through 5.2-4, are consistent with NEI 18-04 and NEI 21-07 as endorsed in RG 1.233 and RG 1.253; or, if deviations were used, that the applicant provided a reasonable basis. As part of its evaluation, the staff conducted an audit that involved relevant plant design basis documents and information associated with how probabilities, mechanistic source terms (MSTs), and radiological consequences were calculated. The applicant's responses to audit questions pertaining to the PSFs as documented in the General Audit Report and audit of underlying SSC classification documentation supported the staff in confirming the PSAR information.

The PRA described in section 3.1 of the PSAR was used in the preliminary determination of safety-significant PSFs. As described in section 3.1 of the SE, due to the preliminary nature of the design, some SSCs are not currently modeled within the PRA. Because the SSC safety classification process relies on the results of the PRA to determine classification and risk significance, these SSCs will need to be incorporated in the PRA prior to the OL application to confirm that the appropriate SSC safety classification and risk significance was assigned. At this preliminary design stage, USO identified these SSCs through the DP and included them in PSAR table 5.2-4. In section 5.2.2 of the PSAR, USO committed to performing safety analysis evaluations to confirm the risk-significance of all non-SR PSFs at the OL stage.

The staff reviewed the preliminary safety-significant PSFs described in PSAR section 5.2 and determined that the results are reasonable based on the preliminary PRA, LBEs described in chapter 3 of the PSAR, the SSC safety classification process described in section 5.1 of the PSAR as evaluated in section 5.1 of this SE, and the preliminary design available at this time.

5.3 Principal Design Criteria

The regulations in 10 CFR 50.34(a)(3)(i), require a CP applicant to include in the PSAR the PDC for the facility. The PDC for KU1 are incorporated by reference into the PSAR via topical report (TR), NATD-LIC-RPRT-0002-A, "Principal Design Criteria for the Sodium Advanced Reactor," Rev 1 (ML24283A066). PSAR section 5.3, lists the PDC from the TR and provides a

brief description of how each PDC is addressed by the design. The staff found the PDC for the Natrium plant to be acceptable, as documented in its SE included with the approved version of NATD-LIC-RPRT-0002-A. The staff's conclusions in the SE were subject to two limitations and conditions (L&Cs):

1. An applicant or licensee referencing this TR must propose a design that is substantially similar to the Natrium design as discussed in SE section 1, or otherwise justify that any departures from these design features do not affect the conclusions of the TR and this SE.
2. The use of this TR is restricted to those applicants using the risk-informed, performance-based licensing process described in NEI 18-04, Rev. 1, as endorsed by RG 1.233. Because the proposed PDCs may not fully address all performance requirements for SSCs defined as safety-significant under the NEI 18-04 process, applicants or licensees referencing this TR must augment the PDC in the TR with appropriate PDC for any SR or NSRST SSCs whose safety function relates to BDBEs, or NSRST SSCs needed for DID adequacy, or otherwise justify that the Natrium PDCs as described in the subject TR are adequate.

The staff reviewed the design information presented in the PSAR, particularly in chapters 1, 3, and 7, and determined it is consistent with the design as described in NATD-LIC-RPRT-0002-A and section 1 of the SE that evaluates the TR. Consistent with the TR and the staff's SE evaluating it, the design presented in the PSAR is (1) a pool-type, metal-fueled SFR where the reactor heats sodium in the primary heat transport system and transfers the heat via an integral heat exchanger to the intermediate sodium loop; (2) the intermediate loop transfers the heat to sodium-salt heat exchangers where it heats a molten salt loop; and (3) molten salt is pumped between the sodium-salt heat exchangers and the energy island where it can be stored and converted to electricity. Based on this, the staff determined L&C 1 is addressed. The staff also identified that the KU1 application uses the NEI 18-04 process as described throughout this chapter of the SE and elsewhere in this SE, so this aspect of L&C 2 is adequately addressed.

The second aspect of L&C 2 relates to defining PDC associated with performance of safety functions for BDBEs or as needed for DID adequacy. The staff reviewed the PDC in section 5.3 of the PSAR and noted that USO did not identify any additional PDC. However, the staff reviewed PSAR tables 5.2-1 through 5.2-3 and found that they identify SR functions needed to mitigate BDBEs and link them to performance of safety-related design criteria (SRDC) for these events.

PSAR table 5.2-4 identifies NSRST functions and indicates whether each is risk-significant or required for DID adequacy. It also identifies the SSCs needed to perform each function. The staff reviewed the safety-significant SSC descriptions in chapter 7 of the PSAR and identified that each section appropriately references the safety-significant functions supported by the SSC, including those that are NSRST (regardless of whether they are risk-significant or needed for DID adequacy), and provides performance requirements for the SSC to provide support for each function.

The staff's SE for NATD-LIC-RPRT-0002-A stated that "the NRC staff expects that any additional PDCs needed would be established with a minimum scope and content similar to that discussed in NEI 21-07, Section C, Section 5.6." This section of NEI 21-07 states that the complementary design criteria, which are intended to provide performance requirements similar to the PDC for NSRST SSCs, may "be defined at the functional level (related to the PSFs that

are satisfied by the NSRST SSCs).” Because the performance requirements provided in PSAR chapter 7 describe functional performance necessary to support the NSRST functions, the staff determined that these performance requirements are consistent with the concept of complementary design criteria as discussed in NEI 21-07, and therefore meet the staff’s expectations articulated in the SE. Because the staff determined above that performance of SSCs in BDBEs are appropriately tied to SRDC in PSAR tables 5.2-1 through 5.2-3 and performance of NSRST SSCs are appropriately tied to additional criteria in PSAR chapter 7, the staff determined the second aspect of L&C 2 is appropriately addressed.

Based on staff determinations above, and its SE as incorporated by reference from NATD-LIC-RPRT-0002-A, the staff determined that the information regarding the PDC in PSAR section 5.3 is sufficient and meets the guidance in NEI 21-07 and regulatory requirements in 10 CFR 50.34(a)(3)(i). The table below describes where in the SE SSCs are evaluated relative to each PDC described in PSAR section 5.3.

Table 5.3-1: Kemmerer Unit 1 Principal Design Criteria

Criterion	Title	SE Section(s)
I. Overall Requirements		
1	Quality Standards and Records	1.4.4, 8.1, and chapter 7
2	Design Bases for Protection Against Natural Phenomena	6.1, 6.3, 6.4 and chapters 2 and 7
3	Fire Protection	7.5.3
4	Environmental and Dynamic Effects Design Bases	6.1.1 and chapter 7
5	Sharing of Structures, Systems, and Components	Not Applicable – KU1 is a single unit site
II. Multiple Barriers		
10	Reactor Design	3.11, 3.12, 7.1.1, 7.1.2, 7.1.3, 7.1.4, 7.2.2, 7.6.2, and 7.6.3
11	Reactor Inherent Protection	3.11, 7.1.1, and 7.1.2
12	Suppression of Reactor Power Oscillations	3.11, 7.1.1, 7.1.2, and 7.2.5
13	Instrumentation and Control	7.6
14	Primary Coolant Boundary	6.4.1, 7.1.2, 7.1.3, 7.2.3, 7.2.4, 7.2.5, and 7.2.6
15	Primary Coolant System Design	3.12, 7.1.2., 7.1.3, 7.2.3, 7.2.4, 7.2.5, and 7.6
16	Containment Design	1.3.2.1 and chapter 7

Criterion	Title	SE Section(s)
17	Electrical Power Systems	7.7.1
18	Inspection and Testing of Electric Power Systems	7.7.1
19	Control Room	7.5.1, 7.6.7, and 7.8.4
III. Reactivity Control		
20	Protection System Functions	7.6.3, 7.6.4, and 7.6.5
21	Protection System Reliability and Testability	7.6.3, 7.6.4, and 7.6.5
22	Protection System Independence	7.6.3, 7.6.4, and 7.6.5
23	Protection System Failure Modes	7.6.3 and 7.6.4
24	Separation of Protection and Control Systems	7.6.2, 7.6.3, 7.6.4, 7.6.5, and 7.6.8
25	Protection System Requirements for Reactivity Control Malfunctions	7.6.2, 7.6.3, and 7.6.4
26	Reactivity Control Systems	3.11, 7.1.1, 7.1.2, 7.2.5, 7.6.2, and 7.6.3
27	Combined Reactivity Control Systems Capability	In accordance with RG 1.232, this PDC was deleted and incorporated into PDC 26
28	Reactivity Limits	7.1.1, 7.2.5, and 7.6.2
29	Protection Against Anticipated Operational Occurrences	7.1.1, 7.2.5, 7.6.2, 7.6.3, 7.6.4, and 7.6.5
IV. Fluid Systems		
30	Quality of Primary Coolant Boundary	7.1.2, 7.1.3, 7.2.3, 7.2.4, and 7.2.5
31	Fracture Prevention of Primary Coolant Boundary	6.4.1, 7.1.2, 7.1.3, 7.2.3, 7.2.4, and 7.2.5
32	Inspection of Primary Coolant Boundary	7.1.2, 7.1.3, 7.2.3, 7.2.4, 7.2.5, and chapter 8
33	Primary Coolant Inventory Maintenance	7.1.2, 7.2.4, and 7.6.3

Criterion	Title	SE Section(s)
34	Residual Heat Removal	3.12, 7.1.2, 7.1.3, 7.1.4, and 7.2.2
35	Emergency Core Cooling	7.1.2, 7.1.3, 7.1.4, and 7.2.1
36	Inspection of Emergency Core Cooling System	7.1.2, 7.1.3, 7.2.1, and chapter 8
37	Testing of Emergency Core Cooling System	7.1.2, 7.1.3, and 7.2.1
38	Containment Heat Removal	As described in TR NATD-LIC-RPRT-0002-A, Rev 1, the staff concluded that these PDC are not applicable to the Sodium design because the design does not include a pressure containing reactor containment structure. The functional containment basis is evaluated as part of PDC 16.
39	Inspection of Containment Heat Removal System	
40	Testing of Containment Heat Removal System	
41	Containment Atmosphere Cleanup	
42	Inspection of Containment Atmosphere Cleanup Systems	
43	Testing of Containment Atmosphere Cleanup Systems	
44	Structural and Equipment Cooling	7.5.1
45	Inspection of Structural and Equipment Cooling Systems	7.5.1
46	Testing of Structural and Equipment Cooling Systems	7.5.1
V. Reactor Containment		
50	Containment Design Basis	As described in TR NATD-LIC-RPRT-0002-A, Rev 1, the staff concluded that these PDC are not applicable to the Sodium design because the design does not include a pressure containing reactor containment structure. The functional containment basis is evaluated as part of PDC 16.
51	Fracture Prevention of Containment Pressure Boundary	
52	Capability for Containment Leakage Rate Testing	
53	Provisions for Containment Testing and Inspection	
54	Piping Systems Penetrating Containment	
55	Reactor Coolant Pressure Boundary Penetrating Containment	
56	Primary Containment Isolation	

Criterion	Title	SE Section(s)
57	Closed System	
VI. Fuel and Radioactivity Control		
60	Control of Releases of Radioactive Materials to the Environment	7.2.3, 7.4.1, 7.6.2, 7.6.6, 9.1, and 9.3
61	Fuel Storage and Handling and Radioactivity Control	7.1.2, 7.2.4, 7.3.1, 7.3.2, 7.3.3, 7.4.1, 7.6.2, and chapters 8, 9.1.1, and 9.3.1
62	Prevention of Criticality in Fuel Storage and Handling	3.13, 7.1.2, and 7.3
63	Monitoring Fuel and Waste Storage	7.2.3, 7.2.4, 7.3, 7.4.1, 7.6.2, 7.6.6, and chapter 9
64	Monitoring Radioactivity Releases	7.2.1, 7.4.1, 7.5.1, 7.6.2, and 7.6.6
VII. Additional Sodium-Cooled Fast Reactor Design Criteria¹⁶		
70	Intermediate Coolant System	7.1.4, 7.2.2, 7.2.3, and 7.2.4
71	Primary Coolant and Cover Gas Purity Control	7.1.3, 7.2.3, and 7.2.4
72	Sodium Heating Systems	7.1.2, 7.1.3, 7.1.4, 7.2.2, 7.2.3, 7.2.4, 7.3.2, and 7.6.2
73	Sodium Leakage Detection and Reaction Prevention and Mitigation	7.1.2, 7.1.4, 7.2.2, 7.2.3, 7.2.4, 7.3.2, and 7.5.3
74	Sodium/Water Reaction Prevention/Mitigation	7.1.2, 7.1.3, 7.1.4, 7.2.2, 7.2.3, 7.2.4, 7.3.1, 7.3.2, and 7.5.2
75	Quality of the Intermediate Coolant Boundary	7.1.4, 7.2.2, 7.2.3, and 7.2.4
76	Fracture Prevention of the Intermediate Coolant Boundary	7.1.4, 7.2.2, 7.2.3, and 7.2.4
77	Inspection of the Intermediate Coolant Boundary	7.1.4, 7.2.2, 7.2.3, 7.2.4, and chapter 8

¹⁶ PDC 70 through 79 are found in RG 1.232 appendix B. These PDC apply to both pool- and loop-type sodium-cooled fast reactors. As noted in the RG applicants/designers may also develop entirely new PDC as needed to address unique design features in their respective designs.

Criterion	Title	SE Section(s)
78	Primary Coolant System Interfaces	7.1.2, 7.1.3, and 7.1.4
79	Cover Gas Inventory Maintenance	7.1.2 and 7.2.3
VIII. Additional Criteria Unique to the KU1 Design¹⁷		
80	Reactor Vessel and Reactor System Structural Design Basis	7.1.1 and 7.1.2
81	Reactor Building Design Basis	7.8.1
82	Provisions for Periodic Reactor Building Inspection	7.8.1

5.4 Safety-Related SSCs

This section of the SE documents the staff's review of the SR functions and SR SSCs identified in PSAR section 5.2 and documented in PSAR table 5.2-1 through table 5.2-3. The staff reviewed PSAR section 5.2 to determine if the preliminary list of SR functions and SR SSCs was developed consistently with the methodology in NEI 18-04 and NEI 21-07 as endorsed in RG 1.233 and RG 1.253, respectively with the exceptions noted and evaluated in section 5.1 of this SE. To support its review, the staff conducted an audit that included relevant supporting design documents to assess the details of the supporting data associated with the calculation of probabilities, MSTs, and radiological consequences that are part of the PRA. The staff also notes that an overview of all three fundamental safety functions and the SSCs necessary to support them with adequate defense-in-depth (including both SR and NSRST SSCs) is provided in section 1.3.1 of this SE.

Two SR functions for control of heat generation were identified for the preliminary design in PSAR table 5.2-1. These are the (1) scram – gravity driven absorber insertion by latch release and (2) reactor scram on loss of power. The former is the primary scram mechanism for the reactor and credited in most LBEs. The latter is a SR function within the SR control system to initiate a scam on loss of power, “in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at safe shutdown condition.” Additional information on this SR function and the SSCs performing this SR function is provided in NAT-4950-A, “Instrumentation & Control Architecture and Design Basis Topical Report,” Rev. 2 (ML25232A241), PSAR section 7.6, and section 7.2.1 of this SE. The staff determined this preliminary set of SR functions for the control of heat generation is reasonable because it provides at least one SR shutdown method for the LBEs requiring shutdown to not exceed 10 CFR 50.34 dose limits or the F-C target.

Thirteen SR functions for heat removal were identified for the preliminary design in PSAR table 5.2-2. Of these, nine are passive SR functions: one for coastdown of the primary sodium pumps (PSPs), two for natural circulation in the reactor air cooling system (RAC) and the primary heat transport (PHT) system, and six for passive heat removal aspects of various plant

¹⁷ Consistent with RG 1.232 appendix B, the applicant included PDC 80, PDC 81, and PDC 82 based on the unique features in the KU1 design.

systems including fuel handling systems. The remaining four SR functions are pump trips for situations that would indicate failure or hazardous conditions in the systems that remove heat from the core, including high primary sodium temperature, low primary sodium level, and high primary sodium level.

Both the PSPs and the intermediate sodium pumps (ISPs) trip on high primary sodium temperature to limit heat addition to the primary coolant. As described in PSAR section 7.6.3.2, the PSP and ISP trip on high primary coolant temperature only occurs if a scram signal exists in the RPS and neutron flux is below the setpoint, in addition to primary sodium cold pool temperature being above the setpoint. There is also a SR trip of the ISPs on high primary sodium level, which would indicate a leak in the intermediate heat exchanger (IHX). The final SR pump trip is a trip of the primary sodium processing system (SPS-P) pump on low primary sodium level, which would indicate a leak within the SPS-P system. This trip prevents the reactor vessel sodium from draining below its safe level.

To summarize, RAC is the SR means of heat removal from the reactor core, while other systems containing fuel (like the ex-vessel storage tank and ex-vessel handling machine) have similar SR passive heat removal functions. These functions are supported by SR functions related to natural circulation flow and pump trips necessary to limit heat addition or maintain coolant inventory. The staff determined that this preliminary set of SR functions providing control of heat removal is reasonable because they, in conjunction with the NSRST heat removal functions, sufficiently address heat removal considerations for the preliminary set of LBEs described in chapter 3 of the PSAR. For a limited number of LBEs, the staff found that the analysis supporting sufficient heat removal was inconsistent with the description in chapter 3 of the PSAR. However, this did not impact the staff's determination for these SR functions because available SSCs, including SR SSCs, are capable of providing sufficient heat removal. As noted in chapter 3 of the SE, the staff expects the event descriptions and analyses to be updated as needed at the OL stage to ensure consistency with the final LBEs.

Ten SR functions for radionuclide retention were identified for the preliminary design in PSAR table 5.2-3. These controls are radionuclide barriers with the exception of the reactor enclosure system (RES) pressure relief, which is located in the SCG and prevents pressure in the RES and supporting system functional containment barriers from exceeding their design pressure. The pressure relief is directed into the SCG vapor trap cell which has an NSRST function to isolate based on the position indication of the reactor vessel relief valves, as described in PSAR section 7.2.3.

The SR radionuclide retention barriers include the fuel cladding, primary coolant boundary (including RES), vessels used for fuel handling (e.g., ex-vessel storage tank (EVST) and bottom loading transfer cask (BLTC) barriers), and the pin removal cell (PRC) cell barrier. Due to the preliminary nature of the design, many details for these barriers are unavailable. Because barrier performance in mitigating radionuclide release is highly dependent on the detailed design of the barrier, this prevented the staff from making any determinations related to whether the mitigation factors assumed for these barriers used in the generation of source terms in section 3.2 are supported by the design. However, the assumptions regarding barrier performance provided in the PSAR give a target for the final design, and as such the staff determined that consistent with 10 CFR 50.35(a)(4), there is reasonable assurance that these design aspects will be resolved at or before the completion of construction of the proposed facility and the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public. The staff expects to review additional design information at the OL stage to support findings on the acceptability of these barriers and

their mitigating effects, including information on heating, ventilation and air conditioning (HVAC), and other connections and the isolations needed to achieve the target leak rates.

The staff notes that failure of SR and NSRST radionuclide barriers were generally treated as a degraded performance condition, rather than complete failure. In most cases, the degraded performance was assumed to be 100 percent leakage per day or approximately 4 percent per hour. As discussed in SE section 5.1, justification for degraded performance assumed post-failure will be reviewed at the OL stage.

The SCG isolation radionuclide retention SR function, DL3-RR10, was identified as necessary during the staff's review. This function provides a SR isolation of the normal process flow path out of the reactor vessel. Without this isolation, radionuclide releases into the reactor vessel would be exhausted through this pathway, leading to EAB TEDE values that could potentially exceed the 10 CFR 50.34(a)(1)(ii)(D) dose criteria.

The staff notes that the need for an SR SCG isolation was not originally identified from the LBE results in PSAR chapter 3 because the preliminary PRA did not include this SR function as a PSF or as a potential bypass path for the RES barrier. The bypass path of an SCG pipe break was included, but with the low probability of a pipe break occurring concurrently with a release into the reactor vessel, the function to automatically close the SCG valve on leak detection (DL4-RR4a) to mitigate this bypass is classified as an NSRST function. The detection for this isolation is also NSRST and located outside the SCG piping, meaning it would not be effective in detecting a release in the reactor vessel without a concurrent pipe break.

LBEs involving radionuclide releases into the RES were evaluated with MSTs that assumed the RES barrier was isolated and could maintain 1 percent leakage per day. By comparison, the normal process flow path that was not modeled would move the release from the RES into the SCG and subsequently into the RWG within hours based on preliminary design flowrates, effectively bypassing the RES function. The addition of the SCG isolation SR function, including the necessary detection and actuation signals, allows the RES barrier (DL3-RR1) to perform its function. USO updated table 5.2-3 in Rev. 1 of its PSAR to reflect this addition.

USO also added statements to the chapter 3 radiological consequence sections associated with the nine LBEs impacted, documenting that this pathway was not modeled and that LBE analyses that include this pathway and the mitigation provided by the SR SCG isolation will be provided at the OL stage. The addition of this flow path to the analysis is expected to result in some increases to the associated source terms due to the radionuclides that exit through the SCG prior to detection and isolation. Because the time prior to isolation is expected to be short relative to the overall duration of the event and there is margin between the current consequence results and the F-C target, the current consequence results are sufficient to support the conclusions at the CP stage and the full impacts of this change will be reviewed at the OL stage.

The staff notes that one of the most risk significant barriers that will require additional information in the OL application is the PRC, a hot cell where pins are removed from lead test assemblies (LTAs) by operators using manipulators, to be sent for further analysis. The failure of a pin during this operation has a frequency in the AOO range. In addition, the LTAs generally have less decay time than other fuel assemblies handled outside of the RES, meaning there is a higher radionuclide inventory available for release.

The staff gained an understanding of the current design for this cell through audit discussions. Since the design is preliminary and not yet fully justified, the staff could not determine if the consequence evaluation for the AOO involving the PRC (RFH-OERC-BL) was conservative. USO indicated that a more accurate design assessment including detailed design information, consistent with the level of associated risk, would be provided and staff will review the PRC at the OL stage. The staff determined that it is appropriate to consider detailed design information at the OL because the PRC barrier is already identified as SR and reasonable special treatments associated with the design and construction of the PRC were provided in the PSAR, and are evaluated in SE section 7.3.2.

5.4.1 Conclusion

The staff reviewed the SR functions and SR SSCs described in PSAR section 5.2 and determined that the SR SSCs meet the requirements in 10 CFR 50.34 and 10 CFR 50.35. The SR SSCs identified provide reasonable assurance of protection for the LBEs described in PSAR chapter 3 and were appropriately identified using the SSC safety classification process described in section 5.1 of the PSAR and evaluated in the corresponding section of this SE based on preliminary design. SR SSCs will continue to be evaluated by USO as the design progresses and will be reviewed again by staff at the OL stage when additional design information is provided. Justification should be provided for any radionuclide retention barrier leak rates and isolation functions needed to achieve those leak rates.

5.5 Non-Safety-Related with Special Treatments SSCs

This section of the SE documents the staff's review of the NSRST functions and SSCs identified in PSAR section 5.2 and documented in PSAR table 5.2-4. The staff reviewed PSAR section 5.2 to determine that the preliminary list of NSRST functions and NSRST SSCs is consistent with NEI 18-04 and NEI 21-07 as endorsed in RG 1.233 and RG 1.253, respectively. To support its review, the staff conducted an audit that included relevant design documents to confirm the details of the supporting data associated with the calculation of probabilities, MSTs, and radiological consequences that are reflected in the PRA results.

PSAR table 5.2-4 provides a preliminary list of all NSRST PSFs. This table includes whether the NSRST function is risk-significant or was identified through the DID evaluation. Risk-significant NSRST functions are those needed to maintain the 95th percentile LBE results below the F-C target and those that have a greater than 1 percent impact on the cumulative risk metrics. As described in SE section 5.1, USO used the 95th percentile when identifying RSFs and the additional step within the DID evaluation compensated for the NSRST classification step that was inadvertently missed. As described in SE section 5.1, USO has committed to perform the step as described in NEI 18-04 for the OL.

As discussed in PSAR section 5.2, SSCs that have no associated LBEs have been identified as DID but may be risk-significant once modeled within the PRA. As stated in PSAR section 5.2, USO has committed to performing analyses to confirm the risk significance of all non-SR PSFs at the OL stage. The following are NSRST actions that are redundant to SR functions. The following NSRST functions are not credited in the PRA and are identified in PSAR table 5.2-4 as necessary for DID adequacy:

- Manual reactor scram
- Manual PSP trip

- Manual ISP trip
- Manual sodium processing system (SPS) pump trip on leak indication
- PSP trip automatic backup
- ISP trip automatic backup

The following are NSRST alternative shunt trips that are identified as risk-significant defense line functions in PSAR table 5.2-4, but are not described in detail within the PSAR due to the timing of when USO added them to the PRA relative to the submittal of the PSAR:

- Alternative shunt trip on high core exit temperature
- Alternative shunt trip on low primary sodium pump outlet pressure
- Alternative shunt trip on high IHT level
- Alternative shunt trip on low IHT level

The staff determined that because these are backups to SR trip functions, the classification of risk-significant NSRST is expected to be appropriate or conservative relative to the final safety classification. Therefore, the staff determined that the finalization of this aspect of the design can be left to the OL stage with reasonable assurance that it will be resolved at or before the completion of construction of the proposed facility and the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public, consistent with 10 CFR 50.35(a)(4).

The staff determined that this preliminary set of NSRST trip functions for control of heat generation and heat removal is reasonable because they, in conjunction with the SR functions, sufficiently detect and terminate adverse conditions present in the preliminary set of LBEs described in chapter 3 of the PSAR and this SE because they provide redundant trip capability to the SR functions discussed in SE section 5.4. Risk significance determinations will depend on the reliability of the primary trip function and potential consequences of failure and will be reviewed at the OL stage.

Other NSRST functions identified as risk-significant in PSAR table 5.2-4 include:

- CRD driveline scram follow
- FHP emergency makeup

Additional NSRST DID functions include the automatic seismic trip, ISP trip on low IHT level, IAC passive mode operation, post-accident monitoring, and intermediate leak guard piping. With the exception of the IAC passive mode operation, these functions do not have associated LBEs evaluated in the PSAR.

The staff determined that these additional NSRST functions for controlling heat generation and heat removal are reasonable and consistent with the preliminary set of LBEs described in

chapter 3 because they provide trips based on conditions that could lead to more severe events, redundant heat removal capability used in many non-DBA LBEs, and monitoring for potential releases following LBEs. As the PRA continues to mature and LBEs are developed to evaluate the risk significance of these functions, SSC and risk significance classifications may change. In particular, the staff notes that the SSC classification for the intermediate leak guard piping might change as a result of the fire PRA which will be provided for review at the OL stage.

There are two guard vessels listed in table 5.2-4 of the PSAR as NSRST for DID adequacy: the guard vessel for the reactor vessel and the EVST guard tank. These do not have an associated LBE in the PSAR, which the applicant confirmed in audit is due to the low likelihood of failure associated with the reactor vessel and the EVST, causing events involving these functions to be below the 5×10^{-7} frequency cut-off for LBEs. Further evaluation of the basis for reactor vessel leak frequencies is provided in section 3.4.1 of this SE. Functions and SSCs with very low failure rates will be reviewed by the staff at the OL stage to ensure that the bases support the frequencies being used within the PRA.

The staff developed SE table 5.5-1 based on the source term and LBE descriptions in PSAR chapter 3 and the tables in PSAR chapter 5, which provides a summary of the radionuclide retention functions mapped to the material-at-risk (MAR) they are mitigating and the LBE scenarios with which they are associated. The staff noted that all NSRST radionuclide retention functions in PSAR table 5.2-4 are listed as required for DID adequacy, however, the staff notes some of these functions could be classified as risk-significant at the OL stage once the risk significance determination has been performed.

SE table 5.5-1 identifies where USO assumed degraded performance of the radionuclide retention barrier within their LBE evaluations. USO generally credited degraded performance of NSRST radionuclide barriers in LBEs where that barrier was assumed to be failed, or for NSRST barriers within a DBA analysis, assuming that even in a failed state there would be some amount of mitigation provided by these barriers. For the SPS cell barrier, the nominal leakage was assumed to be 10% per day and the degraded performance was assumed to be 50 percent per day. Due to the short half-lives of sodium isotopes, the degraded performance of the SPS cell has a significant impact on consequences. As noted in SE section 5.1, USO will need to provide bases for the nominal and degraded performances of radionuclide retention barriers used in the source term analyses at the OL stage.

The staff made the following observations from SE table 5.5-1, which was generated from information in chapter 3 of the PSAR, regarding the role of NSRST radionuclides barriers in LBE mitigation for KU1:

- For events with the largest MAR (i.e., fuel releases during operation), an SR barrier (in addition to the SR fuel cladding) and an NSRST barrier are needed to provide sufficient safety as determined through the SSC safety classification.
- For fuel release after shutdown, an SR barrier (in addition to the SR fuel cladding) is generally sufficient unless the frequency of the initiating event plus SR barrier failure is high enough that an LBE involving the SR barrier failure is included. In this case, an NSRST barrier is needed to provide mitigation when the SR barrier fails.
- For releases from auxiliary systems, the MAR is generally low enough that only one NSRST barrier is needed, in addition to the process barrier, to reach acceptable risk.

- The gaseous radwaste processing system (RWG) inventory and tritium MAR are low enough that no mitigation is needed to reach acceptable risk when the process barrier fails, as demonstrated by the SSC safety classification results and confirmed through the staff's evaluation.

The staff determined that this preliminary set of NSRST radionuclide retention barriers is reasonable based on the preliminary set of LBEs described in chapter 3 of the PSAR because they provide sufficient mitigation and DID to maintain LBEs below the F-C target and within 10 CFR 50.34 limits.

5.5.1 Conclusion

The staff reviewed the NSRST functions and NSRST SSCs described in PSAR section 5.2, as discussed above, and determined that the NSRST SSCs meet the requirements in 10 CFR 50.34 and 10 CFR 50.35, and are consistent with guidance in NEI 18-04, as endorsed in RG 1.233. The NSRST SSCs identified provide reasonable assurance of protection for the LBEs described in PSAR chapter 3 and were identified using the SSC safety classification process described in PSAR section 5.1, with the clarifications in SE section 5.1, based on the preliminary design. The staff will review the risk significance of all NSRST functions at the OL stage based on the impact of the function on the LBE results relative to the F-C target and the total impact on the facility risk based on the cumulative risk metrics.

Table 5.5-1: Radionuclide retention functions mapped to MAR and LBEs

Material-at-risk (MAR)	Primary Barrier	Secondary Barrier	Tertiary Barrier	Licensing Basis Event (LBE)
Fuel Inventory	Fuel Cladding (SR) ¹⁸	RES boundary (SR) SCG Isolation (SR)	Head Access Area (HAA) barrier (NSRST)	LFF-SAO-BL (DBE) DHP-LOOP-3 (BDBE) DHP-LOOP-4 (BDBE)
			HAA barrier (NSRST), degraded performance	LFF-SAO-1 (BDBE) LFF-SAO-CN (DBA) RFH-FDIV-CN (DBA)
		RES boundary (SR) SCG Isolation (SR) Ex-Vessel Handling Machine (EVHM) cask barrier (SR)		RFH-FDIV-1 (DBE) RFH-FDIV-3 (DBE)
		RES boundary (SR), degraded performance SCG Isolation (SR)	Reactor building (RXB), degraded performance (NST)	RFH-FDIV-2 (BDBE) RFH-FDIV-4 (BDBE)
			HAA barrier (NSRST)	LFF-SAO-2 (BDBE) ¹⁹
		EVHM cask barrier (SR)		RFH-LMCA-1 (BDBE) RFH-FDEM-1 (BDBE)

¹⁸ Fuel Cladding is credited as a SR radionuclide retention barrier and prevents larger radionuclide release but is assumed to fail for the pins involved in the LBEs.

¹⁹ The event description for LFF-SAO-2 is that the vessel head fails while the HAA succeeds. However, the analysis performed for the PSAR had the vessel head succeeding and the HAA failing. See SE section 3.7.1.6 for additional discussion on the evaluation of this event.

OFFICIAL USE ONLY — PROPRIETARY INFORMATION

Material-at-risk (MAR)	Primary Barrier	Secondary Barrier	Tertiary Barrier	Licensing Basis Event (LBE)
		EVHM cask barrier (SR), degraded performance		RFH-LMCA-2 (BDBE) RFH-FDBL-2 (BDBE) RFH-FDEM-2 (BDBE)
		PRC cell barrier (SR)		RFH-OERC-BL (AOO) RFH-FDRC-1 (BDBE)
		EVST barrier (SR)		RFH-FDET-1 (BDBE)
		BLTC barrier (SR), degraded performance	Fuel handling building barrier and filtration (NSRST)	RFH-ESWR-2 (BDBE)
				RFH-ESWR-3 (BDBE)
		BLTC barrier (SR)		RFH-FDBL-1 (BDBE)
		BLTC barrier (SR) PIC radionuclide boundary (NSRST)		RFH-ESWR-1 (DBE) RFH-FDPI-BL (BDBE) RFH-ESWR-CN (DBA)
		PIC radionuclide boundary (NSRST), degraded performance		RFH-FDPI-1 (BDBE)
		Failed Fuel Canister (SR)		No associated LBEs

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Material-at-risk (MAR)	Primary Barrier	Secondary Barrier	Tertiary Barrier	Licensing Basis Event (LBE)
		Spent Fuel Pool (NST, for radionuclide retention)		RFH-FDSP-1 (DBE) RFH-FDSP-2 (BDBE) RFH-FDSP-CN (DBA)
SPS Inventory	Primary SPS barrier (NSRST)	HAA barrier (NSRST), degraded performance		RRS-SPLX-CN (DBA)
		HAA barrier (NSRST) ²⁰	RXB superstructure (NST), 2-hour exhaust time	RRS-SPLX-BL (DBE)
		SPS cell barrier (NSRST)	RAB substructure (NSRST, not for radionuclide retention), 2-hour exhaust time	RRS-SPLA-BL (DBE)
		SPS cell barrier (NSRST), degraded performance		RRS-SPLA-CN (DBA)
SPS Tritium Inventory	Intermediate cold trap SPS barrier (NSRST)			RRS-ISPL-BL (DBE) RRS-ISPL-CN (DBA)
SCG Inventory	Primary SCG barrier (NSRST)	Vapor trap cell barrier (NSRST), degraded performance		RRS-CGR-CN (DBA)
		Vapor trap cell barrier (NSRST)	NHV filtration (NST)	SUD-CGR-2 (BDBE)

²⁰ SPS piping has an additional barrier for fire protection surrounding the piping in the HAA that is not credited in the LBE analyses.

OFFICIAL USE ONLY — PROPRIETARY INFORMATION

Material-at-risk (MAR)	Primary Barrier	Secondary Barrier	Tertiary Barrier	Licensing Basis Event (LBE)
				RRS-CGR-1 (DBE) ²¹
		HAA barrier (NSRST)	NHV isolation (NSRST) and filtration (NST)	RRS-CGR-BL (DBE)
			NHV filtration (NST)	SUD-CGR-1 (BDBE)
RWG Inventory	Gaseous radwaste barrier (NSRST)	Holdup tank vault (NST)		RRS-RWG-1 (DBE)
		Vent Stack Filtration (NST)		RRS-RWG-2 (DBE)
				RRS-RWG-CN (DBA)

²¹ RRS-CGR-1 uses the results from RRS-CGR-BL. See SE section 3.6.1.3 for additional discussion on the evaluation of this event.

6 SAFETY-SIGNIFICANT SSC CRITERIA AND CAPABILITIES

The Kemmerer Power Station Unit 1 (KU1) PSAR chapter 6 includes both safety-related (SR) and non-safety-related with special treatment (NSRST) structures, systems, and components (SSC) criteria and capabilities. The PSAR includes elements that address Regulatory Guide (RG) 1.253, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Revision (Rev.) 0 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23269A222), and DANU-ISG-2022-01 “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap” (ML23277A139).

6.1 Design Requirements of Safety-Related SSCs

PSAR section 6.1 includes design requirements for SR SSCs and includes a discussion of the design basis hazard levels (DBHLs) in PSAR section 6.1.3.1, a discussion of the approach to the safety-related design criteria (SRDC) in PSAR section 6.1.3.2, and a summary of DBHL-related requirements for non-safety-related SSCs in PSAR section 6.1.3.3.

The regulatory requirements applicable to the staff’s review of PSAR section 6.1 are:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, “Contents of applications; technical information,” paragraph (a), “Preliminary safety analysis report;” and
- 10 CFR 50.35, “Issuance of construction permits.”

Applicable guidance for the staff’s review of PSAR section 6.1 includes:

- DANU-ISG-2022-01;
- RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Rev. 1 (ML070360253);
- RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” Rev. 2 (ML21253A071);
- RG 1.91 “Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants,” Rev. 3 (ML21260A242);
- RG 1.189, “Fire Protection for Nuclear Power Plants,” Rev. 5 (ML23214A287);
- RG 1.204, “Guidelines for Lightning Protection for Production and Utilization Facilities,” Rev. 1 (ML23241A957);
- RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Rev. 0 (ML20091L698), which endorses with clarifications, the guidance in NEI 18-04, “Risk-

Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Rev. 1 (ML19241A472); and

- RG 1.253, Rev. 0, which endorses with clarifications and additions NEI 21-07, “Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 Methodology,” Rev. 1 (ML22060A190).

The principal design criteria (PDC) as defined in PSAR section 5.3 and evaluated by the staff in the corresponding section of this safety evaluation (SE), that apply to the review of PSAR section 6.1 are:

- PDC 2, “Design Bases for Protection Against Natural Phenomena;”
- PDC 3, “Fire Protection;”
- PDC 4, “Environmental and Dynamic Effects Design Basis;”
- PDC 19, “Control Room;” and
- PDC 81, “Reactor Building Design Basis.”

6.1.1 Technical Evaluation

6.1.1.1 Design Basis Hazard Levels

The staff reviewed PSAR section 6.1.1, which discusses the DBHLs for SSCs designed to withstand the effects of internal plant hazards and hazards that are external to the plant, ensuring no adverse impact on their ability to perform intended safety functions. The applicant stated that DBHL-related requirements for NSRST and non-safety-related with no special treatment (NST) SSCs are based on preventing interactions with SR SSCs. The design characteristics and parameters for DBHLs of external hazards, including wind and tornado, external flood, missile protection, seismic hazards, and extreme winter precipitation, are presented in PSAR table 2.1-1. The applicant’s justification for these characteristics and parameters is evaluated in section 2.1 of this SE.

The staff performed its evaluation of DBHLs in PSAR section 6.1.1, using the guidance in RG 1.233 and RG 1.253. Specifically, NEI 21-07, endorsed with clarifications and additions in RG 1.253, states that construction permit (CP) application content for section 6.1.1 should be as complete as possible, based on the site characterization required in 10 CFR 50.34(a). NEI 21-07 further states that DBHLs extend beyond environmental hazards external to the plant and include both external and internal hazards, such as seismic events, wind (including tornadoes and wind-generated missiles), external flooding, hazards from external facilities, internal fires, internal floods, high-energy line breaks, and internally generated missiles.

The staff also conducted an audit to confirm and enhance understanding of the DBHL-related information provided in the PSAR. The audit report is available in ADAMS (ML25302A443). It provides the DBHL-related documentation reviewed by the staff, the questions raised during the audit, and question resolutions.

USO selected a set of DBHLs as part of its design bases that the staff reviewed and, for the reasons described below, determined were consistent with RG 1.233. As a result of the staff's review, USO updated the PSAR to clarify that DBHLs are selected in accordance with approved methods and the information provided is consistent with the content guidance in section 6.1.1 and table 6-1 as described in NEI 21-07, as well as certain subsections of chapters 3.2 through 3.7 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (ML16084A812, ML070570001, ML070570002, ML070550043, ML070570003, ML070370569, ML070380167, ML063600395, ML14190A180, ML070510635, ML100331298, ML070460362, ML070570004, ML070550032, ML16088A041, ML14198A460, ML13198A223). The applicant also stated in PSAR section 6.1.1 that a hazards screening analysis was performed for all applicable hazards and the list of DBHLs will be re-assessed and updated at the OL stage.

The staff reviewed USO's approach for identifying, selecting, and assessing DBHLs and, for the reasons described below, determined it was reasonable and that the DBHLs preliminarily identified for the CPA provide a reasonable basis for hazard evaluation. Detailed staff evaluations of individual hazards are provided in the following subsections. Specifically, for each hazard the staff evaluated the information provided in the PSAR to ensure it was reasonable to establish the design bases for protection against hazards for applicable KU1 SSCs consistent with PDC 2, 3, 4, 19, or 81. Notably, for some hazards that rely on final design information (e.g., internal fire, external fire, and internal flooding) the staff assessed the applicant's plans to ensure safety-significant KU1 SSCs will have appropriate protection against these hazards at the OL stage.

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PDC 3 requires that safety-significant SSCs are designed and located to minimize probability and effect of fires and explosions. PDC 4 requires safety-significant SSCs to be appropriately protected against dynamic effects, including those resulting from missiles, pipe whipping, and discharging fluids caused by equipment failures or external events and conditions outside the nuclear power unit. PDC 19 requires, in part, that adequate habitability measures be provided to permit access to and occupancy of the control room during normal operations and under accident conditions. PDC 81 requires the reactor building to be designed to structurally protect the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink during postulated accidents.

High Winds

The staff reviewed the applicant's assessment of high wind hazards and the site-specific high wind parameters provided in PSAR section 2.4.1.3.2. The applicant stated that for the structural design of safety-significant SSCs, wind loading conditions are based on 3-second gust wind speeds of 110 and 115 miles per hour, in accordance with American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) Standard 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures." The applicant's evaluation was based on local historical data for the KU1 site area, as documented in PSAR section 2.4.1.3.2 and table 2.1-1.

Based on the staff's review of PSAR section 2.4.1.3.2 and its determination documented in SE chapter 2, the staff determined that the DBHL established for high winds is consistent with the design basis high wind parameters, as documented in section 2.4.1.1.5 of this SE and is therefore consistent with PDC 2.

Tornadoes

The staff reviewed the applicant's assessment of tornado hazards and the site-specific tornado parameters described in PSAR section 2.4.1.3.3. The applicant stated that the design-basis tornado characteristics applicable to safety-significant SSCs are consistent with RG 1.76, Rev. 1 including: maximum wind speed, translational speed, maximum rotational speed, radius of maximum rotational speed, pressure drop, and rate of pressure drop.

Based on the staff's review of PSAR section 2.4.1.3.3 and table 2.1-1 and its determination documented in SE chapter 2, the staff determined that the DBHL established for tornadoes is consistent with the design basis tornado characteristics, as documented in section 2.4.1.1.6 of this SE and is therefore consistent with PDC 2.

External Flooding

PSAR section 2.5.1 provides flooding information for the vicinity of KU1 site and summarizes the types and combinations of flood-producing phenomena considered in establishing the design-basis flood. The postulated flood scenarios include:

- Rivers and streams;
- Dam breaches;
- Surges and seiches;
- Tsunami;
- Snow melts and ice jams; and
- Channel diversion.

Additionally, the impacts of local intense precipitation (LIP) are described in PSAR section 2.5.1.

As described in PSAR section 6.1.1.2, SR structures located below grade are designed with damp-proofing and waterproofing to minimize potential groundwater intrusion. To prevent external flooding from entering an NSRST structure and propagating to an SR structure, the top-of-concrete elevation for NSRST structures is designed to be above the design-basis flood level described in table 2.1-1. In addition, exterior doors of NSRST structures are located above the ponding level associated with LIP, and the site is graded away from these doors. Therefore, the design-basis external flood hazards are not expected to inundate SR SSCs.

Based on the staff's review of PSAR section 2.5.1 and table 2.1-1 and its determination documented in SE chapter 2, the staff determined that the DBHL established for external flooding is consistent with the design basis external flood characteristics, as documented in section 2.5.1 of this SE and is therefore consistent with PDC 2.

Internal Flooding

PSAR section 6.1.1.3 states that, “The design basis internal flooding analysis will consider the single postulated fluid system failure that results in the most limiting flood conditions in each area that contains safety-significant SSCs.” The PSAR clarifies that the internal flooding sources listed below from sodium, molten salt, oil, and water systems are considered in the internal flooding analysis:

- Pipe failures (as described in PSAR section 6.1.1.5);
- Circumferential breaks of piping systems not designed to retain function following seismic events (i.e., not designed to SCS1, SCS2, or SCN1 seismic standards as defined in PSAR section 6.4.1.1) and piping systems not designed to be protected from the tornado DBHL;
- Rupture of tanks not designed to retain function following seismic events and tanks not designed to be protected from the tornado DBHL;
- Communication from other areas, including communication through the floor drain system;
- Operation of the fire protection system;
- Flooding due to impact of postulated missiles on fluid systems (as addressed in PSAR section 6.1.1.4.1); and
- Flooding due to postulated heavy load drops.

PSAR section 6.1.1.3 identifies potential design features to protect safety-significant SSCs from postulated flooding, including use of isolation valves, physical barriers, and drains. The PSAR also notes that systems in the energy island (EI) (e.g., water, salt, and oil systems) are physically separate and at a lower elevation than the safety-significant SSCs, which are located in the NI structures.

The PSAR states the seismically-qualified fluid systems consist primarily of sodium systems. Due to the reactive nature of sodium, internal flooding from sodium pipe leaks is largely accounted for in the sodium leak fire protection strategy. PSAR section 5.3.1.4 states that sodium-containing SSCs external to the reactor vessel are provided with measures such as guard pipes, jackets, location within lined cells, or catch pans to collect leakage from moderate energy pipe cracks in seismically-qualified SSCs.

Based on its review of the internal flooding information in the PSAR as confirmed through audit, the staff determined that the internal flooding hazard at KU1 is reasonably characterized for the CP stage, and that planned design measures discussed by the applicant in the PSAR are appropriate to ensure SR SSCs will maintain their safety-significant functions under postulated internal flood conditions, consistent with PDC 4. The staff will review USO’s evaluation of the necessary flood protection design features at the OL stage, based on the final design information.

Turbine Missile

The KU1 turbine missile hazard is assessed and discussed in PSAR section 2.4, table 2.1-2, and section 6.1.1.4. PSAR table 2.1-2 indicates that the turbine generator is located on the EI within the turbine facility building, approximately 500 ft south of the nuclear island, with its rotor oriented in the north-south direction. Based on this location and orientation, USO concluded that the potential for damage from turbine-generated missiles to safety-significant SSCs is insignificant. The staff confirmed through its audit that the rotor orientation and the separation distance between the turbine and the nuclear island are sufficient to ensure that safety-significant SSCs are protected from potential turbine missile impacts.

The staff reviewed PSAR section 2.4, table 2.1-2, and section 6.1.1.4, and determined turbine missiles are appropriately addressed because the orientation of the turbine rotor and the separation of the turbine and nuclear island ensure negligible impacts to safety-significant SSCs, consistent with PDC 4.

Internally Generated Missile

The hazard associated with missiles generated from rotating equipment is assessed in PSAR section 6.1.1.4. For missiles potentially generated from pressurized systems and components, including tanks and cylinders containing compressed gas, USO states in the PSAR that a probability analysis is being performed to support a future OL application. The applicant states in PSAR 6.1.1.4.1 that if the probability of a missile occurrence, impacting a SR target and causing an adverse effect on its ability to perform an SR function, is less than 1×10^{-7} per year, the missile is not considered a design-basis missile.

As described in PSAR section 6.1.1.4.2, SR SSCs are either protected from or designed to withstand impacts from a single design-basis missile without loss of capability to perform their required safety functions. Design-basis missiles are addressed using one or more of the following methods:

- Physically separating or orienting potential design-basis missile sources to prevent impacts on SR SSCs;
- Demonstrating that a missile impact does not adversely affect the ability of SR SSCs to perform their SR functions; and
- Providing missile barriers. When missile barriers are employed, the design procedure for the barrier will be provided at the OL stage.

For the design-basis missile assessment, USO considered missiles generated from pressurized systems and components, turbine missiles, rotating equipment, secondary missiles generated from primary missile impacts, wind- and tornado-generated missiles, and site-proximity missiles.

The staff evaluated, and confirmed through audit, the screening approach and mitigation for a single internal missile provided in PSAR section 6.1.1.4 and determined that the DBHL established for the internal missile hazard is consistent with the design basis internal missile and is therefore consistent with PDC 4.

Rupture of Piping

PSAR section 6.1.1.5 states that high-energy liquid piping systems are not planned for installation in safety-significant structures. Due to the physical separation between the EI and the nuclear island, the applicant concluded that the effects from pipe ruptures in fluid systems located in the EI do not affect SSCs in the nuclear island. Circumferential ruptures and longitudinal splits are not postulated in moderate-energy piping.

PSAR section 6.1.1.5 also states that SR SSCs are designed such that design-basis pipe ruptures will have no adverse impact on their capability to perform required SR functions. Design-basis pipe ruptures are addressed by one or more of the following methods:

- Physically separating piping systems subject to postulated design-basis pipe ruptures from SR SSCs where feasible.
- Providing guard piping, barriers, or fluid collection systems to mitigate the effects of design-basis pipe ruptures on SR SSCs.
- Demonstrating that design-basis pipe ruptures do not adversely impact the ability of SR SSCs to perform their SR functions.

The staff evaluated, and confirmed through audit, the methods for ensuring piping ruptures do not adversely impact SR functions provided in PSAR section 6.1.1.5 and determined that the applicant's approach for the pipe rupture DBHL is reasonable. The staff will review USO's evaluation of pipe ruptures and their impacts at the OL stage, based on the final design information and piping layout.

Seismic

The seismic hazard at the KU1 site is addressed in several sections of the PSAR, with the detailed discussions provided in PSAR section 2.6 and the associated seismic design described in PSAR section 6.4.1. In addition to the PSAR, the staff conducted an audit that included supporting documents that define plant-level requirements to meet seismic design-basis hazards to confirm information in PSAR sections 2.6 and 6.4.1.

The staff's review of the KU1 seismic hazard and evaluation of the potential impact of a seismic event on SSCs are provided in sections 2.6, 6.1.1.3, 6.4, and 7.8 of this SE. In those sections, the staff determined that USO has provided necessary information to address the design-basis seismic hazard and to support its evaluation of how the design addresses the hazard.

Although not explicitly considered in the DBHLs, sufficient seismic design margin will need to be demonstrated at the OL stage to meet the LMP risk target for beyond-design-basis events (BDBEs). The applicant's approach is consistent with the discussion in RG 1.253, Appendix A.3.7, which states that certain hazards may be dispositioned by crediting DBHLs in lieu of explicitly modeling them in the PRA and evaluating their consequences. Based on this, the staff will confirm the results are acceptable against the F-C target when the full scope PRA has been completed at the OL. The staff evaluation of seismic hazard is documented in SE section 2.6.2, and the seismic design evaluation is provided in SE section 6.4.

Based on the staff's review of PSAR sections 2.6 and 6.4.1 and its determinations documented in SE chapters 2, 6, and 7, the staff determined that the DBHL established for seismic is consistent with the design basis seismic characteristics, as documented in sections 2.6, 6.1.1.3, 6.4, and 7.8 of this SE and is therefore consistent with PDC 2 and 81.

Extreme Winter Precipitation

The KU1 design-basis extreme winter precipitation hazard, along with its parameters and supporting bases, is described in PSAR section 2.4.1.3.5, table 2.1-1, table 2.4-15, and section 6.4. Loads resulting from extreme winter precipitation have been considered in the design of SSCs at KU1.

Based on the staff's review of PSAR section 2.4.1.3.5, table 2.1-1, table 2.4-15, and 6.4, and its determination documented in SE chapters 2 and 6, the staff determined that the DBHL established for extreme winter precipitation is consistent with the design basis extreme winter precipitation parameters, as documented in section 2.4.1.1.8 of this SE and is therefore consistent with PDC 2.

Internal Fire

As described in section 6.1.1.8 of the PSAR, internal fire is a hazard that will be fully evaluated at the OL stage. The applicant stated that during the OL stage, fire protection features will be developed or refined to satisfy DBA analysis assumptions and internal fire PRA results, or DBA analysis assumptions will be updated as necessary based on the evolving design.

As described in PSAR section 5.3.1.3, the NI will be designed in accordance with RG 1.189. Future fire hazard analyses and modeling will define fire areas and establish appropriate design features to suppress fires within the area and prevent fire propagation beyond the area. These design features will include sodium leak detection, containment, inert environments, equipment material selection to reduce combustible quantities, fire detection, and both automatic and manual dry-gas and water suppression systems.

Based on its review of the internal fire information in the PSAR as confirmed through audit, the staff determined that the internal fire hazard at KU1 is reasonably characterized for the CP stage, and that planned protection and suppression design features discussed by the applicant in the PSAR are appropriate to ensure SR SSCs will maintain their safety-significant functions under postulated internal fire conditions, consistent with PDC 3. The staff will review USO's evaluation of the fire protection and suppression design features at the OL stage, based on the final design information.

External Fire

USO described external fire hazards in section 2.3.3.1.7 of the PSAR. The staff evaluated fire hazards in section 2.3.7 of the SE, where it was concluded that wildfires do not pose a significant hazard in the region surrounding the KU1 site. As identified in table 3.1-2 of the PSAR, USO qualitatively screened out the external fire hazard. The staff's review of PSAR section 2.3.7 and table 3.1-2 determined that the applicant should reassess the external fire hazard screening at the OL stage to ensure it remains valid and in conformance with the non-LWR PRA standard and RG 1.247. Accordingly, the staff determined that USO's approach to further evaluate potential fires from the sodium test and fill facility (TFF) and EI at OL stage is reasonable because it is consistent with the guidance in RG 1.253.

Section 5.3.2.10 describes the PDC related to control room habitability and the potential impact of external smoke on the habitability of the main control room (MCR). Additionally, PSAR sections 7.5.1, 7.6.7, and 7.8.4 describe design considerations for the MCR. Collectively in these sections, USO addressed the design of the control room envelope and its associated heating, ventilation, and air conditioning (HVAC) system. As described in table 1.4-1 of the PSAR, the applicant clarified that the KU1 design fully conforms to RG 1.189 which ensures control room habitability, 72-hour remote shutdown complex (RSC) habitability, and proper interface with the nuclear island fire protection (NFP) system to maintain HVAC operation during fire events.

The staff evaluated, and confirmed through audit, the screening approach for external fire events provided in PSAR section 2.3.3.1.7 and determined that it is reasonable. The staff also determined that the control room habitability is consistent with PDC 19, based on the preliminary facility design. The staff will review USO's evaluation of the potential fires from the sodium TFF and the EI at the OL stage.

Transportation

The PSAR provides a discussion of nearby transportation routes and incident-related hazards in chapters 2 and 7. PSAR section 2.3 describes the assessment of hazards associated with nearby industrial, transportation, and military facilities and included consideration for waterway, railway, highways, and pipelines in accordance with RG 1.78 and RG 1.91. As described in PSAR section 2.3.3.1.4, none of the examined transportation hazards present a design basis hazard to the KU1 site. The applicant also evaluated each transportation mode to identify chemicals potentially transported along these routes that could constitute a design basis event. The analysis established minimum safe distances for postulated accidents, which were then compared to the proximity of these transportation routes to KU1. The staff's assessment of transportation is documented in section 2.3.3 of this SE. In support of this assessment, the staff conducted an audit that included the supporting detailed analysis of transportation hazards, including the potential chemical release of propane.

Based on its review of the PSAR, as confirmed through audit, the staff determined that the evaluation of potential transportation hazard at KU1 provided in PSAR section 2.3.3.1.4 is reasonably characterized, and that none of the examined transportation hazards present a design basis hazard to KU1 SSCs at the CP stage.

Volcanic Hazards

Volcanic hazards in the vicinity of the KU1 site are assessed and discussed in PSAR section 2.7. Based on its assessment of volcanic phenomena in topical report (TR) NAT-3226-A, "An Analysis of Potential Volcanic Hazards at the Proposed Sodium Site near Kemmerer, Wyoming," Rev. 0A (ML24303A409), which was incorporated by reference into the PSAR, USO concluded that the site is located at sufficient distances from potential future volcanic eruption sources to preclude the following hazard phenomena from impacting the site:

- The potential for the opening of a new volcanic vent
- Proximal hazards
- Lava flows
- Pyroclastic density flows.

Tephra falls, including volcanic ash and debris flows, were assessed and described in sections 8 and 9 of NAT-3226-A. The staff separately reviewed the USO characterization of tephra fall hazards in NAT-3326-A.

The staff reviewed the incorporation by reference of NAT-3226-A and found it to be acceptable in section 2.7 of this SE. As such, the associated conclusions included with the approved version of the TR are incorporated in this SE and referenced in section 2.7 of this SE.

Thunderstorm and Lightning Hazards

Thunderstorm and lightning strike hazards at the KU1 site are assessed and discussed in PSAR chapters 1 and 2. Specifically, PSAR section 1.1.4.3.11 describes that the grounding, earthing, and lightning protection (NGL) system provides protection for plant personnel and equipment from transient over-voltages that can result from electrical faults or lightning strikes. The applicant states the system also provides a proper ground reference for instrumentation signals.

PSAR section 1.1.4.3.11 states the nuclear island lightning protection system is designed to protect exposed buildings, structures, stacks, substations, and associated electrical and electronic circuits from hazards associated with transient over-voltages due to lightning strikes and switching surges. Additionally, as documented in PSAR table 1.4-1, USO clarified that its approach conforms to RG 1.204 for lightning protection.

Based on the staff's review of PSAR section 1.1.4.3.11 and its determination documented in SE chapter 1, the staff determined that the protective design features for thunderstorm and lightning hazards are appropriately considered and in place and are therefore consistent with PDC 2. The staff will review USO's complete thunderstorm and lightning hazard protection design at the OL stage, based on the final design information.

Aircraft Impact Accident

PSAR table 3.1-2 indicates that the aircraft impact accidents are quantitatively screened out of the PRA. The staff audited the screening process and confirmed that the applicant's internal documentation supported the PSAR statement. However, the applicant should reassess the screening of aircraft impact accident at the OL stage to ensure it remains valid and in conformance with the non-LWR PRA, should the applicant apply for an OL.

6.1.1.2 Safety Related Design Criteria

PSAR section 6.1.2 states that the SRDC are provided in PSAR tables 5.2-1, 5.2-2 and 5.2-3 to show their connection to the SR functions. SR functions are described in PSAR section 5.2.1 and section 5.4 of this SE. PSAR table 5.2-1 lists the SR functions supporting control of heat generation and the associated SRDC, table 5.2-2 lists SR functions supporting control of heat removal and the associated SRDC, and table 5.2-3 lists SR functions supporting retaining radionuclides and the associated SRDC. The staff evaluation of the SR function and the associated preliminary SRDC is provided in chapter 7 of this SE, in the section(s) that correspond to the SRDC. Table 6.1-1 of this SE provides the SRDCs and their corresponding systems and sections within the PSAR and the SE.

6.1.1.3 DBHL-related Requirements for Non-Safety-Related SSCs

In PSAR section 6.1.3, the applicant describes the design requirements for NSRST and NST SSCs such that the DBHLs will not cause an NSRST or NST SSC to adversely impact the ability of an SR SSC to perform an SR function. The applicant states that the design of the facility will prevent DBHL-related interactions (seismic and tornado) between NSRST or NST SSCs and SR SSCs. The focus of the staff's review of PSAR section 6.1.3 was the potential for seismic or tornado missile interactions between SR and non-safety-related SSCs. Seismic design of NSRST SSC to perform safety-significant functions and protection of NSRST SSCs from seismic interaction are covered in PSAR section 6.4.1 and evaluated by the staff in section 6.4.1.1 of this SE.

PSAR section 6.1.3.1 describes how potential seismic interactions are identified through an area-based review process that treats safety-significant SSCs as seismic interaction targets and NSRST and NST SSCs as potential seismic interaction sources. USO states that the source SSCs will be designed to ensure that interactions will not lead to adverse consequences and will meet the LMP risk target (including in the BDBE region), which will be demonstrated at the OL stage by the seismic PRA (SPRA). PSAR section 6.1.3.1 states credible seismic interactions are identified if the target is within a specified physical zone of influence around the source. USO states that if physical rearrangements and barriers cannot prevent interactions, then seismic interaction prevention design requirements are applied.

USO stated that to prevent seismic-related DBHL interactions, NSRST and NST SSCs with seismic interaction prevention design requirements will be designed in accordance with ASCE/SEI 7-16 at the SCN1 seismic classification level. The parameters for SCN1 are provided in PSAR table 6.4-1B and defined in PSAR section 6.4.1.1. While ASCE/SEI 7-16 is the code of record, a supplemental evaluation is performed by the applicant to ensure that source SSCs are designed with a performance level greater than or equal to that of the target SSC and with inelastic deformation at or below that corresponding to Limit State B in ASCE/SEI 43-19. PSAR section 6.1.3.1 states that if the Limit State B capacity is not initially met, then additional design requirements, beyond the base ASCE/SEI 7-16 requirements, are added for these SSCs until they meet ASCE/SEI 43-19 Limit State B in this iterative design process.

The PSAR states that these additional design requirements could include, for example: limiting structural systems to those endorsed by ASCE/SEI 43-19, including ductile detailing in design, or limiting ASCE/SEI 7-16 Response Modification Coefficients (R and R_p) based on ASCE/SEI 43-19 inelastic energy absorption factors (F_p) for Limit State B. The applicant also evaluates the seismic separation between NSRST or NST structures and adjacent SR SSCs using the methods in ASCE/SEI 43-19 section 7.3. This methodology contributes to meeting the requirements of PDC 81, providing a methodology to prevent the RXB superstructure from impacting the RAC and its ability to passively remove residual heat from the RES.

In addition to the physical interactions between sources and targets, USO will also treat sodium-containing SSCs as special seismic interaction targets in their analysis of seismic interactions because of the potential for seismic-induced fires. Sodium-containing SSCs will be designed to retain their sodium inventory following the SSE. USO states in PSAR section 6.1.3.1 that this seismic performance requirement for sodium retention will be achieved either through their design (a performance level equivalent to SR SSCs) or alternative means. Additional details on sodium retention seismic interaction design requirements are expected at the OL stage.

With seismic interaction prevention requirements being applied separately from SSC classification, there could be NST SSCs whose failure could lead to an event with consequences exceeding the F-C target curve based on seismic interaction with an SSC performing a safety-significant function. To address the safety concerns associated with this, USO stated in PSAR section 6.1.3.1 that seismic interaction sources have design, analysis, and appropriate programmatic controls applied at a level consistent with NSRST SSCs. Programs that USO plans to apply to SSCs with seismic interaction prevention requirements include the Quality Assurance Program Description (QAPD), the Post-Construction Inspection, Testing, and Analysis Program (PITAP), and the Sodium Maintenance Program.

For tornado DBHL interaction (PSAR section 6.1.3.2), the applicant will design NSRST/NST SSCs to the provisions of ASCE/SEI 7-16, and the tornado and tornado missile parameters specified in RG 1.76 (PSAR table 1.4-1). Specifically, these parameters include design basis tornado hazard wind speed, atmospheric pressure drops, and tornado missiles for the design of NSRST/NST SSCs as presented in PSAR table 2.1-1. As discussed in PSAR section 7.8.2.2.3, USO has applied additional design requirements from American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities" and American Concrete Institute (ACI) 349-13, "Nuclear Safety-Related Concrete Structures," to the NSRST fuel handling building superstructure for determining permissible ductility and displacements to absorb the strain energy due to tornado-generated missiles. In addition, concrete anchors used to anchor NSRST structures to SR concrete are designed according to ACI 349-13, as stated in PSAR section 6.4.2.

The staff finds USO's DBHL-related requirements with regard to the seismic and tornado interaction for non-safety-related SSCs that could have adverse interactions with SR SSCs acceptable for the CP application based on the following:

- The process of identifying target and source pairs based on a physical zone of influence appears to be a systematic and thorough method for identifying SSCs needing seismic interaction prevention requirements to ensure their failure will not prevent a safety-significant SSC from performing its PRA safety function. This evaluation is consistent with the guidance in RG 1.233.
- The application of ASCE/SEI 7-16, along with the supplemental evaluation to ensure that inelastic deformation will not exceed what is allowed under ASCE/SEI 43-19 Limit State B, and confirmation of adequate separations using ASCE/SEI 43-19 section 7.3 methods provide assurance at the CP stage that seismic interaction will not impact safety-significant SSCs for design basis loading. This design approach, including additional design requirements (i.e., to meet ASCE/SEI 43-19 Limit State B), should be confirmed by USO through the seismic PRA at the OL stage.
- Appropriate programmatic control, including QAPD requirements consistent with NSRST SSCs, have been applied to ensure that the critical characteristics associated with the seismic interaction prevention requirements are maintained from design through construction and operation.
- The design approach for the NSRST/NST SSCs for tornado interaction is adequate for the preliminary design because USO proposed high wind and tornado site characteristics, which are listed in PSAR table 2.1-1, based on those in table 1 (region 3)

of RG 1.76. The staff's review of local climatology is provided in section 2.4.1.1 of this SE.

6.1.2 Conclusion

Based on the evaluations described in the previous subsections, the staff concludes that USO has reasonably characterized the DBHLs and that its design approach provides assurance that SR and NSRST SSCs will withstand the effects of natural and man-made hazards without loss of safety function. Accordingly, the staff finds that the information in the PSAR satisfies the applicable requirements of 10 CFR 50.34(a) and 10 CFR 50.35 for issuance of a CP.

Specifically, the staff determined the information summarized in PSAR section 6.1 is reasonable for USO to define the design requirements for SR SSCs, the DBHLs, the associated SRDC, and DBHL requirements for non-safety-related SSCs, for the following reasons:

1. Regarding the external hazards, USO assessed and developed DBHL parameters for high winds and tornadoes, external flooding, missile hazard, seismic hazard, extreme winter precipitation, volcanic hazard, transportation hazard, and thunderstorms and lighting as described in multiple chapters of the PSAR. In addition to the staff's evaluation provided in this section, the staff's review of these hazards is also documented in chapter 2 of this SE.
2. Regarding the internal flood hazard, USO stated that the SR SSCs are designed to withstand the design basis internal flooding with no adverse impact on their capability to perform their SR functions. Internal flooding is a hazard that will be assessed at the OL stage. The staff determined that it is reasonable to address flooding hazards from design basis pipe rupture with physical separation of safety-significant SSCs, guard piping, protective barriers, and fluid collection systems, or by demonstrating that safety-significant SSCs are not adversely affected should flooding occur.
3. Regarding the internal fire hazard, USO stated that the design basis internal fire hazard will be evaluated in accordance with RG 1.189, Rev. 5, including potential fire impacts from sodium piping leakage or rupture, at the OL stage. The staff determined that the applicant's approach to internal fire hazard is consistent with RG 1.253 guidance.
4. Regarding the design basis missile hazard, USO assessed tornado-generated missile hazards, as documented in PSAR section 2.4.1.3.3. USO is conforming to the guidance in RG 1.76 and provided necessary information to address the missile hazard at the KU1 site, including consideration of interactions between NSRST and NST SSCs and SR SSCs.
5. Regarding the seismic and tornado interactions, the staff determined, based on the evaluation above in section 6.1.1.3 of this SE, that the applicant's approach to seismic and tornado interaction effects for non-SR SSCs provides assurance at the CP stage that potential interactions during the SSE and tornado DBHL will be adequately addressed. Seismic design requirements for SR and NSRST SSCs to ensure performance of safety-significant functions are documented in section 6.4.1 of this SE.

Table 6.1-1: Safety-Related Design Criteria from the KU1 PSAR and Associated PSAR and SE Sections

PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
DL3-RC1	Scram - Gravity Driven Absorber Insertion by Latch Release	Upon receipt of a scram signal from RPS, the Control Rod Drive System shall release the control rod assemblies in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Control Rod Drive System (CRD)	7.2.5
		Upon CRD rod release, control rod assemblies shall insert into the core to a depth and within a time limit to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition	Reactor Core System (RCC)	7.1.1
		Upon CRD rod release, RES shall provide structural support and position control preventing binding of CRDL and ensure control rod release is uninhibited to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Enclosure System (RES)	7.1.2
		The RPS shall receive monitored signals and generate a scram signal upon exceeding a scram setpoint in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Protection System (RPS)	7.6.3
		XIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Nuclear Instrumentation System (XIS)	7.6.4
		RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Instrumentation System (RIS)	7.6.5

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
DL3-RC2	Reactor Scram on Loss of Power	Upon receipt of a scram signal from RPS, the Control Rod Drive System shall release the control rod assemblies in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Control Rod Drive System (CRD)	7.2.5
		Upon power loss, the RPS shall fail safe by generating a scram signal in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Protection System (RPS)	7.6.3
DL3-HR1	PSP Coastdown	The Primary Sodium Pump (PSP) shall coast down upon a pump trip to support transition to natural circulation, without flow reversal within the fuel channels, to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Primary Heat Transport System (PHT)	7.1.3
DL3-HR2	PSP Trip on High High Primary Sodium Temperature	Upon receiving a trip signal from the RPS, the PSPs shall trip to eliminate heat generation to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Primary Heat Transport System (PHT)	7.1.3
		The RPS shall receive monitored signals and generate a PSP trip signal upon exceeding a high temperature limit with a low flux signal in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Protection System (RPS)	7.6.3
		XIS and RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Nuclear Instrumentation System (XIS)	7.6.4
		XIS and RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Instrumentation System (RIS)	7.6.5

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
DL3-HR3	ISP Trip on High High Primary Sodium Temperature	Upon receiving a trip signal from the RPS, the Intermediate Sodium Pumps (ISP) shall trip to eliminate heat generation to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Intermediate Heat Transport System (IHT)	7.1.4
		The RPS shall receive monitored signals and generate an ISP trip signal upon exceeding a high temperature limit with a low flux signal in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Protection System (RPS)	7.6.3
		XIS and RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Nuclear Instrumentation System (XIS)	7.6.4
		XIS and RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Instrumentation System (RIS)	7.6.5
DL3-HR4	Inherent - RAC Operation	RAC shall continuously transfer heat to the atmosphere via natural circulation at a rate during accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Air Cooling System (RAC)	7.2.1
		RAC shall continuously transfer heat to the atmosphere via natural circulation at a rate during accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Enclosure System (RES)	7.1.2
		RAC shall continuously transfer heat to the atmosphere via natural circulation at a rate during accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Building (RXB)	7.8.1

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
DL3-HR5	Natural Circulation of Sodium in Primary System	The primary flow circuit elevations and geometries shall promote natural circulation to remove heat from the core at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Primary Heat Transport System (PHT)	7.1.3
		The primary flow circuit elevations and geometries shall promote natural circulation to remove heat from the core at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
		The primary flow circuit elevations and geometries shall promote natural circulation to remove heat from the core at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Enclosure System (RES)	7.1.2
DL3-HR6	Passive Heat Removal in EVHM	The Ex-Vessel Handling Machine (EVHM) shall passively remove decay heat from spent fuel assemblies at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
DL3-HR7	Passive Heat Removal in EVST	The Ex-Vessel Storage Tank (EVST) shall passively remove decay heat from spent fuel assemblies at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
		Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
DL3-HR8	Passive Heat Removal in BLTC	The Bottom Loading Transfer Cask (BLTC) shall passively remove decay heat from spent fuel at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
DL3-HR9	Passive Heat Removal in PRC	The Pin Removal Cell (PRC) shall passively remove decay heat from spent fuel at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
DL3-HR10	Passive Heat Removal in FHP	The spent fuel pool shall passively remove decay heat from spent fuel at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Water Fuel Pool Handling System (FHP)	7.3.1

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
		Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
DL3-HR11	SPS Pump Trip on Low Low Primary Sodium Level	Upon receiving a trip signal from the RPS, the SPS pumps shall trip in time to maintain reactor vessel primary sodium level high enough to establish reasonable assurance that natural circulation in the primary sodium flow circuit can develop.	Sodium Processing System (SPS)	7.2.4
		RIS shall send monitored signals to the RPS in time to maintain reactor vessel primary sodium level necessary to establish reasonable assurance that natural circulation in the primary sodium flow circuit can develop.	Reactor Instrumentation System (RIS)	7.6.5
		The RPS shall monitor received signals, and if primary sodium level exceeds the setpoint after a scram shutdown, the RPS shall trip the SPS pumps in time to maintain reactor vessel primary sodium level high enough to establish reasonable assurance that natural circulation in the primary sodium flow circuit can develop.	Reactor Protection System (RPS)	7.6.3
DL3-HR12	ISP Pump Trip on High High Primary Sodium Level	Upon receiving a trip signal from the RPS, the ISPs shall trip in time to prevent over-pressurization and failure of the primary boundary, to establish reasonable assurance primary inventory is maintained, and establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Intermediate Heat Transport System (IHT)	7.1.4
		XIS and RIS shall send monitored signals to the RPS in time for the plant to respond to prevent over-pressurization and failure of the primary boundary, resulting in a safe shutdown condition. The functions will provide reasonable assurance primary inventory is maintained and any associated radiological dose calculation results are maintained under the 10 CFR 50.34 dose criteria.	Reactor Instrumentation System (RIS)	7.6.5

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
		XIS and RIS shall send monitored signals to the RPS in time for the plant to respond to prevent over-pressurization and failure of the primary boundary, resulting in a safe shutdown condition. The functions will provide reasonable assurance primary inventory is maintained and any associated radiological dose calculation results are maintained under the 10 CFR 50.34 dose criteria.	Nuclear Instrumentation System (XIS)	7.6.4
DL3-HR12 (cont.)	ISP Pump Trip on High High Primary Sodium Level	The RPS shall monitor received parameters, and if primary level exceeds the setpoint after a scram shutdown, the RPS shall trip the ISPs. This function shall occur in time to prevent over-pressurization and failure of the primary boundary, which provides reasonable assurance the primary inventory is maintained, and radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Protection System (RPS)	7.6.3
DL3-HR13	Passive Heat Removal in the Failed Fuel Canister	The Failed Fuel Canister shall passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that the radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Water Fuel Pool Handling System (FHP)	7.3.1
		Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
DL3-RR1	Primary Coolant Boundary including RES Barrier	The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Enclosure System (RES)	7.1.2
		The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological	Primary Heat Transport System (PHT)	7.1.3

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
		dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.		
		The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Sodium Processing System (SPS)	7.2.4
DL3-RR1 (Cont.)	Primary Coolant Boundary including RES Barrier	The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Sodium Cover Gas System (SCG)	7.2.3
		The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Control Rod Drive System (CRD)	7.2.5
		The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	In-Vessel Fuel Handling System (FHI)	7.3.3
DL3-RR2	Cladding barrier	During accident conditions, the fuel cladding shall retain radionuclide fission products to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Core System (RCC)	7.1.1
DL3-RR3	EVHM Cask barrier	During postulated accidents, the EVHM, in a standalone state, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
		During postulated accidents, when the EVHM is joined with the reactor head, the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		During postulated accidents, when the EVHM is joined with the EVST, the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
DL3-RR4	EVST Barrier	During postulated accidents the EVST, in a standalone state, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		During postulated accidents, when the EVST is joined with the BLTC, the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		The EVST Auxiliary Systems, up to and including the isolation valves, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		During postulated accidents the EVST, in a standalone state, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Sodium Processing System (SPS)	7.2.4

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
		The EVST Auxiliary Systems, up to and including the isolation valves, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Sodium Processing System (SPS)	7.2.4
DL3-RR5	BLTC barrier	During postulated accidents the BLTC, in a standalone state, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		During postulated accidents, when the BLTC is joined with the Pool Immersion Cell (PIC), the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
DL3-RR5 (Cont.)	BLTC barrier	During postulated accidents, during all fuel movements in and out of the BLTC, the installed equipment shall temporarily contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2
		During postulated accidents, during all fuel movements in and out of the BLTC, the installed equipment shall temporarily contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Water Pool Fuel Handling System (FHP)	7.3.1
DL3-RR6	PRC Cell barrier	The PRC shall contain radionuclide releases from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Ex-Vessel Fuel Handling System (FHE)	7.3.2

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PRA Safety Function	Function Description	SRDC from PSAR Table 5.2-1 through 5.2-3	SSC	PSAR/SE Section
DL3-RR7	RES Pressure Relief	Prior to reaching the RES and supporting system functional containment design pressure, SCG shall actuate the primary pressure relief valve to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Sodium Cover Gas System (SCG)	7.2.3
		Prior to reaching the RES and supporting system functional containment design pressure, SCG shall actuate the primary pressure relief valve to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Reactor Enclosure System (RES)	7.1.2
DL3-RR8	Failed Fuel Canister Barrier	During accident conditions, the Failed Fuel Canister containing failed fuel shall retain radionuclide fission products to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition.	Water Pool Fuel Handling System (FHP)	7.3.1

6.2 Reliability and Capability Targets for Safety-Significant SSCs

The staff reviewed the USO's assessment of reliability and capability targets for safety-significant SSCs provided in PSAR section 6.2. In addition, the staff audited the specific reliability targets supporting documentation which present reliability targets for a range of safety-significant SSCs identified through the applicant's probabilistic risk assessment (PRA) and design processes.

The design reliability assurance program (D-RAP), as described by USO, is intended to ensure that safety-significant SSCs are designed, procured, constructed, operated, and maintained to meet reliability and availability levels commensurate with their safety importance, consistent with NRC guidance mentioned in the previous section.

For each SSC identified as safety-significant, the applicant established preliminary quantitative reliability targets intended to reflect the level of performance necessary to ensure the successful completion of the associated safety functions under both design-basis and beyond-design-basis conditions. These targets are documented in the D-RAP tables and are intended to guide design decisions, procurement specifications, equipment qualification, testing and inspection programs, and maintenance activities.

Regarding capability targets, in response to the staff's audit question, USO stated that these targets are derived from the PRA's success criteria analysis, the integrated plant analysis of licensing basis events (LBEs), and by source term analyses that influence event consequence evaluations. The success criteria and other LBE analyses serve as the starting point for capability target development, as they establish the SSC performance necessary to meet the applicable risk metrics. Where appropriate, the capability targets also account for the conditions under which the SSCs must perform their functions during the LBEs, as well as the timeframes within which active SSCs must operate to mitigate the LBEs.

The regulatory requirements applicable to the staff's review of PSAR section 6.2 are:

- 10 CFR 50.34(a); and
- 10 CFR 50.35.

Applicable guidance for the review of PSAR section 6.2 includes:

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition—Severe Accidents," section 17.4, "Reliability Assurance Program," Rev. 1 (ML13296A435)
- RG 1.233, Rev. 0
- RG 1.253, Rev. 0

6.2.1 Technical Evaluation

The staff audited supporting documentation to confirm the description of reliability and capability targets in PSAR section 6.2.

Based on its review, the staff determined that the general approach described in the D-RAP, identifying safety-significant SSCs, establishing reliability targets, and integrating these targets into the design control and assurance process, is consistent with NRC principles for a graded approach to reliability assurance. This integrated approach provides a framework for ensuring that performance expectations for SSCs are defined early in the design process and maintained throughout subsequent design, procurement, construction, and operational phases.

However, the staff identified several areas where additional information is needed at the OL stage. Specifically, the staff noted that certain reliability targets were set below the typical values established for comparable safety functions in other reactor designs. At the CP stage, there are insufficient technical justifications to fully substantiate the adequacy of these lower targets. In addition, because the current PRA is preliminary and the SSC designs are not yet finalized, the determination on the proposed targets and if they are technically achievable or provide sufficient margin to ensure reliable performance of the associated safety functions will occur at the OL stage.

Accordingly, at the CP stage, the staff evaluated the adequacy of the process to determine frequency and probability values for reliability targets were acceptable. The finding on the acceptability of the specific frequency or probability values for reliability targets will be subject to further review at the OL stage, when more detailed design information, including final SSC configurations, failure modes analyses, and supporting analyses, will be available.

With respect to capability targets, the PSAR indicates that the specific operational performance criteria are included as part of the profile of the success of a PRA safety function. These quantitative targets are based on analytical models used to demonstrate the capability of the plant to respond to hazards. The success criteria define the physical characteristics of systems or design features that are required to meet the safety functions defined. At the preliminary design stage reflected in the PSAR, these may still be at the system level with characteristics like total leakage rate specified, but as the design progresses specific characteristics for each SSC that contributes to the success of the PSF will need to be determined. These become the capability targets for those SSCs, relied upon to meet risk metrics. The staff confirmed in audit space that the process USO is using to develop the capability targets from the PRA success criteria, safety analysis, and source term analysis as described in the PSAR, is reasonable.

The staff expects the applicant to provide, during the OL stage, the specific reliability, availability, and capability targets for safety-significant SSCs, including:

- The technical basis for each target;
- How the targets are derived from PRA and deterministic analyses;
- The verification and validation methods; and
- The programs and processes to ensure these targets are maintained throughout the plant's operational life.

The staff recognizes that, given the preliminary level of design detail at the CP stage, it is reasonable to leave the finalization of the reliability, availability, and capability targets to the OL stage when sufficient design, procurement, and system integration information is available.

6.2.2 Conclusion

Based on the review described above, the staff determined that USO's general approach to establishing reliability and capability targets for safety-significant SSCs, and implemented through the D-RAP, is reasonable for the CP stage. As discussed above, the applicant will need to fully demonstrate the target justification and quantitative capability criteria during the OL stage, based on the full-scope PRA developed in conformance with the non-LWR PRA standard.

The staff recognizes that, given the preliminary level of design detail at the CP stage, it is reasonable to leave the finalization of the reliability, availability, and capability targets to the OL stage when sufficient design, procurement, and system integration information is available. At that time, the staff will verify whether the reliability, availability, and capability targets derived from the PRA and other deterministic approaches are appropriately justified and acceptable to ensure continued compliance throughout the facility's operational life.

Accordingly, the staff determined that the applicant's derivation of reliability and capability targets, through the D-RAP and success criteria analysis, is an acceptable approach for the CP stage and expects the applicant to address the following at the OL stage:

- The preliminary reliability targets provided are reassessed, verified, and validated;
- Technical justifications for certain less stringent reliability targets are fully developed;
- Quantitative capability targets for safety-significant SSCs are defined and documented;
- The RAP integrated decision process panel will reevaluate and confirm the established targets, and
- The applicant demonstrates that these targets are achievable and will be maintained throughout the design, construction, and operational phases of the plant.

6.3 Special Treatment Requirements for Safety-Significant SSCs

PSAR section 6.3 describes the special treatment requirements for safety-significant SSCs (i.e., SR and NSRST SSCs). Special treatments are those requirements that provide increased assurance beyond normal industrial practices that safety-significant SSCs perform their design basis functions. Special treatments are developed to provide reasonable confidence that safety-significant SSCs meet the reliability and capability targets.

The regulatory requirements applicable to the staff's review of PSAR section 6.3 are:

- 10 CFR 50.34(a); and
- 10 CFR 50.35.

Applicable guidance for the review of PSAR section 6.3 includes:

- DANU-ISG-2022-01;
- RG 1.233, Rev. 0; and
- RG 1.253, Rev. 0.

6.3.1 Technical Evaluation

PSAR section 6.3 indicates that preliminary special treatments applied to safety-significant SSCs are identified in PSAR chapter 7 considering the PRA safety functions of the SSC, as well as the safety-significance, risk significance, and equipment type of the SSC. The PSAR further states that special treatments will be finalized and applied to safety-significant SSCs once the reliability and capability targets are finalized. The final special treatments will be provided at the OL stage of the review.

The staff reviewed the information in PSAR section 6.3 and audited supporting documentation related to the process for selecting and applying special treatments. The staff compared the information in PSAR section 6.3, as well as the audited reports, with the guidance in NEI 21-07 as endorsed by RG 1.253 and determined that the process for establishing special treatments is consistent with the guidance.

6.3.2 Conclusion

Based on its review described above, the staff determined the preliminary information regarding the process of identifying special treatments requirements is consistent with RG 1.253 and acceptable for the CP review.

6.4 Design of Safety-Significant SSCs

PSAR section 6.4 provides design attributes for safety-significant SSCs (i.e., SR and NSRST SSCs). Subsections within this chapter include:

- 6.4.1 Seismic Design
- 6.4.2 Design of Safety-Significant Structures
- 6.4.3 Mechanical Systems and Components
- 6.4.4 ASME BPVC, Section III Piping Systems, Piping Components, and Associated Supports
- 6.4.5 Application of Industrial Codes

The applicable regulatory requirements for the design of safety-significant SSCs are:

- 10 CFR 50.34(a),
- 10 CFR 50.35, and
- 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

The applicable guidance for the review of PSAR section 6.4 includes:

- DANU-ISG-2022-01, March 2024
- DC/COL ISG-017, "Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses," March 2010 (ML100570203)

- RG 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” Rev. 3 (ML17094A831)
- RG 1.29 “Seismic Design Classification for Nuclear Power Plants,” Rev. 6 (ML21155A003)
- RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” Rev. 2 (ML2328/ML23284A272)
- RG 1.87, “Acceptability of ASME Section III, Division 5, High Temperature Reactors,” Rev. 2 (ML21091A276)
- RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Rev. 3 (ML12220A043)
- RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” Rev. 1 (ML003739367)
- RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” Rev. 3 (ML20141L613)
- RG 1.166, “Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake,” Rev. 1 (ML19266A616)
- RG 1.199, “Anchoring Components and Structural Supports in Concrete,” Rev. 1 (ML19336A079)
- RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” Rev. 0 (ML110940300)
- RG 1.243, “Safety-Related Steel Structures and Steel-Plate Composite (SC) Walls for other than Reactor Vessels and Containments,” Rev. 0 (ML21089A032)
- NUREG-0800
 - Section 3.5.1.1, “Internally Generated Missiles (Outside Containment),” Rev. 3 (ML070370569)
 - Section 3.5.1.2, “Internally Generated Missiles (Inside Containment),” Rev. 3 (ML070380167)
 - Section 3.5.1.3, “Turbine Missiles,” Rev. 3 (ML063600395)
 - Section 3.5.3, “Barrier Design Procedures,” Rev. 3 (ML070570004)
 - Section 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” Rev. 3 (ML070550032)
 - Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping,” Rev. 3 (ML16088A041)

- Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” Rev. 1 (ML063600396)
- Section 3.7.2, “Seismic System Analysis,” Rev. 4 (ML13198A223)
- Section 3.9.3 “ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures” Rev. 3 (ML14043A231)

The applicable PDC that apply to the review of this section are:

- PDC 1 “Quality Standards and Records”
- PDC 2, “Design Bases for Protection Against Natural Phenomena”
- PDC 4, “Environmental and Dynamic Effects Design Bases”
- PDC 14, “Primary Coolant Boundary”
- PDC 31, “Fracture Prevention of Primary Coolant Boundary”
- PDC 80, “Reactor Vessel and Reactor System Structural Design Basis”
- PDC 81, “Reactor Building Design Basis”

6.4.1 Technical Evaluation

6.4.1.1 Seismic Design

PDC 2 requires that safety-significant SSCs are designed to withstand the effects of natural phenomena including earthquakes. USO is using a graded approach to seismic design considering the safety significance of SSCs. The performance targets based on the seismic classification and corresponding design approach will be modeled within a SPRA. An iterative process will then be used by USO to determine if requirements should be raised or can be lowered. Initial seismic classifications are assigned at the CP stage and will be confirmed or adjusted for the OL stage based on the SPRA. The final performance targets and special treatments are determined by iterating between the design of SSCs and the SPRA until the overall seismic risk for the facility is acceptable.

External hazard PRAs, including the SPRA, will be developed by USO for the OL application, but the external hazard PRAs are not required at the CP stage. Review of the USO seismic approach focused on understanding the basis for selection of initial seismic classifications, codes and standards, and special treatments to provide reasonable assurance that the USO’s approach provides an adequate level of safety at the CP stage. The final SPRA developed at the OL stage will confirm that the facility can be constructed and operated safely.

With respect to seismic instrumentation, PSAR section 6.4.1 refers to section 1.1.4.3.16, which states that the seismic monitoring system (SMS) equipment is designed, built, and tested in full conformance with RG 1.12. The PSAR further states that the SMS provides time history acceleration data in the free-field and safety-related structures and indicated that acceleration

sensors are located in the NCB, RXB, FHB, and NI yard areas. In addition, the SMS provides the data necessary to directly compare the measured motion with both site-specific ground motion response spectra (GMRS) and the design in-structure response spectra to evaluate the seismic response of nuclear power plant features important to safety promptly after an earthquake as required by 10 CFR 50 Appendix S.

The staff reviewed PSAR section 1.1.4.3.16 and determined that more specificity will be needed at the OL stage regarding the actual locations of all proposed instrumentation as well as the bases for these locations. In addition to the instrument type and locations, a discussion regarding instrument operability, characteristics, installation, remote indication, and maintenance (per RG. 1.12) should be provided at the OL stage. The staff also determined that more specificity regarding the proposed shutdown criteria (per RG 1.166), including the operating basis earthquake (OBE) and SSE exceedance criteria, will be needed at the OL stage. The staff determined that this information can reasonably be left to the OL stage of the review, and the staff will review USO's seismic instrumentation program at that time. Additionally, as described in section 6.4.1.3 of this SE, justification for whether additional seismic monitoring is needed on the isolated portion of the facility will also be reviewed at the OL stage.

6.4.1.1.1 Seismic Classification of SSCs

The staff reviewed PSAR section 6.4.1 and tables 6.4-1A and B and 6.4-2 to understand the preliminary approach to seismic classification of SSCs and corresponding seismic design requirements. Safety-related SSCs are classified by USO as either SCS1 or SCS2. USO's preliminary seismic classification approach for SR SSCs is to assign SCS1 to all SR SSCs unless failure of the SR SSC in a seismic event does not adversely affect the safety significant function, in which case the SR SSC is assigned an SCS2 classification.

USO will design SR SSCs to meet the requirements in Appendix S to 10 CFR Part 50. An SCS1 classification imposes seismic requirements consistent with the Seismic Category I classification from RG 1.29. USO applies the following codes and standards to SCS1 and SCS2 SSCs, which are consistent with historical SR SSC design for light water reactors:

- ASCE/SEI 4-16, "Seismic Analysis of Safety-Related Steel Structures for Nuclear Facilities."
- ACI 349-13, "Building Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary."
- American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities."

The difference between SCS1 and SCS2 SR SSCs is the qualification criteria. SCS1 qualification requires reasonable assurance of function under seismic loads consistent with SC-1 SSCs described in RG 1.29, while SCS2 qualification requires reasonable confidence of function under seismic loads consistent with ASCE/SEI 7-16.

According to USO's methodology and classification scheme as summarized in PSAR table 6.4-1B and section 6.4.1.1, NSRST SSCs are classified as either SCN1, SCN2, or SCN3.

The initial seismic classification approach for NSRST SSCs assigns SCN3 as a minimum. The seismic classification is then increased to SCN1 or SCN2 as appropriate based on factors such as SSC function, importance to life safety, and potential contribution to events with dose consequences. SCN1 will be assigned as the seismic classification for active components and for SSCs identified as being seismic risk significant. Seismic risk significance is determined by evaluating the role of the SSC in preventing or mitigating LBEs and the potential unmitigated dose consequences from a common cause seismic failure. SCN2 or SCN1 are assigned as appropriate to meet minimum building code requirements. SCN3 will not be applicable to structures but is reserved for non-structural systems or components.

USO performed the seismic design of NSRST structures in accordance with the following commercial industrial codes and standards:

- International Building Code (IBC) 2021, “International Building Code”
- ASCE/SEI 7-16, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures”
- ACI 318-19, “Building Code Requirements for Structural Concrete”
- ANSI/AISC 360-16, “Specification for Structural Steel Buildings”

The staff determined that the approach to initial seismic classification is reasonable, and that the design codes associated with each seismic classification are reasonable based on the risk-informed approach described in NEI 18-04, as endorsed in RG 1.233. USO committed to confirm or adjust the initial seismic classifications and associated design criteria at the OL stage based on the seismic PRA, which provides reasonable confidence that the SSCs will meet the intended performance targets when evaluated at the OL stage using the SPRA.

6.4.1.1.2 Seismic Design Basis

PSAR section 6.4.1.2.1 states that the seismic design for SR SSCs is based on the SSE. The development of the SSE design response spectra is described in PSAR section 2.6.2. The site horizontal and vertical SSE response spectra are shown in PSAR figure 2.6-81 and the spectral acceleration data is provided in PSAR table 2.6-15. The applicant stated in PSAR section 2.6.2.4.2 that the site GMRS, generated at 15 ft below the ground surface, is selected as the SSE. The staff reviewed and determined the applicant’s information was acceptable regarding the development of the GMRS in SE section 2.6.2.4. USO did not develop a separate seismic analysis involving the OBE in accordance with RG 1.61, because the OBE is one-third of the SSE.

For seismic design, USO stated in PSAR section 6.4.1.2.2 that distinct foundation input response spectra (FIRS) are developed for the RXB, FHB, and NCB. The full column method, which includes soil layers above the respective foundation levels, was used to develop the FIRS. USO stated in PSAR section 2.6.4.10.1 that the shallow foundations are constructed on compacted backfill and weathered rock and the deeper foundations are constructed on fresh rock. The dynamic properties of soil layers, shear wave velocities, and strain-compatible shear moduli and damping values required for the full column method are discussed in PSAR section 2.6.4.2.9 and evaluated by the staff in SE section 2.6.4.

To implement the full column method, a soil column analysis is typically performed using one-dimensional equivalent-linear wave propagation analysis. This approach is accepted as standard practice in geotechnical engineering for developing FIRS, in accordance with ASCE 4-16, and consistent with staff guidance in NUREG-0800 (Section 3.7.2) and DC/COL ISG-017 for the seismic design of nuclear power plants. According to PSAR section 6.4.1.2.2, the FIRS are developed at distinct foundation levels of below grade SR structures. The details of where the FIRS are developed for each of the substructures will be provided and reviewed at the OL stage. The staff will confirm at the OL stage that the final design is based on the methodology described and that the applicant used site specific geotechnical data for developing seismic design input.

According to PSAR section 6.4.1.2.3, USO will develop the seismic response spectra for NSRST SSCs in accordance with ASCE/SEI 7-16 considering site-specific ground motion parameters. USO also stated that the response spectra for SCN1, SCN2, and SCN3 will envelope both the SSE and the risk-targeted maximum considered earthquake (MCE_R). The specific factors used to modify the SSE per ASCE/SEI 7-16, including importance factors, are listed in PSAR table 6.4-1B. Response Modification Coefficients depend on the type of SSC being analyzed, as defined in tables 12.2-1, 12.14-1, 15.4-1, and 15.4-2 of ASCE/SEI 7-16.

The staff determined that the development of the design ground motions for SR and NSRST SSCs is reasonable because, i) they rely on the seismic hazard analysis that was reviewed and accepted by the staff in SE section 2.6.2, and ii) USO is following guidance in RG 1.61 to define the OBE based on the SSE. Furthermore, USO will develop the FIRS at the respective foundation levels for substructures based on the SSE. The specific FIRS will be developed and evaluated at the OL stage. The staff are evaluating the MCE_R with respect to the review of the specific SSCs that these are applied to as described in sections 6.4.1.1.5 and 6.4.1.2.2 of this SE.

6.4.1.1.3 Seismic Design and Analysis of SR SSCs

USO's methodologies for seismic analysis and design of SR SSCs are detailed in PSAR section 6.4.1.3 with application of the methodology to the preliminary design of the SR buildings and structures detailed in PSAR section 7.8. The seismic analyses for SR SSCs are based on the equivalent static method or dynamic analysis.

Dynamic analysis methods include response spectrum analysis, time history analysis, or random vibration theory. Response spectrum and time history analysis methods are performed in accordance with guidance in RG 1.92. For time history analysis, earthquake components are applied in three directions, either in separate analyses or concurrently in a single analysis. The random vibration theory is based on ANSI/ASCE 4-16 for soil structure interaction (SSI) analysis.

USO will apply the equivalent static load method described in the PSAR for SR SSCs per ASCE/SEI 4-16 if i) the SSC can be represented by a simple model and it produces conservative results, ii) the method accounts for relative motion between points of support, and iii) the dynamic amplification factor, per section 4.5.1 of ASCE/SEI 4-16, is applied to obtain an equivalent static load for the SSC that can be represented by a simple model.

The approaches to SSI analyses, described in PSAR section 6.4.1.3.1 address (i) a range of soil and rock properties by considering three site profiles, (ii) effects of ground motion amplification, (iii) kinematic effects of embedded structures including inertial effects, and (iv)

flexibility and impedance of embedded foundations. The applicant stated in PSAR section 7.8 that SSI for RXB and FHB substructures is modeled using SASSI.

USO considered the effects of structure-soil-structure-interactions (SSSI) because the proximity of the structures to each other may influence their foundation input motions. SSSI effects are addressed by modeling both structures in SSI analyses, evaluating interactions between RXB, RAB, and FHB structures.

Damping values are consistent with RG 1.61. Material damping is developed using site-specific geotechnical data and limited to 15 percent of the critical damping.

The staff determined that the methodologies for dynamic analysis, SSI, and SSSI are adequate since they are consistent with ASCE/SEI 4-16 and NUREG-0800 (section 3.7.2). The considerations for dynamic analysis provided in PSAR section 6.4.1.3.1 are reasonable and consistent with traditional approaches for dynamic analysis of SR SSCs. The staff also determines that the damping values for reinforced concrete and foundation soils are acceptable because they are consistent with the guidance in RG 1.61 and NUREG-0800 (section 3.7.1). The information provided gives reasonable assurance that the plant can be constructed and operated safely. During the OL stage, the staff will confirm that the inputs and methodologies used for the analysis and the final design follow the methodology described in the PSAR and are consistent with PDC 2.

According to PSAR section 6.4.1.3.2, USO will develop three-dimensional finite element models for linear seismic analyses of SR structures. The seismic modeling will follow ASCE 4-16 provisions, including effective stiffness of reinforced concrete, mass inertia properties, dynamic mass determination, mass discretization, model discretization, one-step response analysis, two-step analysis for less refined models, and lumped mass stick models.

USO will model dynamic coupling of the primary SR structure with secondary equipment per ASCE/SEI 4-16. USO will use seismic anchor motions and differential movement for design of systems that span multiple levels or transition between seismically isolated and non-seismically isolated portions of the structure. These details will be reviewed at the OL stage.

The staff determined the applicant's methodology for analytical modeling of SR SSCs, as discussed in PSAR section 6.4.1.3.2, is adequate because the approach aligns with ASCE/SEI 4-16, section 3, and standard engineering practices accepted in NUREG-0800 (section 3.7.2). The final design of specific SR SSCs will be reviewed at the OL stage.

PSAR section 6.4.1.3.3 states that modal responses are combined using the square root of the sum of the squares (SRSS) method. For closely spaced and high-frequency modes, the complete quadratic combination (CQC) method will be used. The staff determines this approach acceptable because it is consistent with RG 1.92.

USO will use SSI results that consider the geometry and embedment of structures, to develop natural frequencies and responses for SR structures embedded below the finished grade with multiple foundation levels. The staff determined this approach is reasonable as it aligns with ASCE/SEI 4-16.

PSAR section 6.4.1.3.5 addresses the development of in-structure response spectra (ISRS) at equipment locations for dynamic analyses and equipment support design. USO's approach is consistent with ASCE/SEI 4-16 and includes evaluating spectral damping values, identifying

SSC floor locations in the SSI model, performing seismic analysis of structures for each soil cases, combining responses, ensuring that soil cases are enveloped and gaps are filled, smoothing and broadening is performed to produce the final ISRS results, and generating time histories per ASCE/SEI 4-16 and ASCE/SEI 43-19. The staff determined this method is acceptable as it follows NUREG-0800, section 3.7.2, and RG 1.122. The staff determined that the process that will be used to generate time histories for the ISRS is consistent with ASCE/SEI 4-16 and ASCE/SEI 43-19. Review of the ISRS and its use in equipment and systems design will be performed at the OL stage.

In accordance with ASCE/SEI 4-16 section 3.1, USO will consider accidental torsional effects in the seismic analysis and design of embedded SR structures. USO plans to increase shear forces in shear walls and other lateral load-carrying members by 5 percent to account for these effects. The staff determined that this approach is acceptable because it meets the requirements in ASCE/SEI 4-16 and NUREG-0800, section 3.7.2.

PSAR section 6.4.1.3.7 states that the seismic stability of SR substructures is not evaluated when the center of gravity is below grade level. The staff verified that this is consistent with guidance in ASCE 43-19 section 7.2.1 and determined it is a reasonable approach.

6.4.1.1.4 Modeling and Analysis of Underground SR Structures Directly Supporting NSRST Superstructures

USO discussed the modeling and analysis of underground SR structures directly supporting NSRST superstructures in PSAR section 6.4.1.4. For SR concrete substructures supporting NSRST surface steel-framed structures, dynamic interaction effects are considered through coupled dynamic analysis. The NSRST superstructures are modeled as a lumped mass connected to the detailed finite element model of the substructure. Sensitivity analyses are used to verify that the coupling effects adequately represent the lumped mass. The staff determined this approach is acceptable as it aligns with ASCE/SEI 4-16 section 3.7.

6.4.1.1.5 Seismic Interaction

The methodology to evaluate seismic interactions of NSRST and NST SSCs with SR SSCs is discussed in PSAR section 6.1.3.1 and evaluated in section 6.1.1.3 of this SE. In addition to those considerations, PSAR section 6.4.1.5 states that USO's design process ensures that potential seismic interactions will not adversely impact the ability of SR SSCs to perform their required safety functions. Additionally, interactions of NSRST and NST SSCs with NSRST SSCs will not adversely impact the ability of a seismic risk significant NSRST SSC to perform its safety function. Seismic capabilities of NSRST SSCs are associated with their seismic classification as described in PSAR section 6.4.1.1 and evaluated in section 6.4.1.1.1 of this SE.

The staff determined USO's approach for addressing seismic interactions is acceptable as it provides reasonable assurance at the CP stage that seismic interactions will not impact safety-significant SSCs for design basis loading. This approach should be confirmed through the seismic PRA at the OL stage.

6.4.1.1.6 Seismic Analysis Methods and Procedures Used for Analytical Modeling for NSRST SSCs

PSAR section 6.4.1.6 states that the seismic analysis and design of NSRST SSCs is performed using commercial design codes and standards, based on ASCE/SEI 7-16, including: i) seismic

load effects in ASCE/SEI 7-16 section 12.4.2, ii) strength design and allowable stress approaches per ASCE/SEI 7-16 sections 2.3.6 and 2.4.5, respectively, and iii) seismic loading applied in two orthogonal horizontal directions independently, per ASCE/SEI 7-16 section 12.5. Structural damping of 5% of the critical damping is considered in the analysis.

For seismic analysis based on equivalent lateral force, USO will use ASCE/SEI 7-16 section 12.8. For response spectrum analysis, the number of modes will either have a combined mass participation factor of 100 percent or achieve 90 percent in each orthogonal horizontal direction. The design response spectrum is modified by applying the R/I_e factor (where I_e is the importance factor and R is the response modification factor) and the calculated displacement is modified by the C_d/I_e factor to obtain inelastic displacement, per ASCE/SEI 7-16. The square root of the sum of the squares or quadratic combination methods are used to combine response parameters.

Accidental torsion effects are addressed using ASCE/SEI 7-16 sections 12.8.4.2 and 12.9.2.2.2. When using the equivalent lateral force procedure, the amplification of accidental torsion is considered per section 12.8.4.3 of ASCE/SEI 7-16. Staff confirmed during the audit that the fragility parameters used in the design of NSRST SSCs exceed the minimum values in ASCE/SEI 7-16 and are sufficient to provide reasonable assurance that these SSCs, which are SCN1 or SCN2, will maintain their safety functions at the DBHL.

SSCs that are not safety-significant and do not have seismic interaction requirements are outside the scope of the staff's review for seismic design considerations. For NSRST SSCs requiring special treatment, the staff's review of the seismic analysis methods is described in section 6.4.1.2.2 of this SE.

6.4.1.2 Design of Safety-Significant Structures

PSAR section 6.4.2 states that the RXB, FHB, RAB, and NCB buildings include below-grade reinforced concrete substructures and above-grade steel-framed superstructures. USO classified the RXB, FHB, and NCB substructures as SR and the RAB substructure as NSRST. The steel superstructures of the FHB, NCB, and RAB are classified as NSRST, while the RXB superstructure is classified as NST. The above-ground steel structures of the FHB, RAB, and NCB are supported on the grade slab of their respective concrete substructures. The RXB superstructure foundation is independent and not structurally connected to the RXB substructure. PSAR chapter 1 illustrates the proposed structural configurations.

Preliminary design information for these buildings is discussed in PSAR section 7.8 and reviewed in section 7.8 of this SE. Detailed design information, including building dimensions and details about the structural members and foundations, will be reviewed by the staff at the OL stage.

For clarity, this section is divided into subsections 6.4.1.2.1, "Design of SR Structures," and 6.4.1.2.2, "Design of NSRST Structures," differing from the PSAR's topic-based organization.

For SR structures, the staff reviewed USO's design methodology per 10 CFR Part 50 (50.34(a)(3)(i), 50.34(a)(3)(ii), and Appendix S), including seismic design inputs, analysis methods, modeling methods, applicable codes and standards, loads and load combinations, and materials.

The NSRST classification of SSCs based on the LMP framework is being used for the first time under 10 CFR Part 50. For NSRST structures, the staff reviewed USO's proposed seismic design methodology and analysis. USO aims to establish performance targets for NSRST structures through the SPRA and the LMP framework, modifying the design if needed to meet performance targets and safety criteria. This review at the CP stage provides reasonable assurance that the final design developed with the proposed methodology will meet the intended performance targets when evaluated at the OL stage using the SPRA.

6.4.1.2.1 Design of SR Structures

Methodology for Design and Analysis

USO discussed applicable codes and standards (including specifications) and regulatory guidance in PSAR section 6.4.2.1. USO provided a list of codes and standards for design, fabrication, construction, testing, and inspection of safety-significant structures in PSAR tables 6.4-2, 6.4-3, and 6.4-4. The design codes for SR substructures are ACI 349-13 for reinforced concrete and ANSI/AISC N690-18 for structural steel. USO stated in PSAR section 6.4.2.3 that the concrete substructures are designed using the strength-based limit state design method with load and resistance factors from ACI 349-13 and RG 1.142.

The staff determined that USO's use of the referenced codes and standards is appropriate for the design of SR structures. ACI 349-13 and ANSI/AISC N690-18 are acceptable consensus codes and standards for concrete and structural steel design and construction of nuclear power plants in accordance with the guidance in NUREG-0800 (section 3.8.4), and as endorsed in RG 1.142 and RG 1.243.

PSAR section 6.4.2.2 describes loads and load combinations for structural design, including normal, severe environmental, extreme environmental, and abnormal loads. The load combinations for SR concrete substructures are in PSAR table 6.4-5, and for structural steel in table 6.4-6.

The staff determined that USO's information on loads and load combinations is appropriate for the design of SR structures because the load combinations and the load factors are consistent with ACI 349-13, RG 1.142, ANSI/AISC N690-18, and RG 1.243. The staff will review the application of these loads and load combinations at the OL stage.

USO stated in PSAR section 6.4.2.3.2 that structural analysis and design for SR concrete substructures is performed using GT STRUDL Version 40 to model walls and slabs with shell elements. The dynamic mass includes the dead weight of the structure plus 25% live load, superstructure load (dead weight plus 25 percent live load and 75 percent snow load), and equipment load. The superstructure is modeled separately, with its load represented as a dynamic mass in the substructure model.

The applicant used best-estimate stiffness properties (e.g., effective stiffness) for concrete per ASCE/SEI 4-16 table 3-2. Concrete sections are assumed cracked in flexure for out-of-plane responses and uncracked in axial and shear. Shear walls supported by soil are not considered cracked in flexure.

The staff determined that USO's information on the analysis of SR substructures is adequate for preliminary design because

- GTSTRUDL is an industry-standard
- The dynamic mass is consistent with ASCE/SEI 4-16 section 3.4.2.
- The stiffness properties for cracked and uncracked concrete are consistent with ASCE/SEI 4-16 (section 3.3.2).
- The mesh sizes for the finite element models are adequate to transmit the entire frequency range of interest.
- The superstructure reactions on the substructure are adequately considered.

The staff will review the final design at the OL stage.

Materials

In PSAR section 6.4.2.4, USO discussed material standards for concrete and steel used in SR and NSRST structures. Concrete material properties are in PSAR table 6.4-10, with a compressive strength of 5000 psi for safety-significant structures. Concrete works follow ACI 301-16. Reinforcing bars are Grade 60, per ASTM A615-09b and ASTM 706-09b, and headed reinforcing bars per ASTM A970-07. Structural steel conforms to ANSI/AISC N690-18 for SR structures, and the stainless-steel liner plate for the FHB spent fuel pool follows ASTM A240-16a.

The staff determined that the applicant's information on the concrete and steel material properties is acceptable because it conforms to applicable design and ASTM codes and standards. The staff will verify materials used in the final design at the OL stage.

Summary of the staff's determinations

The staff reviewed USO's methodology for SR substructure design and determined that the information provided gives reasonable assurance that the plant can be constructed and operated safely and that the final design will meet PDC 2, 80, 81, and 10 CFR Part 50, Appendix S requirements for withstanding earthquakes without impairing structural integrity and performance. This conclusion is based on:

1. USO provided sufficient information on applicable codes, standards, and specifications; loads and loading combinations; design and analysis procedures; structural acceptance criteria; and materials, quality control, and special construction techniques that will be used in the final design.
2. USO will design the structure to withstand the most severe earthquake established for the site and includes adequate consideration of the combinations of the effects of normal and accident conditions with environmental loadings such as earthquakes and other natural phenomena.
3. USO's seismic analysis method is based on appropriate procedures for structural modeling, SSI, development of floor response spectra, inclusion of torsional effects, consideration of foundation stability, and adequate description of geotechnical and structural material properties.

4. USO's structural analysis and design conform with established criteria, codes, standards, and specifications acceptable to the staff, including RGs 1.61, 1.122, 1.92, 1.142, 1.243, and 1.199, and industry standards ASCE 4-16, ACI 349-13, and ANSI/AISC N690-18.
5. USO's information is consistent with DANU-ISG-2022-01 section 1.1.2 which references NUREG-0800.

The staff determined the PSAR information meets 10 CFR 50.34(a) and 50.35 requirements for issuing a CP. The final design will be confirmed during the OL stage.

6.4.1.2.2 Design of NSRST Structures

Methodology for Design and Analysis

USO will use the provisions in ASCE/SEI 7-16, chapter 12 for the design of the FHB and NCB superstructures, and chapter 15 for the RAB superstructure, which is categorized under non-building structures that are similar to buildings. As discussed in section 6.4.1.1.2 in this SE, the MCER is derived from the SSE, with seismic classifications of NSRST SSCs as evaluated in 6.4.1.1.1 of this SE. The seismic design parameters for NSRST structures classified as SCN1 or SCN2 are provided in PSAR table 6.4-1B. The primary design codes and standards for concrete and steel structures for NSRST substructures and superstructures are discussed in PSAR sections 6.4.2.1. For NSRST substructures and superstructures, the applicant plans to use ACI 318-19 for concrete and ANSI/AISC 360-16 for structural steel. Concrete structures will use the ultimate strength design method, and structural steel frames will use the LRFD design method.

PSAR section 6.4.2.2.5 discusses loads and loading combinations for NSRST structures. USO will use loads and load combinations from IBC-2021 for NSRST structures or those in ACI 318-19 and ANSI/AISC 360-16 for more stringent loading. For extraordinary events, load combinations and factors from ACI 349-13 for concrete and ANSI/AISC N690 for steel will be used, as detailed in PSAR tables 6.4-7, 6.4-8, 6.4-11, and 6.4-12.

Design of NSRST Superstructures

PSAR section 6.4.2.3.3 describes the design of NSRST superstructures. Reinforced concrete structural components included in NSRST superstructures will be designed using ACI 318-19, and structural steel frames using AISC 360-16 LRFD methodology, complying with AISC 325-17, AISC 341-16, ASCE/SEI-7, and IBC 2021. USO will design bolt connections following ANSI/AISC 360-16 and AISC 348-14, and weld connections following ANSI/AISC 360-16. USO will design concrete anchors and steel embedment following ACI 318-08. Per PSAR section 6.4.2.3.3, NSRST superstructures will be designed to withstand tornado missile impacts, but the building envelope will only be designed for normal wind loading and will not be relied on to protect the building interiors from tornado wind and missile impacts.

Analysis of NSRST Superstructures

PSAR section 6.4.2.3.4 states that structural modeling of NSRST steel superstructures will use GT STRUDL. The lateral force resisting system will be represented in a three-dimensional finite element model, with loads, load combinations, and seismic analysis methodology in accordance

with ASCE/SEI 7-16. The model will provide structural member design, deflection checks, support reactions, dynamic responses, and structural steel quantities.

Summary

Although NSRST structures are safety significant, the NRC has not established detailed regulatory positions for them as it has for SR structures in NUREG-0800. Consequently, applicants should propose their own approach to designing safety-significant NSRST structures, including the design codes and standards that could be different from those approved by the staff for SR structures. USO's proposed approach uses ASCE/SEI 7-16 and associated codes and standards, which are well-established for commercial construction. However, these codes and standards have not been used in a license application for the design of safety-significant structures, nor has the NRC independently reviewed ASCE/SEI 7-16 for generic application to nuclear power plant licensing.

The staff's review in this SE is not evaluating or endorsing ASCE/SEI 7 for general use. Instead, the staff focused on USO's design approach, NSRST SSC classification, and the basis for the codes and standards used for NSRST structure design. USO's classification of NSRST structures at the CP stage will be confirmed or adjusted at the OL stage based on a risk-informed process and seismic PRA. The staff determined the overall design approach is acceptable because its initial classification of NSRST SSCs at the CP stage aligns with RG 1.253 (which endorses NEI 21-07) and RG 1.233 (which endorses NEI 18-04). USO's commitment to confirm or adjust classifications and associated design criteria at the OL stage based on seismic PRA provides reasonable confidence that the performance target will be achieved.

6.4.1.3 Mechanical Systems and Components

PDC 1, 2, and 4 require, in part, that SSCs important to safety be constructed and tested to quality standards commensurate with the importance of the safety functions to be performed and designed with appropriate margins to withstand the effects of anticipated normal occurrences, natural phenomena such as earthquakes, and postulated accidents.

PSAR section 6.4.3 describes the design and analysis of ASME Boiler and Pressure Vessel Code (BPVC) Section III, Division 5 (III-5) mechanical components, supports, and core support structures. PSAR section 6.4.3.1.1 describes the plant transients used in fatigue analyses. The service level loadings considered in the identification of plant transients include:

- Level A Service Loadings – Loading arising from system startup, hot standby operation and system shutdown
- Level B Service Loadings – Loadings that are anticipated to occur often enough that the design should include a capability to withstand the loadings without operational impairment
- Level C Service Loadings – Loadings require a shutdown for correction or repair of damage in the system

- Level D Service Loadings – Loadings associated with postulated events of extremely low probability whose consequences are such that the integrity and operability of the nuclear system may be impaired
- Testing Loadings – Loadings that occur during hydrostatic tests, pneumatic tests, and leak tests

The transients used in the design of ASME BPVC Section III components are listed in PSAR table 6.4-16. The design transients cover all normal, off-normal, and test operations that are expected to occur during the design life of the plant. The events account for filling and draining of the sodium, heat up and cool down of the systems, startup and shutdown operations, power maneuvering operations, and plant responses to upset, emergency, and faulted initiating events. During the audit, the staff confirmed the design transients and the preliminary number of occurrences for each design transient. Based on the staff's review, the staff determines the identified plant transients and the number of transients is appropriate for the design life.

PSAR section 6.4.3.1.2 identifies the computer programs used in the static and dynamic analysis of mechanical equipment. The staff has familiarity with these codes and their capabilities and determines the identified computer programs appropriate for use in the design and analysis of mechanical equipment. The staff will review the application of these codes for detailed analyses of mechanical equipment for the final design at the OL stage.

PSAR section 6.4.3.2 describes the seismic analysis of mechanical systems and components. The seismic analysis of mechanical systems and components follows the methodology described in PSAR section 6.4.1. The staff evaluation of the seismic design and analysis methods for safety-significant SSCs is documented in section 6.4.1.1 of this SE. PSAR section 6.4.1.2 also describes the methods for analysis of the seismically isolated Reactor Enclosure System (RES). As discussed in section 7.1.2.3, the RES is supported inside the RXB substructure with a seismic isolation system (SIS). The design and qualification methodology for the SIS will be in accordance with TerraPower's SIS TR [NAT-8922, Rev. 2, (ML25195A156)]. The staff reviewed the methodology for the seismic analysis of the RES and found that the approach includes: (i) selection of frequency; (ii) determination of number of earthquake cycles; (iii) consideration of damping for modal response; and (iv) consideration of static vertical load factors acting simultaneously with two horizontal dynamic loadings. The staff determined that the approach is acceptable because it is consistent with guidance on seismic analysis of mechanical components and systems in NUREG-0800, sections 3.7.2 and 3.7.3. The staff will review the applications of these analyses to mechanical equipment in the final design at the OL stage.

In the SE for NAT-8922 (ML25296A226), the staff determined that the methodology provides an acceptable approach for future applicants using the Sodium design to establish the design criteria and qualification requirements for the 3D isolation system described, subject to limitations and conditions (L&Cs). The staff evaluated these L&Cs to confirm they were evaluated as part of the PSAR and either met or could be reasonably left to the OL stage when the design is finalized. The staff's SE for NAT-8922 imposed the following L&Cs:

1. The conclusions reached in this SE only address the content provided in section 7 of the TR. Thus, any licensee or applicant referencing this TR must evaluate the other aspects of the information described in the remaining six sections of the TR for any site-specific application.

2. An applicant or licensee referencing this TR must use the Natrium design, as summarized in sections 5.1 and 6 of the TR, or justify that any departures from these design features do not affect the conclusions of the TR and this SE.
3. The methodology described in the TR and the conclusions reached in this SE are based on a component 3D isolation system using ISUs and IDUs which limit displacement and are arranged to ensure an even distribution of loads within the SIS and limits the seismic demands exerted on the reactor. The details of the methodology in TR section 7 that were reviewed by the NRC staff are limited to this specific component 3D isolation technology.
4. If an applicant or licensee referencing this TR chooses to follow a generic qualification process as described in TR section 6.1, they must perform seismic analyses to confirm that the site-specific motion (based on the site and design of RXB, SIS, and supported subsystems) is enveloped by the generic ground motion. For conditions in which site-specific ground motion spectra are not enveloped by the bounding generic ground motion spectra, a new bounding spectra must be generated and the qualification process must be repeated.
5. The conclusions reached in this SE are limited to a design that does not include a dynamic stop. If impact between the IDU piston and housing under extreme earthquake loading is deemed possible, then a sensitivity analysis should be conducted to bound the impact loads and response of critical SIS supported SSCs to determine the potential risks.
6. The conclusions reached in this SE are based on TerraPower's methodology for verification and validation of numerical models capable of predicting results of dynamic testing of the prototype isolators consisting of linear springs and viscoelastic dampers. To use this methodology at a specific site, an applicant or licensee shall confirm that the range of applicability for the numerical models encompass the site-specific conditions.
7. The conclusions reached in this SE are based on TerraPower's multi-step or single-step methodology for plant seismic response analysis. To use this methodology at a specific site, an applicant or licensee shall develop a validation plan for the single-step or multi-step response analysis (including transitions between analysis codes), as applicable, and a verification plan for the codes used in the response analysis.
8. The conclusions reached in this SE are based on the specific codes and standards used to implement the methodology, listed in TR section 7. Applicants or licensees referencing this TR should justify any deviations to these codes and standards. An application using the TR methodology at any site requires a peer review as described in TR table 7-1.
9. Applicants or licensees referencing this TR must provide a basis for the adequacy of seismic monitoring equipment for the site-specific application that addresses the unique considerations for seismically isolated systems and the recommendations provided in NUREG/CR-7253, including justification for the location of instrumentation relative to the location of the seismic isolators.

The staff reviewed and determined that the PSAR's use of NAT-8922 meets TR L&C 1 and 2 because the preliminary Natrium design described in the PSAR is consistent with the design

information provided in the TR, including the descriptions in sections 5.1 and 6 of the TR. PSAR section 7.1.2.3 provides a description of the SIS, which uses isolation technology that is consistent with what is described in the TR, namely a 3D isolation system using isolation spring units (ISUs) and isolation damper units (IDUs). This is sufficient to adequately address L&C 3 at the preliminary design stage. While the PSAR does not explicitly state that the generic bounding ground motion spectra envelops the site-specific spectra, PSAR section 7.1.2.3 states that testing will be done to the full range of loading conditions experienced during a seismic event or per ASME QME-1, scaled units qualified as parent restraints will be used with design similarity and analysis to qualify the full scale units. This information is sufficient to address L&C 4 at the preliminary design stage. At the OL stage, USO should verify explicitly that the generic bounding ground motion spectra envelops the site-specific spectra. USO adequately addressed L&C 5 by confirming, in PSAR section 7.1.2.3, that a dynamic stop is not used. L&Cs 6 and 7 are related to the verification and validation (V&V) of numerical models and analysis codes. The information provided in the TR on the approach to V&V is sufficient for the CP stage. It is reasonable to leave the review of final V&V methods and execution, addressing L&C 6 and 7, to the OL stage. USO did not identify any deviations from the codes and standards used in the methodology described in the TR. If deviations are identified as part of further design, justifications for those deviations will need to be provided at the OL stage to meet L&C 8.

A basis for the adequacy of seismic monitoring equipment that addresses the unique considerations for seismically isolated systems was not provided in the PSAR to address L&C 9. This justification along with additional details on the seismic monitoring system are expected at the OL stage. It is reasonable to leave this to the OL stage because the justification is likely to rely partly on the level of uncertainty present in the numerical models and analysis codes, which are determined through the V&V to be performed between now and the OL stage.

As discussed in the SE for the SIS TR, the methodologies provided in the TR contribute to meeting the requirements of PDC 1, 2, and 80 for the SIS.

PSAR section 6.4.3.3.1 provides the load combinations and service stress limits for mechanical components and associated supports. The loads considered for the design of ASME Code Class A and B mechanical components are summarized in PSAR table 6.4-17 and the loading combinations and the applicable stress limits are provided in PSAR table 6.4-18. The loading combinations and the corresponding stress limits for ASME Code design are defined for the design condition, service levels A, B, C, and D (i.e. normal, upset, emergency, and faulted conditions, respectively), and the test conditions. The staff reviewed the proposed loads, load combinations, and stress limits and concludes that appropriate combinations of operating design transients and accident loadings have been specified to provide a conservative design envelope for the design of mechanical systems and components. The staff will review the application of these loads and load combinations at the OL stage.

PSAR section 6.4.3.3.2 describes the design and installation of pressure relief devices. The applicant states that the reaction loading due to discharge of a relief device is analyzed using the guidance of ASME BPVC Section III, Appendix O. The staff determines this approach acceptable because it is consistent with the guidance in NUREG-0800, section 3.9.3.

6.4.1.4 ASME BPVC, Section III Piping Systems, Piping Components, and Associated Supports

PSAR section 6.4.4 addresses the design and analysis of piping systems and piping supports. The applicant clarified during the audit that this section is only applicable to piping systems and piping supports that select ASME BPVC Section III as their design code.

PSAR section 6.4.4.2 refers to ASME BPVC Section III, Division 1 (III-1) for the design of piping and support in low temperature applications and ASME BPVC III-5 for the design of piping systems in elevated temperature applications. RG 1.87, Rev. 2 endorses the 2017 Edition of ASME BPVC III-5 as acceptable to use for the design and construction of mechanical components that operate in elevated temperature environments. ASME BPVC III-5 refers to III-1 for low temperature applications. Therefore, the staff determines the use of ASME BPVC III-1 and III-5 acceptable.

6.4.1.5 Application of Industrial Codes

PSAR section 6.4.5 addresses the application of ASME BPVC Section VIII, ASME B31.1, and ASME B31.3 industrial codes for the design and analysis of safety-significant SSCs. These industrial codes are generally selected for the design and construction of NSRST mechanical systems and components as well as selected SR SSCs. The applicant stated that supplemental requirements, in addition to the industrial code requirements, are applied as special treatments as necessary to provide enhanced SSC quality for safety-significant SSCs. The application of the supplemental requirements for major NSRST SSCs is addressed in Appendix 14.2 of NAT-13478 (ML25274A130) and the staff evaluation of these special treatments for major NSRST SSCs can be found in chapter 7 of the SE. The staff will review the special treatments for other NSRST SSCs at the final design stage in the OL application.

The applicant noted that ASME BPVC Section VIII, Division 1 (VIII-1) and Division 2 (VIII-2) provide rules for the design and construction of pressure vessels. ASME BPVC VIII-1 provides a design-by-rule approach that is utilized for noncomplex vessel designs. ASME BPVC VIII-2 allows for a design-by-analysis approach which provides more flexibility through detailed analysis.

For safety-significant SSCs designed to ASME BPVC VIII-1, the combination of loads follows the methodology outlined in section UG-23. For safety significant SSCs designed to ASME BPVC VIII-2, table 4.1.2 of the code provides the minimum required load combinations. PSAR section 6.4.5.1 states that the design loads from the ASME BPVC Section III transients are considered as loading input for ASME BPVC Section VIII component design to ensure the SSC will perform its safety-significant function during each transient event. The staff notes that ASME BPVC Section VIII does not specify different allowable stresses for different service level loads but does permit an increase in allowable stress when earthquake or wind loading is considered in combination with other loads.

PSAR section 6.4.5.1 also indicates that the rules in ASME BPVC Section VIII will be supplemented with additional analysis to account for fatigue and creep effects. Since ASME BPVC VIII-1 does not address fatigue failure explicitly, the fatigue evaluation procedure from ASME BPVC VIII-2 will be applied. Furthermore, because Section VIII does not explicitly account for creep effects in long-term service, components subject to non-negligible creep, as determined by screening per ASME BPVC III-5, Mandatory Appendix HCB-III, will be evaluated using the Class A rules in ASME BPVC III-5 or Code Case 2843-3. Code Case 2843-3 provides elevated temperature rules that closely mirror ASME BPVC III-5 Class A rules, including fatigue and creep-fatigue assessment. The staff reviewed the design approach for pressure vessels constructed to ASME BPVC Section VIII with the additional supplemental requirements described above and found the approach acceptable because the design rules in ASME BPVC Section VIII and the supplemental requirements provide assurance that the ASME BPVC

Section III design transients and the effects of creep and fatigue are considered in the design and analysis of these SSCs.

PSAR section 6.4.5.2 discusses the design and analysis of piping systems in accordance with ASME B31.1, "Power Piping," and B31.3, "Process Piping." ASME B31.1 and B31.3 provide rules for the design and construction of industrial power piping systems and process piping systems respectively. These piping codes are generally selected for the design and analysis of NSRST piping, piping components, and associated supports as well as selected SR SSCs.

ASME B31.1 and B31.3 require defining the loads that the component will experience in service, but do not relate these loads to service levels, or specify off-normal conditions. The PSAR states that the ASME BPVC Section III design transients are considered as loading input for the design of ASME B31.1 and B31.3 piping systems and components to ensure the SSCs will perform their safety-significant functions. The staff notes that ASME B31.1 and B31.3 do not specify different allowable stresses for different service level loads but do permit an increase in allowable stress when occasional loads are considered in combination with other loads.

ASME B31.1 and B31.3 piping codes are supplemented with rules from ASME BPVC III-5 for SSCs in elevated temperature service in the creep regime on a case-by-case basis. For piping determined to have non-negligible creep effects by ASME BPVC III-5 Appendix HCB-III, an analysis per ASME BPVC III-5 Subsection HCB-3634 will be applied. Section HCB-3634 provides more conservative allowable stresses for thermal expansion stresses and stress criteria for thermal cycling to account for creep effects. The staff reviewed the design approach for piping constructed to ASME B31.1 or B31.3 with the supplemental requirements described above and found the approach acceptable because the design rules in ASME B31.1 or B31.3 and the supplement requirements provide assurance that the ASME BPVC Section III design transients and the effects of creep are considered in the design and analysis of these SSCs.

6.4.2 Conclusion

Based on its review and findings documented in the preceding subsections, the staff determined the preliminary information regarding the seismic, structural, and mechanical design of safety-significant SSCs is consistent with relevant guidance and acceptable for the CP review. Further information related to the design of safety-significant SSCs can be left for consideration at the OL stage.

6.5 References

American Concrete Institute Standard 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," ACI, 2013 edition.

----. ACI Standard 318, "Building Code Requirements for Structural Concrete," ACI, 2019 edition.

American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) Standard N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities," ANSI/AISC, 2018 edition.

----. ANSI/AISC Standard 360, "Specification for Structural Steel Buildings," ANSI/AISC, 2016 edition.

American Society of Civil Engineers (ASCE) Standard 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," ASCE, 2016 edition.

----. ASCE Standard 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," ASCE, 2020 edition.

----. ASCE/SEI 4-16, "Seismic Analysis of Safety-Related Steel Structures for Nuclear Facilities," ASCE, 2016 Edition.

ASME, BPVC, Section III, Division 1 "Rules for Construction of Nuclear Facility Components," ASME, 2017 Edition.

American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Facility Components, Division 5, High Temperature Reactors," ASME: New York, NY, 2017 edition.

----. ASME, "BPVC, Section VIII: Rules for Construction of Pressure Vessels" ASME, 2021 edition.

----. ASME Standard QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," ASME, 2023 edition.

----. ASME Standard B31.1, "Power Piping," ASME, 2022 edition.

----. ASME Standard B31.3, "Process Piping," ASME, 2020 edition.

International Code Council (ICC), *IBC 2021: International Building Code*. ICC Publications, 2020.

7 DESCRIPTIONS FOR SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS, AND COMPONENTS

This chapter of the safety evaluation (SE) describes the staff's review and evaluation of the Kemmerer Power Station Unit 1 (KU1) preliminary safety analysis report (PSAR) chapter 7, which contains preliminary descriptions of KU1 safety-significant structures, systems, and components (SSCs).

The applicable regulatory requirements for the evaluation of the descriptions of safety-significant SSCs include:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of applications; technical information," paragraph (a), "Preliminary safety analysis report," including:
 - 10 CFR 50.34(a)(1)(ii);
 - 10 CFR 50.34(a)(3)(ii);
 - 10 CFR 50.34(a)(3)(iii); and
 - 10 CFR 50.34(a)(4);
- 10 CFR 50.35, "Issuance of construction permits;" and
- 10 CFR 50.40, "Common standards."

General guidance on content of applications for non-light water reactors related to safety-significant SSCs is provided by Nuclear Energy Institute (NEI) 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors," Revision (Rev.) 1 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22060A190), as endorsed by Regulatory Guide (RG) 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Rev. 0 (ML23269A222). PSAR chapter 7 differs slightly from this content of application guidance by combining the descriptions of safety-related (SR) SSCs and non-safety-related with special treatment (NSRST) SSCs into a single chapter. As this difference changes the organization of the submitted information rather than the substantive content, it is administrative in nature and therefore does not affect reliance on the guidance.

To document the staff's review of PSAR chapter 7, this chapter of the SE is organized into sections and subsections corresponding to each of the major systems identified in the PSAR. Each major system subsection (e.g., 7.1.1) contains the following elements:

- an introduction describing the SSC, any SSC-specific regulatory requirements and guidance, and the applicable principal design criteria (PDC), as described in PSAR section 5.3;²²

²² The applicant relied on an approved topical report (TR) for its PDC. The staff reviewed the applicant's reliance on this TR in SE section 5.3.

- an assessment of the preliminary SSC design against the PDC and the safety functions the SSC supports, as described in PSAR sections 5.3 and 5.2, respectively, and evaluated in the licensing basis event (LBE) analyses in PSAR chapter 3; and
- an evaluation of the design codes and standards and any special treatments necessary to support the SSC's consistency with the PDC and performance of the safety functions.

For each SSC, the staff reviewed the information provided in the PSAR and verified the SSC met the requirements in the relevant portions of 10 CFR 50.34(a), because the PSAR provides the following information:

- An overall description of the SSC, its location, and its modes of operation;
- A description of how the SSC supports the required safety functions (RSFs) for SR SSCs, or other safety-significant probabilistic risk assessment (PRA) safety functions (PSFs) for NSRST SSCs;
- A discussion of natural phenomena that could impact the SSC;
- Instrumentation needed to ensure SSC performance;
- Safety classification;
- Applicable codes and standards; and
- Preliminary materials of construction.

These items are consistent with the guidance for the content of construction permit (CP) applications found in sections C.6.4 and C.7.3 of NEI 21-07. As stated in RG 1.253, these sections of NEI 21-07 provide an acceptable method to develop information for SSC design requirements and capabilities, design basis hazard levels (DBHLs), special treatments, and system descriptions.

Technically relevant Three Mile Island (TMI) action items (see SE chapter 1, table 1.8-3) are evaluated in chapter 7 SE sections that pertain to the SSCs that address the TMI item. The applicability of TMI-related items to the KU1 design is addressed in SE section 1.3.2.

The staff also reviewed the safety-related design criteria (SRDC) provided in PSAR section 5.2 and discussed in PSAR section 6.1.2 and SE section 6.1.1.2 to ensure the safety functions associated with each SSC are consistent with the SRDC. Based on the staff evaluation described below, the staff determined that the SRDC are consistent with their associated safety functions and the preliminary design of the SSCs. The SRDC will be reviewed again at the operating license (OL) stage to ensure they are consistent with the final design and SR functions relied upon for the final safety analysis.

7.1 Nuclear Heat Supply Systems

PSAR section 7.1 describes the following systems that are involved in the generation of nuclear energy and transfer of heat from the reactor core to the energy island:

- Reactor core system (RCC)
- Reactor enclosure system (RES)
- Primary heat transport system (PHT)
- Intermediate heat transport system (IHT)

7.1.1 Reactor Core System

PSAR section 7.1.1 describes the RCC, which consists of all the replaceable core assemblies, including the fuel, reflector, shield, and control assemblies, that make up the bulk of the core. Fuel assemblies include both normal assemblies and lead demonstration and lead test assemblies (LDAs and LTAs, respectively). These assemblies, their acceptance criteria, and qualification data and/or qualification plans are detailed in the Nuclear Regulatory Commission (NRC)-approved TR NAT-2806-A, “Fuel and Control Assembly Qualification” (ML25083A296), which is incorporated by reference into the KU1 PSAR. Fuel design is also discussed in PSAR section 3.11 and in SE section 3.10.

The RCC also includes material surveillance assemblies, neutron source assemblies, and compressive assemblies. Given the core restraint system design, the compressive assemblies are intended to ensure the core behaves appropriately at low temperatures and are the focus of research and development (R&D) efforts described in PSAR chapter 13. Simulated core assemblies are used to support filling and testing the core and may assist in recovering a stuck assembly if needed. Neutron source assemblies are used in the initial reactor startup. Material surveillance assemblies hold material specimens for data collection.

Fuel and control assemblies are classified as SR in their entirety. Structural elements of other RCC components are classified as SR, while other elements are classified as non-safety-related with no special treatment (NST).

Applicable regulations and guidance are provided at the beginning of this chapter. PSAR sections 7.1.1 and 5.3 identify the following PDC as applicable to the RCC: 1, 2, 4, 10, 11, 12, 16, 26, 28, 29, and 80.

7.1.1.1 Technical Evaluation

7.1.1.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. No codes or standards were identified in PSAR section 7.1.1 related to the design of RCC. Consensus codes and standards used in the design of fuel and control assemblies are discussed in NAT-2806-A and were determined to be acceptable by the staff. The staff determined that the preliminary RCC design is consistent with PDC 1 because the applicable codes and standards discussed in NAT-2806-A were used.

7.1.1.1.1 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.1.1.2 states that the reactor core is located inside the RES, which is located in the reactor building (RXB) substructure. The RES and RXB provide protection against most external hazards. This section further states that the seismic response of the core is analyzed, including the effects on reactivity due to core assembly displacements. These effects are specifically represented by the seismic design basis accident (DBA) described in PSAR section 3.9.3. The PSAR states that the RCC is designed to maintain the capability to perform safety-significant functions under an appropriate combination of normal and accident conditions. This is appropriately reflected by the design criteria provided in NAT-2806-A for fuel, control, reflector, and shield assemblies.

With respect to seismic design, PSAR section 7.1.1.2 states that SR elements of RCC are classified as safety-related seismic category 1 (SCS1) and that RCC components are qualified to withstand seismic loads associated with the safe shutdown earthquake (SSE) without losing the capability to perform safety-significant functions, in accordance with the methods described in PSAR section 6.4.1. As discussed in PSAR section 7.1.1.1, structural elements of RCC assemblies are classified as SR. PSAR section 7.1.1.3 clarifies that these elements include the load pads, duct tubes, handling sockets, and inlet nozzles. Because these structural components encapsulate any NST components of the core assembly, the staff determined that seismic qualification of the SR components will be sufficient to ensure the NST components do not interfere with the ability of safety-significant SSCs to perform their safety-significant functions following the SSE.

The staff determined that the preliminary design of RCC is consistent with PDC 2 because its location within the RXB substructure and RES provides protection from most natural phenomena, its safety-significant components are designed to withstand postulated seismic events, and the RCC is evaluated to ensure it can perform its safety functions under the effects of natural phenomena not mitigated by the RXB and RES.

7.1.1.1.2 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences (AOOs), and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. Compatibility of reactor core components with the hot sodium operating environment is addressed in NAT-2806-A, where the staff found it to be acceptable. PSAR section 7.1.1.2 states that vibratory loading of the core is assessed, consistent with the discussion in NAT-2806-A.

PSAR section 7.1.1.3 states that a portion of the core assemblies will include hard facing (also referred to as coatings) to protect against wear, galling, self-welding, and excessive inter-assembly friction. The preliminary design includes hard facings on duct load pads, assembly inlet nozzles, and seal rings between the assembly duct and inlet nozzle. These hard facings are the subject of ongoing R&D as described in PSAR chapter 13 and section 11 of NAT-13478, "Natrium Demonstration Plant Long-Lived Passive Structural Materials of Construction Selection and Development" (ML25274A130). The applicant is also considering applying coatings to other core components in the Natrium design, as described in table 11.2 of NAT-13478. The staff evaluation of the hard facing and coating R&D activities is provided in SE chapter 13.

The staff determined that the preliminary design of the RCC is consistent with PDC 4 because it is evaluated to ensure compatibility with the environmental conditions in the PDC, as discussed in NAT-2806-A, and the planned and potential hard facing applications in the RCC are the subject of a reasonable R&D plan as evaluated in SE chapter 13.

7.1.1.1.3 Reactor Design, Reactor Inherent Protection, and Suppression of Reactor Power Oscillations (PDC 10, 11, and 12)

PDC 10 requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded during normal operation or AOOs. The staff confirmed that this is demonstrated based on the AOO analyses provided in PSAR section 3.6, where no AOOs result in fuel limits being exceeded; for these LBEs, the fuel limits are used as a surrogate for the SARRDLs because maintaining cladding integrity ensures there will not be an increase in circulating radionuclide inventory.

PDC 11 requires the inherent reactivity feedback mechanisms of the core and associated systems to compensate for rapid increases in reactivity. Reactivity feedback mechanisms associated with the design of RCC include Doppler, non-fuel material temperature coefficients, coolant temperature coefficients, fuel axial expansion, and fuel bowing. The staff confirmed that the nuclear and thermal hydraulic design analyses, described in PSAR sections 3.12 and 3.13 and provided in a supplemental technical report, TP-LIC-RPT-0011, "Core Nuclear and Thermal Hydraulic Design Technical Report" (ML25276A289), demonstrate that the net result of the feedback mechanisms is to compensate for a rapid increase in reactivity under normal steady-state conditions. The staff also confirmed that the LBE analyses described in PSAR sections 3.6 through 3.9 demonstrate the same behavior during transients. The staff notes that the core design is not final, as discussed in TP-LIC-RPT-0011, and future changes may affect power distribution and specific reactivity feedback values. The staff will review this information at the OL stage.

PDC 12 requires power oscillations that can result in conditions exceeding SARRDLs to not be possible or to be reliably and readily detected and suppressed. PSAR section 3.12.1.6 incorporates by reference applicable sections of NAT-9393-A, "Stability Methodology Topical Report," Rev. 0 (ML25211A276), which provides a description of the method used in evaluating and demonstrating compliance with PDC 12 for the Sodium design. As discussed in SE section 3.11.2.5, the staff determined that the application of the approved TR to KU1 to be acceptable and determined the preliminary analysis using design parameters demonstrates that there are no instabilities over the evaluated power range.

The staff determined the RCC design is consistent with PDC 10, 11, and 12 based on the analyses provided in chapter 3 of the PSAR.

7.1.1.1.4 Containment Design (PDC 16)

PDC 16 requires a functional containment consisting of multiple barriers to be provided. PSAR section 7.1.1 states that the fuel cladding serves as a functional containment barrier under normal conditions and for certain LBEs in which cladding failure does not occur. This is consistent with PSAR section 1.3.2.1 which discusses the functional containment barriers. The PSAR also states that, although the metallic fuel matrix is expected to retain a portion of the radionuclides, it is not credited as a functional containment barrier. The staff determined this approach is acceptable, and that the preliminary RCC design is consistent with PDC 16,

because fuel cladding integrity is evaluated as part of the LBEs discussed in PSAR chapter 3, and because radionuclide release from the fuel (both release from the plenum when cladding is failed and any retention in the metallic fuel matrix) is considered in the NRC approved mechanistic source term methodology TR, NAT-9392-A, "Radiological Source Term Methodology," Rev. 0 (ML25211A271), as discussed in SE sections 3.2 and 3.10.

7.1.1.1.5 Reactivity Control Systems (PDC 26)

PDC 26 requires a minimum of two reactivity control systems or means to provide the capability to insert negative reactivity and maintain the reactor shutdown under various conditions. This PDC is primarily addressed in SE section 7.2.5. However, the RCC plays a key role in fulfilling this PDC through the design of the primary and secondary control rod assemblies (CRAs), as well as through the performance of the fuel assemblies, ducts, and related components to ensure that CRA insertion is not impeded. The primary and secondary CRA designs and their roles are described at a high level in PSAR section 7.1.1.3.1 as well as in NAT-2806-A and TP-LIC-RPT-0011. Additionally, NAT-2806-A identifies detailed design criteria for key core components that support maintaining the ability to insert control rods into the reactor. The staff determined that the RCC design is consistent with PDC 26 because the design of the control assemblies and the control assembly ducts ensures that negative reactivity can be reliably inserted under a variety of conditions.

7.1.1.1.6 Reactivity Limits and Protection Against AOOs (PDC 28 and 29)

PDC 28 requires limits on the amount and rate of reactivity increase to ensure the primary coolant boundary and core are not damaged. This is controlled by the control rod drive system (CRD) design, as discussed in SE section 7.2.5, as well as the fuel and control assembly design. The fuel and control assembly acceptance criteria in NAT-2806-A include consideration of significant damage to the core that could impair the ability to cool the reactor. The staff determined that the RCC design is consistent with PDC 28 because the applicant has established appropriate limits on reactivity insertion through the design of the CRAs and CRD. The adequacy of the proposed design is evaluated through the safety analysis in PSAR chapter 3. The staff notes that the core design is not yet finalized, as discussed in TP-LIC-RPT-0011, and there may be changes, for example to rod insertion limits, that may substantially affect the amount of reactivity that can be inserted with the CRAs. The staff will review this information at the OL stage.

Similarly, PDC 29 requires the reactivity control system to be designed to assure an extremely high probability of accomplishing its safety function during an AOO. This is supported in part by the design of the fuel and control assemblies. The staff considers the preliminary RCC design consistent with PDC 29 because NAT-2806-A provides acceptance criteria for the fuel and control assemblies that prevent them from being damaged during normal operation and AOOs, which ensure they can fulfill their safety function.

7.1.1.1.7 Reactor Vessel and Reactor System Structural Design Basis (PDC 80)

PDC 80 requires, in part, that the reactor system is designed to maintain its integrity during postulated accidents. This ensures the geometry necessary for passive removal of residual heat from the reactor core is preserved and that sufficient insertion of neutron absorbers remains possible to provide for reactor shutdown. Fuel and control assembly design plays a key role in ensuring long-term core cooling and shutdown capability for the reactor during postulated accidents. The acceptance criteria provided in NAT-2806-A ensure that passive heat removal is

maintained and that control rod insertion is not impeded during a postulated accident. As such, the staff determined that the RCC design is consistent with PDC 80 because there are appropriate fuel design criteria to ensure residual heat removal and negative reactivity insertion capability.

7.1.1.1.8 Safety Functions

7.1.1.1.8.1 Reactivity Control Functions

PSAR section 7.1.1.2 states that the RCC supports two reactivity control functions: DL3-RC1, “Scram – Gravity Driven Absorber Insertion by Latch Release,” and DL4-RC3 “Control Rod Drive System Driveline Scram Follow,” which are SR and NSRST, respectively, by maintaining the relative movement or deflection of RCC components with respect to CRD components within limits required to allow control rod release and insertion into the core during plant operations and LBEs. PSAR table 5.2-1 further states that RCC supports DL3-RC1 by ensuring control assemblies insert into the core to a depth and within a time limit that maintains dose under the 10 CFR 50.34 dose limits at a safe shutdown condition. The staff determined that the preliminary design of the RCC is consistent with DL3-RC1 and DL4-RC3 functions because the acceptance criteria presented in NAT-2806-A include limits on deflection or relative movements of fuel and control assemblies sufficient to support insertion of control rods.

7.1.1.1.8.2 Radionuclide Retention Functions

PSAR section 7.1.1.2 states that the RCC supports DL3-RR2, “Cladding Barrier,” which is SR. PSAR table 5.2-3 identifies an associated SRDC for the fuel cladding to retain radionuclide fission products sufficient to provide reasonable assurance that the 10 CFR 50.34 dose criteria are met for accidents. To support this function, PSAR section 7.1.1.2 states that the fuel assemblies maintain natural circulation capability and coolable geometry by limiting structural deformation to acceptable ranges and maintaining the fuel cladding within thermal-mechanical constraints, as well as retaining fission products to establish reasonable assurance that the radionuclide release remains below 10 CFR 50.34 release limits. Fuel cladding integrity is evaluated for the LBEs discussed in PSAR chapter 3, except for those LBEs in which fuel cladding is assumed to fail (e.g., fuel handling accidents and local faults). The fuel design criteria that ensure cladding integrity are described in detail in NAT-2806-A. Based on the PSAR chapter 3 safety analysis demonstrating that design basis events (DBEs) and DBAs remain within the 10 CFR 50.34 dose criteria, the staff considers the preliminary design of the RCC, particularly role of the fuel in radionuclide retention, to be consistent with DL3-RR2.

7.1.1.1.8.3 Heat Removal Functions

PSAR section 7.1.1.2 indicates that the RCC supports several heat removal functions, all related to passive heat removal via natural circulation. These are:

- DL3-HR5 – Natural Circulation of Sodium in Primary System (SR)
- DL3-HR6 – Passive heat removal in the Ex-Vessel Handling Machine (SR)
- DL3-HR7 – Passive Heat Removal in the Ex-Vessel Storage Tank (SR)
- DL3-HR8 – Passive Heat Removal in the Bottom Loading Transfer Cask (SR)

- DL3-HR9 – Passive Heat Removal in Pin Removal Cell (SR)
- DL3-HR10 – Passive Heat Removal in the Water Pool Fuel Handling System (SR)
- DL3-HR13 – Passive Heat Removal in the Failed Fuel Canister (SR)

PSAR table 5.2-3 identifies an SRDC associated with DL3-HR5 that requires the reactor design to support natural circulation to remove heat from the core at a rate sufficient to maintain dose below the 10 CFR 50.34 criteria. For the other functions listed, the SRDCs are similar but relate only to the fuel design.

The PSAR states that RCC supports DL3-HR5 by providing (a) heat that drives natural circulation flow and (b) passages to guide and control the flow of sodium coolant. Decay heat, which drives natural circulation, is an intrinsic property of the nuclear fuel. The flow passages are explicitly provided for by the RCC design and are assured by the fuel acceptance criteria provided in NAT-2806-A. The staff therefore determined that the preliminary RCC design is consistent with this function.

The PSAR states that RCC supports DL3-HR6 through HR10 and HR13 because it maintains the natural circulation capability and coolable geometry of the fuel by limiting structural deformation of the fuel to within acceptable ranges. As discussed previously, limits on fuel assembly deformation are provided as key acceptance criteria in NAT-2806-A. As such, the staff determined that the preliminary RCC design is consistent with these functions.

7.1.1.1.9 Programmatic Special Treatments

Programmatic special treatments applied to the RCC are summarized in PSAR section 7.1.1.4 and include the quality assurance program (QAP), design reliability assurance program (D-RAP), equipment qualification (EQ) program, post-construction inspection, testing, and analysis program (PITAP), and the fuel qualification and testing described in NAT-2806-A.

Although RCC components are reactor internals, the core itself is not subject to the comprehensive vibration assessment program (CVAP). This is consistent with the scope defined in RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Startup Testing," Rev. 4 (ML070260376), and is acceptable to the staff. NAT-2806-A states that mechanical testing, including vibration, will be performed for fuel and control assemblies.

The RCC is also outside the scope of reliability and integrity management (RIM) program under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) section XI (XI), division 2 (XI-2). The staff determined that this is reasonable because fuel and control assemblies are generally outside the scope of the ASME BPVC (notwithstanding certain criteria that may be derived from BPVC requirements as discussed in NAT-2806-A), and because RCC components are replaceable and relatively short-lived.

Based on its review of the functions of the RCC against the scope of the QAP, D-RAP, EQ, PITAP and the fuel qualification and testing described in NAT-2806-A, the staff determined the RCC special treatments are appropriate. Plant programs are evaluated further in SE chapter 8.

7.1.1.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.1.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.1.2 Reactor Enclosure System

PSAR section 7.1.2 describes the RES, which provides radionuclide retention, reactivity control, and reactor heat removal functions during normal and off-normal conditions. The RES forms a significant portion of the primary coolant boundary, which includes portions of the systems enclosing the primary sodium coolant and primary coolant cover gas. The RES includes the reactor vessel (RV) and head, the reactor internal structures, the guard vessel (GV), and the reactor support structures. The RV is a vertical, cylindrical shell design with an integrally welded, torispherical bottom head. The reactor head, including the rotatable plug assembly (RPA), forms the top head closure for the RV. Internal structures within the RV provide structural support for reactor core assemblies and PHT components, guide primary coolant flow through the core, define sodium pool regions (hot pool, cold pool, and warm pool), and provide in-vessel core assembly storage. The RV is surrounded by a GV, which provides a defense-in-depth (DID) coolant and functional containment boundary in the event of RV leakage. The RES is supported by the reactor support structure (RSS), which transmits loads to the RXB substructure.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the RES:

- RG 1.20 Rev. 4;
- RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Rev. 4 (ML13211A485);
- RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," Rev. 1 (ML15026A664);
- RG 1.44, "Control of the Processing and Use of Stainless Steel," Rev. 1 (ML101680225);
- RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," Rev. 1 (ML090820425);
- RG 1.87, "Acceptability of ASME Code, Section III, Division 5, High Temperature Reactors," Rev. 2 (ML21091A276);
- RG 1.100, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Rev. 4 (ML19312C677);
- RG 1.124, "Service Limits and Loading Combinations for Linear-Type Supports," Rev. 3 (ML13141A666);

- RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell Type Component Supports," Rev. 3 (ML13141A667); and
- RG 1.246, "Acceptability of ASME Code, Section XI, Division 2, 'Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,' for Non-Light Water Reactors," Rev. 0 (ML22061A244).

PSAR sections 7.1.2 and 5.3 identify the following PDC as applicable to the RES: 1, 2, 3, 4, 10, 11, 12, 14, 15, 16, 26, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 61, 62, 72, 73, 74, 78, 79, and 80.

7.1.2.1 Technical Evaluation

7.1.2.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. For the RES, nearly all components are SR and, consistent with the guidance in RG 1.87, will be designed to ASME BPVC section III (III), division 5 (III-5). Based on the endorsement of ASME BPVC III-5 in RG 1.87, the staff determined that the use of ASME BPVC III-5 for SR SSCs is consistent with PDC 1 for the preliminary design of the RES.

The staff notes that ASME BPVC III-5 does not address environmental effects on materials, and HBB-2160(a) from ASME BPVC III-5 specifically indicates that the Owner is responsible for giving "specific attention ... to the effects of service conditions upon the properties of the material." NAT-13478 states that resolution of environmental effects on materials may include assumptions on through-life design properties, which could be validated through future testing or performance monitoring, such as materials surveillance during plant operations. The staff determined that this is a reasonable approach to satisfy HBB-2160(a) at the preliminary design stage and will review the approach to address environmental effects on through-life design properties during the OL review of the final design.

As noted in PSAR table 7.1.2-4 and NAT-13478, the material of construction for most major RES components is 316H stainless steel with ER16-8-2 weld metal. NAT-13478 also states that the applicant is working with ASME code committees to extend the time-dependent allowable stresses for 304H and 316H SS from the current limit of 300,000 hours to a proposed limit of 500,000 hours to support the 60-year design life of the Sodium reactor. The applicant is using interim 500,000 hour allowable stresses at the preliminary design stage while working to get ASME code updated prior to the OL application. The staff confirmed through audit (ML25302A443) that the interim properties and code updates also cover the weld metal allowable stresses for ER16-8-2. The staff notes this approach of working to update ASME code while using interim properties for design is reasonable. The staff will review the applicant's use of 500,000 hour allowable stresses at the OL stage.

Other SR SSCs in the RES include the head access area (HAA) seal and the well seal, which will both be designed and constructed to ASME AG-1-2019. ASME AG-1-2019 generally covers design of filters, ventilation, and process gas treatment and is endorsed by the NRC in RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Rev. 3 (ML16070A277), for the design, inspection, and testing of normal atmosphere cleanup systems for controlling releases of airborne radioactive materials to the environment in light water reactors (LWRs). Given the preliminary design and function of these SSCs and the limited

information provided in the PSAR, the staff did not make a determination regarding the use of ASME AG-1-2019 for the HAA and well seals at the CP stage. The staff will review the justification for the design and construction codes and standards of these SSCs at the OL application. Given that these are not expected to be long lead time components, the staff determined that this issue may be reasonably left for later consideration.

Guard Vessel

The GV is the only SR SSC in the RES that will not be designed and constructed to a nuclear code and standard, such as ASME BPVC III-5. The PSAR states that RV leakage is considered “unlikely,” and the staff noted that there are no LBEs involving RV leakage in PSAR chapter 3. Through audit discussions, the staff confirmed that the applicant considers the radionuclide retention function DL4-RR1b to be NSRST for DID adequacy based on an assumption that RV leakage frequency would be below the DBE region. However, the applicant did not provide sufficient justification at the CP stage to demonstrate that RV leakage frequency is below the DBE region at the 95th percentile. Therefore, the staff notes that the safety classification of the radionuclide retention function for the GV is expected to be justified at the OL stage and may ultimately be reclassified. While this radionuclide retention function may need to be reclassified in the final design, the GV is already classified as SR based on its heat removal function. As such, the GV is designated as quality level (QL)-1, and the quality assurance requirements of 10 CFR Part 50 “Domestic Licensing of Production and Utilization Facilities,” Appendix B “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” and the reporting requirements of 10 CFR Part 21, “Reporting of Defects and Noncompliance,” are applied to its design, construction, and maintenance.

The staff reviewed the design and construction standards for the GV with awareness of the uncertainty regarding the safety classification of its radionuclide retention function, focusing on the fact that the GV is not the primary boundary and is only needed to perform this function in the event of RV leakage. PSAR table 7.1.2-1 states that the GV will be designed and constructed to ASME BPVC section VIII (VIII), division 1 (VIII-1). In addition, supplemental analysis will be performed using the applicable design rules from ASME BPVC III-5, subsection HBB-3225, for Service Level D service limits under the environmental and design conditions associated with postulated RV leakage. This analysis is intended to demonstrate the GV’s capability to perform its required functions in the event of a RV leak. Furthermore, the staff notes that PSAR table 7.1.2-1 states that all pressure boundary welds will be volumetrically examined to ensure weld quality and that certified material test reports (CMTRs) will be required. Given the GV’s role in radionuclide retention only in the event of RV leakage, the staff determined that the use of ASME BPVC VIII-1, with the identified special treatments, is consistent with PDC 1 for the preliminary design of the GV. The staff will review the final design of the GV and its associated functions at the OL stage.

Other SSCs

Other SSCs in the RES include the fixed in-vessel shielding (FIVS) canisters and the GV omega seal. The FIVS will be designed and constructed to ASME BPVC III-5, made of an austenitic stainless steel allowed by ASME BPVC III-5, and is not identified as safety-significant (SR or NSRST) in the PSAR. [[

]] Based on the endorsement of ASME BPVC III-5 in RG 1.87, the staff determined that the use of ASME BPVC III-5 for the FIVS design is consistent with PDC 1.

The GV omega seal is classified as NSRST and contributes to functional containment by preventing the leakage of potentially radioactive gases from the RV-GV annulus to the HAA. The omega seal will be made of austenitic stainless steel, welded to the reactor head in accordance with ASME BPVC III-5 and to the GV in accordance with ASME BPVC VIII-1. The staff determined that the preliminary design information regarding the codes and standards for the GV omega seal is consistent with PDC 1 because it will be joined to the RV and GV consistent with the specified codes and standards for those SSCs. The staff will review the final design of both the FIVS and GV omega seal during the OL stage.

Summary

Based on its review described above, the staff determined that the preliminary RES design is consistent with PDC 1 because the codes and standards specified for the design and construction of the RES SSCs (other than the HAA and well seals) are acceptable.

7.1.2.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.1.2.2 states that the RES is located within, and protected by, the SR reinforced concrete RXB substructure. This structure provides protection from external flooding, while the RES's location below grade provides protection from other natural phenomena including tornadoes and extreme climate conditions.

The applicant states that the RES is designed to withstand the effects of earthquakes in accordance with the seismic design methods described in PSAR section 6.4.1. RES components are classified as SCS1 and are qualified to withstand seismic loads associated with the SSE without loss of the capability to perform SR functions.

The staff determined that the preliminary RES design is consistent with PDC 2 because the RES is located within the SR RXB substructure which protects it from design basis external hazards and because the SR components of the RES are classified as SCS1.

7.1.2.1.3 Fire Protection (PDC 3)

PDC 3 requires that safety-significant SSCs are designed and located to minimize probability and effect of fires and explosions. PSAR section 7.1.2.2 states that the RES is designed to minimize the potential for sodium leakage and will be constructed using non-combustible and non-fire sustaining materials. The potential for sodium-air reactions is minimized through the use of inert cover gas over the primary coolant free space and within the RV and GV annular space. Based on the use of inert gas spaces to minimize sodium reactions, the staff determined that the preliminary design information for the RES is consistent with PDC 3. Further evaluation of the fire protection systems and program can be found in SE sections 7.5.2 and 8.3.

7.1.2.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, AOOs, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. PSAR section 7.1.2.2 states that materials selection for

primary coolant boundary components considers compatibility with the sodium environment, service temperatures, service degradation of material properties, and applicable degradation mechanisms. The applicant also stated that the location of the RES within the RXB substructure provides protection from external hazards caused by transportation incidents and offsite industrial or chemical facilities.

Demonstrating conformance with PDC 4 for safety-significant SSCs depends on the selected materials, their operating environment, and the safety-significant functions they perform. For components in less severe environments (e.g. lower temperatures, less aggressive chemistry, lower fluence levels), environmental effects are expected to be more limited. Accordingly, materials selection, design, analysis, and performance monitoring are not expected to be as intensive, consistent with the milder operating environment.

For the RES, several safety-significant components will operate in relatively mild environments at temperatures below the creep regime, without normal sodium exposure, and with minimal irradiation. PSAR section 7.1.2.3 indicates that the well seal will be constructed from low alloy steel and is exposed to an ambient air service environment. The HAA seal will be constructed from materials suitable for its service environment and loading conditions and will be qualified by the EQ program if non-metallic materials are used. The GV omega seal will be constructed from austenitic stainless steel and welded to the reactor head and guard vessel in accordance with the codes of construction for those components. Each of these components will be exposed to an ambient air service environment without significant irradiation effects.

Based on the mild service environments and the information provided regarding materials of construction and design, the staff determined that the preliminary design information for the well seal, HAA seal, and GV omega seal is consistent with PDC 4. The staff will review the final design of these components at the OL stage.

Guard Vessel

PSAR table 7.1.2-4 states that the GV and GV flange will be constructed of 2-1/4Cr-1-Mo steel. The normal operating environment for the GV includes temperatures below 650°F, with inert gas on the internal surface and outdoor air on the external surface. The staff notes that this design temperature is below the threshold at which creep effects may become significant, and inert gas and outdoor air are mild chemistry environments. In addition, the staff confirmed in audit that neutron irradiation levels for the GV are expected to have a negligible effect on material properties.

One important postulated accident for the GV operating environment is the Loss of Heat Sink with RAC (DHS-ISTL-2). In this event, GV temperature will increase significantly, reaching up to 850°F for less than 150 hours (as noted in table 4-1 of NAT-13478). This short-duration high temperature event will be evaluated per ASME BPVC III-5, nonmandatory appendix HBB-T, or ASME code case 2843-3, as discussed in PSAR section 6.4.5.1. Given the short duration of this postulated event and the applicant's plans to evaluate creep effects using appropriate ASME BPVC rules, the staff determined that this approach is acceptable for analyzing the GV during a DHS-ISTL-2 event, consistent with PDC 4 requirements for environmental compatibility during postulated events.

A sodium leak into the RV-GV annulus is another important postulated event that affects the GV's environmental conditions. This event, discussed under PDC 1 above, is currently classified as an other quantified event (OQE) but could be classified as a DBE or Beyond design basis

event (BDBE), pending additional analysis at the OL stage. In this event, the GV would be exposed to primary sodium on its inner surface. Note 1 of PSAR table 7.1.2-1 identifies that an analysis of the GV will be performed using ASME BPVC III-5, Level D Service Limits, for the environmental and design conditions associated with postulated RV leakage and will be provided to the staff to support its OL review. Given the availability of leakage monitoring in the RV-GV annulus to quickly detect sodium leakage, and the applicant's plans to analyze the effects of RV leakage on the GV, the staff determined that this preliminary information on the GV design is consistent with PDC 4. Further information can be reasonably left for later consideration at the OL review.

The approach and basis for precluding RV rupture is evaluated in SE section 7.1.2.1.6 with respect to the use of leakage monitoring of the primary coolant boundary. Notably, the applicant confirmed during audit discussions that a catastrophic rupture of the RV will be precluded through leakage monitoring of the RV-GV annulus, based on leak before break (LBB) analyses as described in section 10 of NAT-13478. Therefore, the GV is not designed or analyzed to perform in the event of a catastrophic RV rupture.

RES SSCs in Contact with Primary Sodium or Primary Cover Gas

PSAR table 7.1.2-4 identifies the base and weld materials, design temperatures, and service environments for major components of the RES. With the exception of the GV, all RES components in this table are fabricated from 316H SS, using a weld filler metal approved in table HBB-I-14.1(b) of ASME BPVC III-5. Table 5-2 of NAT-13478 further identifies that ER 16-8-2 will be used as the weld material for these components, which is approved for use in table HBB-I-14.1(b) of ASME BPVC III-5. Some of these RES components are a part of the primary coolant boundary, while others are internal to the RV and support functions, including core support, maintaining the expected coolant flowpath, providing structural support for reactivity control, or enabling instrumentation access. These components are exposed either to liquid sodium or primary cover gas, which consists of argon containing sodium vapor and aerosols. Notably, the reactor head thermal shield plates, which are located in the sodium cover gas space and supported by the RV, improve the environmental conditions for the reactor head (RH) and other SSCs on the RH by reducing the temperature and neutron flux on the RH as well as limiting sodium aerosols in the upper cover gas space.

The remainder of SE section 7.1.2.1.4 contains the staff's evaluation of the preliminary design information addressing environmental effects on safety-significant, long-lived, passive SSCs in contact with sodium or cover gas containing sodium aerosols. This section is referenced by SE sections 7.1.3 on the PHT, 7.1.4 on the IHT, 7.2.2 on the intermediate air cooling system (IAC), 7.2.3 on the sodium cover gas system (SCG), 7.2.4 on the sodium processing system (SPS), and 7.2.5 on the CRD. The staff's review focused on understanding how the applicant evaluated environmental effects on these SSCs to inform its monitoring and non-destructive examination (MANDE) approaches at the final design stage. This included consideration of existing and historical data, operating experience, planned or ongoing testing, and analysis methods that would provide a technical basis for demonstrating appropriate materials performance, given the material, operating environment, and safety function of the relevant SSCs.

Reliability and Integrity Management Degradation Mechanism Assessment

The PSAR and NAT-13478 state that the applicant will implement the RIM program under ASME BPVC XI-2. The RIM process is described in section 6 of NAT-13478 and evaluated in SE section 8.1. In implementing the RIM process, the applicant performs a degradation

mechanism assessment (DMA) for all safety-significant SSCs to inform the development of RIM strategies that ensure all components in the scope of RIM meet their specified reliability targets. The DMA considers design characteristics (e.g. material and geometry), fabrication practices (e.g. welding and heat treatment), operating environment (e.g. temperature, pressure, chemistry, irradiation), and relevant operating experience and research results.

Section 7 of NAT-13478 describes the process to identify and evaluate degradation mechanisms relevant to the Sodium design based on ASME, NRC, and International Atomic Energy Agency (IAEA) technical and guidance documents. The following mechanisms were identified as potentially relevant for components exposed to sodium or cover gas:

- Irradiation effects
 - Neutron irradiation embrittlement
 - Neutron irradiation-induced void swelling
 - Neutron irradiation stress relaxation in bolting
 - Neutron irradiation and sodium effects on creep strength and creep embrittlement
- High temperature effects
 - Thermal aging embrittlement
 - Stress relaxation cracking (SRC)
 - Creep embrittlement
- Liquid and gaseous sodium effects
 - General, galvanic, and flow-accelerated corrosion
 - Mass Transfer, carburization, and decarburization
 - Liquid metal embrittlement
 - Stress corrosion cracking
 - Environmentally assisted fatigue
- Fatigue and creep effects
 - Thermal fatigue (stratification cycling and striping)
 - Thermal fatigue (thermal transients)
 - Fatigue creep interaction

- Deformation (stress relief)
- Deformation (ratcheting)
- Fretting, wear, cavitation

The staff notes that this is a reasonable list of degradation mechanisms to consider for safety-significant SSCs exposed to sodium or cover gas.

Table 7-1 of NAT-13478 summarizes the preliminary screening criteria for most safety-significant SSCs exposed to sodium or cover gas. For some mechanisms, these criteria are supported by technical basis documents that evaluate operating experience and research literature. For others, only the high-level criteria were developed.

During the review, the staff audited both the preliminary screening criteria and technical basis documents for mechanisms where such documentation was available. However, the staff did not make any determinations regarding the acceptability of the applicant's preliminary screening criteria, given the preliminary nature of the screening criteria. The staff's review at the preliminary design stage focused on the criteria for screening out a mechanism as not being active over the 60-year planned operating life. The staff documented observations on the preliminary screening criteria and supporting technical basis documents in Enclosure 2 of the audit report (ML25302A443); these observations include areas for focus to appropriately support and justify the screening criteria at the final design stage.

The staff emphasized in the observations in the audit report the need for the RIM process, including the DMA and its screening criteria, to adequately consider the combined effects of multiple degradation mechanisms or stressors (e.g. temperature, sodium, irradiation) on particular SSCs. The staff also noted the inherent uncertainty introduced from long-term combined effects that cannot be captured in short-term, separate effects testing. Additionally, the staff highlighted the need to consider the potential for less than ideal conditions (e.g. local chemistry variations, chemistry transients, residual stresses, materials variability) and the potential for unexpected degradation due to lack of knowledge or experience, particularly over extended operating periods. Experience from operating LWRs has shown that such unexpected degradation can occur and to ensure safety must be detected early to ensure continued safety.

Materials Testing Activities and Stress Relaxation Cracking

Section 9 of NAT-13478 describes materials testing activities and plans to improve understanding of the materials for structural applications in the KU1 design. These testing activities and plans address a number of important long-term degradation mechanisms, including irradiation effects, liquid and gaseous sodium effects, and fretting and wear. These ongoing and planned activities complement efforts to acquire and qualify historical data as described in section 8 of NAT-13478. The staff notes that one key limitation of historical data and experience is the limited information on environmental effects (e.g., irradiation and sodium exposure) on the ER16-8-2 weld filler metal, which is planned for most safety-significant SSCs exposed to sodium. Section 5 of NAT-13478 notes that ER16-8-2 was used with 316 SS in the Fast Flux Test Facility (FFTF) hot leg, which appears to be the only other sodium service experience for this weld material. The planned and ongoing testing activities described in section 9 of NAT-13478 are therefore a valuable means of improving knowledge and reducing uncertainty regarding the long-term environmental effects on both base and weld materials, including ER16-8-2. The applicant notes that irradiation and sodium effects testing programs are

intended to provide guidance for surveillance or RIM activities, which the staff notes is a useful approach for developing an appropriate basis for the scope and methods of performance monitoring.

One degradation mechanism identified in the DMA that is particularly sensitive to design, fabrication, and construction practices is SRC. SRC is associated with the heat affected zone of austenitic material welds and can lead to cracking early in operation. It is influenced by the residual stress and degree of constraint in a given weldment and can be managed by avoiding the SRC-susceptible temperature range or by performing adequate post-weld heat treatment to relieve weld residual stresses. Section 9 of NAT-13478 describes a testing program pursued by the applicant to investigate weld and fabrication practices aimed at reducing the likelihood of SRC in the selected base and weld materials. The staff notes this is a prudent approach that promotes design and fabrication practices to reduce the likelihood of SRC. However, since precluding the potential for SRC is challenging, in-service inspection activities may be needed to confirm that SRC has not occurred in reactor components. The staff will review the final design, along with the approach to addressing SRC through the RIM program, at OL application stage.

Fixed In-Vessel Shielding

PSAR section 7.1.2.3 describes the design and function of the FIVS canisters. These canisters are filled with boron carbide and surround the reactor core to minimize sodium activation. As discussed in the PDC 1 evaluation, the FIVS canisters will be designed and constructed to ASME BPVC III-5 and made of an austenitic stainless steel allowed by ASME BPVC III-5. These canisters will be located in a liquid sodium-filled annulus space characterized by low flow or stagnant sodium conditions. Over time, the canisters are expected to pressurize due to helium gas generation as boron carbide depletes. The applicant states that the canisters will be designed to withstand anticipated loading conditions, including seismic loads and internal pressurization, in accordance with ASME BPVC III-5. Additionally, the applicant plans to complete a detailed failure modes and effects analysis by the final design stage. This analysis will include consideration of the effects of canister leakage or rupture when pressurized.

The staff notes the unique hazard posed by the presence of these pressurized canisters within the RV. The FIVS location in a low flow or stagnant environment could create unfavorable chemistry conditions over time, due to limited exchange of liquid sodium from this space. This may increase the potential for corrosion or other sodium-related degradation mechanisms. However, the low flow environment also reduces the likelihood that boron carbide particles released from a leaking FIVS canister would be transported beyond the annulus space into the rest of the primary system.

Based on the planned materials, identified codes and standards, and the expected inclusion of the FIVS in the RIM program (due to its anticipated NSRST classification), the staff determined that the preliminary information on the design and construction of the FIVS is consistent with PDC 4. The staff will review the final FIVS design, including any in-service performance monitoring, and the failure modes and effects analysis, during the OL application review.

R&D Item on Assuring Materials Performance

PSAR chapter 13 and section 12.1 of NAT-13478 describe an R&D item associated with assuring adequate structural materials performance in service. This R&D effort includes research activities to improve the understanding of environmental effects, as described in section 9 of NAT-13478, as well as the identification of additional materials testing needs to

ensure environmental compatibility. This R&D item also includes the development of performance monitoring methods, such as material surveillance programs, inspection methods and means of access, and monitoring approaches supported by validated technical bases. Further details on the preliminary inspection and monitoring approaches are discussed and evaluated in SE sections 7.1.2.1.6 and 7.1.2.1.10. The staff proposed two conditions for the KU1 CP related to updates on this R&D item as discussed in greater detail in SE chapter 13.

Based on the identified R&D item, the staff determined that the preliminary design information for RES SSCs in contact with sodium or cover gas is consistent with PDC 4. The staff will evaluate the maturation of the RIM DMA and associated MANDE activities through the required R&D updates, as well as the final design during the OL application review.

7.1.2.1.5 Reactor Design, Reactor Inherent Protection, Suppression of Reactor Power Oscillations (PDC 10, 11, and 12)

PDC 10 requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to ensure that SARRDLs are not exceeded during normal operation or AOOs. PSAR section 7.1.2.2 states that the reactor internal structures maintain alignment of control rod drivelines with control assemblies to provide absorber insertion for reactivity control. The reactor internal structures maintain core support and configuration for adequate coolant flow and heat removal under forced flow and natural circulation conditions to provide margin to SARRDLs during normal operation or AOOs. The staff confirmed that this is demonstrated based on the AOO analyses provided in PSAR section 3.6, where no AOOs result in fuel limits being exceeded. The staff will conduct further detailed evaluation of the core internals during the OL application review to verify compliance with the design codes and standards.

PDC 11 requires the inherent reactivity mechanisms of the core and associated systems to compensate for rapid increases in reactivity. PSAR section 7.1.2.2 states that the design of the RES structurally supports and maintains the geometry of the reactor core assemblies. The RES reactor internal structures contribute to reactivity feedback mechanisms primarily through the design of:

- the lower core support plate, which provides reactivity feedback via radial expansion of the core;
- the core barrel, which affects reactivity feedback by enabling assemblies to bow using the core restraint system; and
- the reactor vessel, which provides reactivity feedback via thermal expansion, which moves the core relative to the control rods.

The staff confirmed that the nuclear and thermal hydraulic design analyses described in PSAR section 3.12 and 3.13 and provided in TP-LIC-RPT-0011 demonstrate that the net result of these feedback mechanisms is to compensate for a rapid increase in reactivity. The staff also confirmed that the LBE analyses described in PSAR sections 3.6 through 3.9 demonstrate the same behavior during transients.

PDC 12 requires power oscillations that can result in conditions exceeding SARRDLs to not be possible or to be reliably and readily detected and suppressed. PSAR section 7.1.2.2 states that

the design of RES contributes to limiting power oscillations that could result in conditions exceeding SARRDLs by restraining core assembly lateral deflection, distortion, or excessive movements that could occur during normal operation or off-normal events. The staff confirmed that the core stability analysis discussed in PSAR section 3.12.1.6 demonstrates that there are no identified instabilities over the evaluated power range and was found acceptable in SE sections 3.11.1.5 and 7.1.1.1.3.

The staff determined the RES design is consistent with PDC 10, 11, and 12 based on the analyses provided in chapter 3 of the PSAR.

7.1.2.1.6 Primary Coolant Boundary (PDC 14, 15, 30, 31, 32, and 33)

PDC 14 requires primary coolant boundary components to be designed, fabricated, erected, and tested to provide an extremely low probability of leakage, rapidly propagating failure, and gross rupture. PDC 15 requires the design conditions of the primary coolant boundary to not be exceeded during conditions of normal operation, including AOOs. PDC 30 requires primary coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 31 requires primary coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 32 requires primary coolant boundary components to permit periodic inspection and functional testing. PDC 33 requires a system to maintain primary coolant inventory for protection against small breaks in the primary coolant boundary as necessary to ensure SARRDLs are not exceeded.

Design of Primary Coolant Boundary (PDC 14, 15, 30, and 31)

PSAR section 7.1.2.3 states that there are no penetrations through the RV, and that all primary coolant boundary penetrations through the reactor head are above the operating level of primary sodium within cover gas space. The PSAR also states that mating surfaces of these penetrations will incorporate redundant sealing methods. While the static seal designs associated with these penetrations are not yet finalized, they may involve metallic seal designs such as metallic O-ring seals. If non-metallic seal designs are used, the non-metallic materials will be qualified for the service environment in accordance with the EQ program. PSAR section 7.1.2 references ASME-QME-1-2023 for mechanical EQ, and PSAR table 1.4-1 states partial conformance with RG 1.100, Rev. 4, which endorses ASME QME-1-2017. The staff notes that the NRC has not yet endorsed ASME QME-1-2023. However, the staff reviewed ASME QME-1-2023 and considers it appropriate if implemented consistent with the regulatory positions specified in Rev. 4 of RG 1.100. Based on the use of redundant sealing methods and the plan to qualify non-metallic materials through the EQ program in accordance with ASME QME-1-2023, the staff determined that the preliminary design information for the static seals is acceptable at the CP stage and consistent with PDC 14, 15, and 30.

The preliminary design for the RPA dynamic seal includes the use of an elastomeric material in a multiple-element seal configuration, with argon gas pressurization between seal elements, or an inflatable seal pressurized by argon gas. PSAR section 7.2.3.2.3 states that the SCG provides this argon and also monitors for leakage. The seal design and materials will be qualified for the service environment under the EQ program, in accordance with ASME QME-1. Based on the proposed seal design with multiple barriers, monitoring capabilities, and plans to qualify the design and materials through the EQ program, the staff determined that the

preliminary design information for the RPA dynamic seal is acceptable at the CP stage and consistent with PDC 14, 15, and 30.

PSAR section 7.1.2.2 states that the primary coolant boundary operates at low pressure and that metallic boundary components are designed and constructed per the requirements of ASME BPVC III-5. The applicant stated that the design and material selection for RES primary coolant boundary components consider service temperatures, service degradation of material properties, and applicable degradation mechanisms. Metallic primary coolant boundary components are constructed of austenitic stainless steel to minimize the effects of irradiation embrittlement and fracture from thermal stresses. The primary coolant boundary is designed and constructed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC III-5. Based on conformance with the staff endorsed requirements in ASME BPVC III-5 and the staff evaluation of environmental effects on primary coolant components documented in the discussion regarding PDC 4 above, the staff determined that the preliminary design of the RES is consistent with PDC 14, 15, 30, and 31.

Inspection of Primary Coolant Boundary (PDC 32)

The applicant stated in the PSAR that the design of the RES permits periodic inspection and functional testing of important areas and features to assess structural and leaktight integrity. This is achieved through implementation of MANDE pre-service and in-service inspection requirements of the RIM program. Section 10 of NAT-13478 describes the MANDE practices being considered for the RIM program. These include:

- Standard nondestructive examination assessment (surface, volumetric and visual)
- Eddy current exams of intermediate heat exchanger (IHX) tubes
- Leakage monitoring system
- Creep-fatigue monitoring system
- Under sodium viewing system
- Future planned MANDE activities
 - Loose parts monitoring system
 - Piezoelectric thick-film sensors
 - Fiber-optic sensors
 - Materials surveillance coupon program

The RES primary coolant boundary leakage monitoring approach includes detection within the RV-GV annulus. Leakage into the GV would be detected through multiple means. Liquid sodium detectors provide direct indication of sodium leakage into the annulus. Additionally, the recirculated argon gas in the RV-GV annulus is monitored for increased radiation levels, changes in oxygen concentration, and increased gas filter differential pressure, all of which could indicate RV leakage.

To support the development of the leakage monitoring approach, calculations based on LBB methodologies are performed to determine necessary leakage detection capabilities. The leak rate associated with the critical flaw size for a given component will be used to inform the design of appropriate leak monitoring systems. NAT-13478 states this approach is under consideration for the RV and high temperature piping in the IHT, SCG, and SPS. LBB concepts have been applied to water-based systems (e.g., NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," section 3.6.3 (ML063600396)); however, there is significant uncertainty in correlating crack morphology to leak rate even in LWR systems with significant research and operating experience. Therefore, while sodium and water share similar density and viscosity, differences in pressure, melting points, and boiling points introduce substantial additional uncertainty when attempting to correlate crack morphology to leak rate in sodium piping without new, sodium-specific validation data. Given the applicant's extensive proposed reliance on leakage monitoring as a RIM strategy, a validated technical basis for this approach is necessary. Additionally, sodium freezing and plugging is a known phenomenon in sodium systems. This issue is not a concern in high pressure water piping leak detection but can undermine the effectiveness of sodium leak detection. The use of probabilistic fracture mechanics (PFM) and leakage monitoring in the manner proposed by the applicant is novel and requires significant justification and a validated technical basis. The staff made no determinations around the preliminary leakage monitoring approach and will review the final design and its use of leakage monitoring, LBB, and PFM at the OL stage. The staff also notes that the applicant stated during audit discussions that a catastrophic rupture of the RV will be precluded through leakage monitoring of the RV-GV annulus supported by LBB analyses and the staff will review this approach at the OL stage.

Section 10 of NAT-13478 states that a creep-fatigue monitoring system (CFMS) is being evaluated as a potential MANDE to address creep-fatigue damage. The staff notes that the currently proposed baseline CFMS does not directly measure creep-fatigue damage in components but is instead limited to recording temperature and operating conditions. While this approach provides valuable information on plant transients, it does not resolve the primary uncertainty regarding how structural components respond to creep-fatigue under long-term service conditions. The CFMS approach uses stress transfer functions to estimate stress and temperature histories from plant instrumentation and then applies experimentally calibrated damage models to evaluate creep and fatigue damage. Although this strategy captures differences between actual operating transients and design assumptions, it does not quantify material degradation or account for environmental effects such as extended time at temperature or neutron irradiation, both of which have been shown to reduce fatigue and creep-fatigue life (Messner, M.C., et al.).

The applicant states in NAT-13478 that enhanced CFMS methods are being considered by employing technologies such as acoustic emission, ultrasonic testing, or strain sensors at critical locations for creep and crack monitoring. The staff notes that enhanced CFMS methods would help address the limitations of the baseline CFMS approach. In summary, while CFMS can provide valuable real-time operating data, its use as a RIM strategy for creep-fatigue monitoring requires justification, a validated technical basis, and may need to be supplemented with additional MANDE methods.

NAT-13478 also discusses plans to develop a materials surveillance coupon program to help address the combined effects of degradation mechanisms and time-dependent material property changes in the operating environment. These plans include identifying key characteristics for monitoring, selecting appropriate testing methods, relating surveillance data

to component life in service, configuring test coupons, and establishing a sampling strategy. The staff notes that a surveillance coupon program can enhance understanding of materials performance, particularly changes in material properties during long-term exposure to environmental effects, that can help provide assurance of component performance when other MANDE methods are impractical to implement.

The staff further notes that the potential MANDE methods discussed in NAT-13478 provide reasonable approaches for performance monitoring of SSCs in service. However, significant developmental work remains to demonstrate the effectiveness of many of these methods for the KU1 design, including under sodium viewing, leakage monitoring, and CFMS, as discussed above. The staff will review the selection of MANDE methods along with the justification and technical basis for their use during the OL review.

Based on the proposed implementation of the RIM program for in-service inspection of primary coolant boundary components, and the MANDE practices discussed above, the staff determined that the preliminary design of the RES is consistent with PDC 32. The staff evaluation of the RIM process is documented in SE section 8.1.1.4.

Primary Coolant Inventory (PDC 33)

PSAR section 7.1.2.2 states that there are no penetrations through the RV and the primary coolant boundary penetrations through the reactor head are above the operating level of primary sodium in the cover gas space. In the unlikely event of RV boundary leakage, the surrounding GV is designed to contain the leaked sodium. The annular space between the RV and GV is sized to ensure that the equilibrium coolant level remains sufficient to remove reactor core heat. The SPS includes piping that penetrates the reactor head and extends into the sodium coolant. The SPS pumps are tripped on a low primary sodium level by the reactor protection system (RPS) to protect against primary coolant inventory loss. Based on the design of the RES with the GV surrounding the RV, design features to minimize primary coolant leakage, and the proposed implementation of the RIM program for monitoring of primary coolant boundary leakage, the staff determined that the preliminary design of the RES is consistent with PDC 33.

R&D Item on Assuring Materials Performance

PSAR chapter 13 and section 12.1 of NAT-13478 describe an R&D item focused on assuring adequate structural materials performance in service. This R&D item covers research activities to support the RIM program and includes the development of performance monitoring methods, such as material surveillance programs, inspection methods and means of access, and monitoring approaches, as well as the development of PFM and LBB methodologies. The staff proposed two conditions for the KU1 CP related to updates on this R&D item as discussed in greater detail in SE chapter 13.

Based on the staff evaluation of materials compatibility for primary coolant boundary components in SE section 7.1.2.1.4, conformance with ASME BPVC III-5 and RG 1.87 for the design and construction of these components, and the proposed implementation of the RIM program and associated R&D activities to ensure adequate material performance in service, the staff determined that the preliminary design of the RES primary coolant boundary components is consistent with PDC 14, 15, 30, 31, 32, and 33.

7.1.2.1.7 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. PSAR section 7.1.2.2 states that the design of the primary coolant boundary supports functional containment to control the release of radioactivity to the environment and ensures that functional containment design conditions are not exceeded.

Functional containment performance is assessed through the mechanistic source term analyses described in sections 3.2 of the PSAR and this SE. At the CP stage, these analyses rely on assumptions regarding leakage rates across barriers. The staff will review leakage rate justifications at the OL stage based on the design of the SSCs that comprise each functional containment boundary, which in this case includes the primary coolant boundary components of the RES. The staff notes the RES design includes penetrations with redundant sealing methods and multiple barriers, such as the GV surrounding the RV, and the HAA seal, well seal, and GV omega seals. Therefore, the staff determined that the preliminary design of the RES is consistent with PDC 16 and will confirm this for the final design during the OL review.

7.1.2.1.8 Reactivity Control Systems, Reactivity Limits, and Protection Against AOOs (PDC 26, 28, and 29)

PDC 26 requires a minimum of two reactivity control systems or means to insert negative reactivity and maintain the reactor shutdown under various conditions. PDC 28 requires limits on the magnitude and rate of reactivity increase to prevent damage to the primary coolant boundary and core. PDC 29 requires the reactivity control system to be designed with an extremely high probability of accomplishing its safety function during an AOO.

The RES supports reactivity control by providing structural support to control rod drive mechanisms (CRDMs) and drivelines and by maintaining control rod driveline alignment with the control assemblies. This enables effective reactivity control and limits the potential magnitude and rate of reactivity increase during postulated reactivity events. The staff determined the preliminary RES design is consistent with PDC 26, 28, and 29 because it provides adequate structural support to the CRDMs and drivelines, allowing them to perform their necessary safety functions.

7.1.2.1.9 Residual Heat Removal (PDC 34)

PDC 34 requires that a system is provided to remove residual heat. PSAR section 7.1.2.2 states that the RES design supports residual heat removal during normal operations and AOOs by establishing and maintaining the flow path for both forced flow and natural circulation of primary coolant within the RV. Reactor core decay heat and other residual heat is transferred to the intermediate system through the IHX at a rate sufficient to ensure that SARRDLs and the RES design conditions are not exceeded. The flow characteristics of the PHT, as well as the capability of the PHT and IHT to remove residual heat, are evaluated in SE sections 7.1.3 and 7.1.4.

7.1.2.1.10 Emergency Core Cooling (PDC 35)

PDC 35 requires a system to ensure sufficient core cooling during postulated accidents and remove residual heat following postulated accidents. The reactor air cooling system (RAC), described in PSAR section 7.2.1, provides emergency core cooling in conjunction with the RES

and PHT. Following a loss of forced flow, natural circulation of primary sodium transfers heat to the RV wall, which is then radiated to the GV and collector cylinder assembly (CCA), heating air within the RAC that is ultimately rejected to the atmosphere. The staff's evaluation on the development of natural circulation and the RAC's heat removal capability are provided in SE sections 7.1.3 and 7.2.1.

PSAR section 7.1.2.2 states that the RV and GV heat transfer surfaces provide sufficient surface area and are enhanced through surface treatment to support heat transfer. PSAR section 7.1.2.3 states that the outer surface of the RV and the inner and outer surfaces of the GV will receive a surface treatment to enhance thermal radiation heat transfer. This surface treatment is identified as a subject of R&D in PSAR section 13.2.1. The supplemental RAC heat transfer performance R&D description (ML25274A124) indicates that a surface roughening approach will be used to increase the roughness of these surfaces [[

]]. This R&D item is evaluated by the staff in SE chapter 13.

In the supplemental RAC heat transfer performance R&D description, the applicant stated that [[

]].

[[

]] Because of this, and because the surface treatments are designed to improve heat transfer, the staff determined that the preliminary design of the RES is consistent with PDC 35. The staff will confirm the RV design with respect to heat transfer and fatigue performance as part of the OL application review.

7.1.2.1.11 Inspection and Testing of Emergency Core Cooling System (PDC 36 and 37)

PDC 36 and 37 address inspection and testing, respectively, for systems required for emergency core cooling. The PSAR indicates these requirements will be met through the RIM program and installed reactor instrumentation. In section 10 of NAT-13478, the applicant identifies potential MANDE methods for components within the RV relied upon for natural circulation. These include under sodium viewing (acoustic and sonar based technologies design to provide visual testing (VT)-1 or VT-3 level viewing), piezoelectric thick film sensors, vibration and loose parts monitoring, and the CFMS.

The staff notes significant uncertainties associated with materials performance in the reactor internals as discussed in SE section 7.1.2.1.4. Potential MANDE methods such as under sodium viewing and piezoelectric thick film sensors need significant development work to

demonstrate effectiveness comparable to visual inspection and ultrasonic testing applied in operating reactors. While vibration and loose parts monitoring is a prudent approach, it generally does not detect issues prior to component failure. Limitations of the CFMS proposed by the applicant are discussed in SE section 7.1.2.1.6.

The staff expects that appropriate MANDE methods will be identified, developed, and justified as sufficient by the OL application. As noted in SE section 7.1.2.1.4, the applicant has identified an R&D item focused on ensuring adequate structural materials performance in service. This includes development of performance monitoring methods such as material surveillance programs, inspection methods and means of access, and monitoring approaches supported by validated technical bases.

Based on this R&D item and the applicant's plan to develop, identify, and justify appropriate MANDE methods by the OL application, the staff determined that the preliminary RES design is consistent with PDC 36 and 37. Further information can reasonably be left for later consideration.

7.1.2.1.12 Fuel Storage and Handling and Radioactivity Control, Prevention of Criticality in Fuel Storage and Handling (PDC 61 and 62)

PDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. PDC 62 requires that criticality in the fuel storage and handling system be prevented by the physical systems or processes, preferably by use of geometrically safe configurations. The design of the RES includes in-vessel storage (IVS) for initial decay and cooling of core assemblies. Stored fuel is cooled by primary coolant flow through the IVS as part of the normal and emergency core cooling flow paths. The IVS is inspected and tested consistent with other RV internal structures and in accordance with the RIM Program. The staff's evaluation of inspection and testing of other RV internal structures is provided in SE section 7.1.2.1.11. The staff evaluated the in-vessel fuel handling system (FHI) in SE section 7.3.3 and the criticality safety methodology in SE section 3.13. Based on the identified RES design features to provide cooling for the IVS and the evaluations of FHI design and criticality safety for the IVS elsewhere in this SE, the staff determined that the preliminary design of the RES is consistent with PDC 61 and 62.

7.1.2.1.13 Sodium Heating Systems (PDC 72)

PDC 72 requires heating systems for safety-significant SSCs that contain sodium. The staff evaluated the design's capability to maintain adequate primary sodium temperature within the RES and the PHT, in conformance with PDC 72, in SE section 7.1.3. Based on the evaluation of PDC 72 in SE section 7.1.3, the staff determined that the preliminary design of the RES is consistent with PDC 72.

7.1.2.1.14 Sodium Leakage Detection and Reaction Prevention and Mitigation, Sodium-Water Reaction Prevention and Mitigation (PDC 73 and 74)

PDC 73 requires a means to detect sodium leakage and to limit and control sodium-air and sodium-concrete reactions. PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water.

PSAR section 7.1.2.2 states that the GV surrounding the RV is provided to contain potential sodium leakage and prevent sodium-concrete reactions. The annular RV-GV space is filled with argon gas to provide an inert atmosphere that prevents sodium-air reactions and associated fires. The GV includes sodium leakage detection features to identify potential RV leakage, which are discussed further in PSAR section 7.2.3.1.1 and evaluated by the staff in SE section 7.2.3. The RES is located within the RXB substructure, which, according to PSAR section 8.2.1, does not house water-filled systems, thereby limiting the potential for contact between primary sodium and water.

The staff determined that the preliminary RES design is consistent with PDC 73 because it includes a means for detecting and containing sodium leakage. The staff also reviewed the RES location within the Natrium facility and confirmed that it is situated to minimize the potential for sodium-water reactions as no water-containing systems are located within the RXB, consistent with PDC 74.

7.1.2.1.15 Primary Coolant System Interfaces (PDC 78)

PDC 78 provides requirements for SSCs containing liquids that interface with the primary coolant system. PSAR 7.1.2.2 states that the design of RES excludes interfaces with SSCs containing fluids that are chemically incompatible with the primary coolant. The interface between primary and intermediate sodium in the IHX is evaluated in SE section 7.1.3. The interface between primary sodium and argon in the cover gas space is evaluated in SE section 7.2.3. Based on the evaluation of the primary coolant system interfaces as discussed in these SE sections, the staff determined that the preliminary design of RES is consistent with PDC 78.

7.1.2.1.16 Cover Gas Inventory Maintenance (PDC 79)

PDC 79 requires a system to maintain cover gas inventory to ensure primary coolant sodium design limits are not exceeded as a result of cover gas loss due to small leaks or ruptures in the primary coolant boundary. Overpressure protection is provided by the SCG based on pressure design limits for RES components. The SCG and PDC 79 are discussed further in PSAR section 7.2.3 and evaluated in SE section 7.2.3. Based on the evaluation of PDC 79 in SE section 7.2.3, the staff determined that the preliminary design of RES is consistent with PDC 79. Final design limits for the primary coolant sodium associated with the RES will be reviewed during the OL application.

7.1.2.1.17 Reactor Vessel and Reactor System Structural Design Basis (PDC 80)

PDC 80 requires, in part, that the RV and reactor system are designed to maintain their integrity during postulated accidents, ensuring sufficient insertion of neutron absorbers to achieve reactor shutdown. PSAR section 7.1.2.2 states that the RV is designed to maintain the geometry of reactor internal structures to support natural circulation of primary coolant through the reactor core. These internal structures also support the reactor core and maintain CRDM driveline alignment, enabling sufficient insertion of the neutron absorbers to achieve reactor shutdown during postulated accident conditions. The RV, reactor head, and reactor internals structures are designed and constructed in accordance with the requirements of ASME BPVC III-5.

The RES is supported by the seismic isolation system (SIS). The design and qualification methodology for the SIS are described in TR NAT-8922, "Reactor Seismic Isolation System

Qualification Topical Report,” Rev. 2 (ML25195A156), and discussed in SE section 6.4.1.3. PSAR section 7.1.2.3 provides some preliminary design information for the SIS, which is consistent with NAT-8922. The methodology and preliminary design provide reasonable assurance that the RES will have sufficient structural support to perform its heat removal and reactor shutdown functions during DBAs. The final seismic design basis will be reviewed during the OL application, when detailed SIS design information and seismic PRA results are available.

The staff determined that the preliminary information on the RV and reactor system is consistent with PDC 80 because these components will be designed and constructed to ASME BPVC III-5 and will be supported by the SIS.

7.1.2.1.18 Safety Functions

7.1.2.1.18.1 Reactivity Control Function

PSAR section 7.1.2.2 states that the RES supports DL3-RC1, “Scram - Gravity driven absorber insertion by latch release,” which is SR. The RES supports reactivity control by establishing the position of reactor control assemblies relative to CRD components, thereby enabling proper control rod positioning and insertion into the reactor core. The staff determined that the preliminary RES design is consistent with the acceptance criteria presented in NAT-2806-A and as such is consistent with this function. However, because final evaluations of the RES against these acceptance criteria have not yet been completed, the staff will review how the final RES design supports this function at the OL stage.

7.1.2.1.18.2 Radionuclide Retention Functions

PSAR section 7.1.2.2 states that the RES supports DL3-RR1 “Primary Coolant Boundary including RES Barrier,” and DL4-RR1 “HAA Barrier, GV Leak Prevention Function,” which are SR and NSRST, respectively. The RES supports radionuclide retention by providing primary coolant boundary components and functional containment barriers. The GV, HAA, and well seals provide a DID functional containment boundary, while the GV serves as a backup to the primary coolant boundary. Based on the staff’s evaluation of materials compatibility of the primary coolant boundary components in SE section 7.1.2.1.4, the evaluation of the PDC related to the primary coolant boundary in SE section 7.1.2.1.6, and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in-service material performance, the staff determined that the preliminary RES design is consistent with its role as a barrier to radionuclide transport and these functions. As discussed in SE section 7.1.2.1.7, the staff will confirm design leakage rates associated with RES barriers are consistent with needed functional containment performance at the OL stage.

7.1.2.1.18.3 Heat Removal Functions

PSAR section 7.1.2.2 identifies that RES supports DL3-HR4 “Inherent RAC Operation,” and DL3-HR5 “Natural Circulation of Sodium in Primary System,” both of which are SR. For all LBEs that rely on DL3-HR4 and DL3-HR5, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The RES supports these SR functions by providing a natural circulation flowpath for primary coolant through the reactor core to facilitate decay heat removal. Additionally, the RV and GV provide heat transfer surfaces that enable heat transfer from the primary sodium to the air in the RAC. Evaluation of these heat transfer surfaces is discussed under PDC 35 above.

The staff audited the details of these LBE analyses and confirmed that the RES design is capable of dissipating heat at a rate necessary to mitigate LBEs that rely on natural circulation of primary sodium and heat rejection to the RAC. The staff determined that the RES design is consistent with these functions because fuel integrity is maintained for these LBEs. The staff notes that the design of RES SSCs relative to heat transfer performance is in development and is the subject of an R&D item, as discussed under PDC 35 above and in chapter 13 of the SE.

7.1.2.1.19 Programmatic Special Treatments

Programmatic special treatments applied to the RES are summarized in PSAR section 7.1.2.4 and include the QAP, D-RAP, EQ program, PITAP, CVAP, and RIM program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs, and therefore would apply to RES. Because the RES contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply as special treatments to the RES. Plant programs are evaluated further in SE chapter 8. Some aspects of these programs, such as the DMA for the RIM program, are evaluated in SE sections 7.1.2.1.4 and 7.1.2.1.6.

7.1.2.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.1.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.1.3 Primary Heat Transport System

PSAR section 7.1.3 describes the SSCs that make up the PHT. The following components are part of the PHT:

- Primary Sodium Pumps (PSPs)
- IHXs

The PHT's primary function is to transfer heat from the reactor core to the IHT for power generation. The PHT consists of two IHXs, two PSPs, and three sodium pools (hot, cold, and warm) within the RV. During normal operations, the PSPs drive primary sodium through the PHT, facilitating heat transfer from the fuel to the IHXs, where heat is transferred to the intermediate sodium in the IHT. The PHT operates in a closed cycle: sodium from the cold pool enters the PSPs, flows through the reactor core to absorb heat, moves into the hot pool, and finally passes through the IHXs to transfer heat to the intermediate sodium before returning to the cold pool. The PSAR states that, if the PSPs are unavailable, natural circulation driven by temperature differences within the PHT will continue to facilitate heat transfer. The PHT supports five SR functions and three NSRST functions, which are evaluated below. PSAR section 7.1.3.1.1 states that the PHT is located within the RXB substructure. PSAR figure 7.1.3-2 provides a simplified diagram of the PHT.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the PHT:

- 10 CFR 50.34(f)(2)(xviii);

- 10 CFR 50.55a(h);
- RG 1.20 Rev. 4;
- RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Rev. 1 (ML070260029);
- RG 1.87, Rev. 2;
- RG 1.100, Rev. 4;
- RG 1.189, "Fire Protection for Nuclear Power Plants," Rev. 5 (ML23214A287); and
- RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," Rev. 4 (ML21181A223).

PSAR sections 7.1.3 and 5.3 identify the following PDC as applicable to the PHT: 1, 2, 3, 4, 10, 11, 12, 13, 14, 15, 16, 30, 31, 32, 33, 34, 35, 36, 37, 71, 72, 74, and 78.

7.1.3.1 Technical Evaluation

7.1.3.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR table 7.1.3-6 lists the construction materials for the PSP and IHX components that form the primary pressure boundary. PSAR sections 7.1.3.2.2 and 7.1.3.3.2 state that these components are designed and constructed to ASME BPVC III-5 and are in full conformance with RG 1.87. Based on the endorsement of ASME BPVC III-5 in RG 1.87, the staff determined that the use of ASME BPVC III-5 for SR SSCs is consistent with PDC 1 for the preliminary PHT design.

The PSAR and NAT-13478 indicate that the IHX will be constructed primarily from 304H with ER 16-8-2 weld metal. The staff's evaluation of the applicant's approach to extending allowable stresses from 300,000 to 500,000 hours, as discussed in SE section 7.1.2.1.1, also applies to the IHX design and materials because it uses the same design and construction code and materials.

Table 5-2 from NAT-13478 identifies 800H plate as the material for the SR IHX tube supports. The staff notes that 800H is an approved material in ASME BPVC III-5 with allowable stresses defined for up to 300,000 hours, which does not cover the planned 60 year design lifetime of KU1. During audit discussions, the applicant noted that the IHX tube supports are not expected to experience high stresses and indicated it will appropriately justify the 800H allowable stresses for the design lifetime at the OL stage. Given the function of the IHX tube supports and the expected low stresses on this component, the staff considers that further information on the 800H allowable stresses for a 60-year design lifetime can be reasonably left for later consideration. The staff will review the applicant's approach on the IHX tube supports and the justification for 800H allowable stresses for the design lifetime during the OL application review.

ASME BPVC III-5 does not explicitly address the design of tube-to-tubesheet welds at high temperatures. In the IHX tube-to-tubesheet supplement²³ (ML25259A175), the applicant addressed the design and fabrication of these welds. The IHX tube-to-tubesheet joint will use a grooved partial penetration weld and will be analyzed as a strength weld using the rules in III-5 HBB-3000, considering the loadings defined in HBB-3110. All IHX tube-to-tubesheet welds will meet the fabrication requirements of ASME BPVC section III, division 1 (III-1), NB-4350 and the examination requirements of NB-5274, both of which are incorporated by reference into ASME BPVC III-5.

Given the absence of specific design rules in ASME BPVC III-5 for tube-to-tubesheet welds at high temperatures, the staff reviewed the applicant's proposed design and fabrication approach. The staff notes that the joint will be analyzed using ASME BPVC III-5 rules to ensure weld strength and fabricated and examined using the ASME BPVC III-1 rules to ensure weld quality. The staff also notes the low safety consequences of a leak or rupture of an IHX tube, as the IHX is a sodium-sodium heat exchanger. In the event of a leak, intermediate sodium would leak into primary sodium due to the higher pressure in the IHT. The ability to bring the reactor to a safe condition after the initiation of an intermediate-to-primary leak is demonstrated by the LBE analyses in PSAR chapter 3.

Based on the proposed use of ASME BPVC section III design rules and the low safety consequences of an IHX failure within the overall KU1 design, the staff determined that the preliminary design and fabrication approach for the IHX tube-to-tubesheet joint, as described in the IHX tube-to-tubesheet supplement, is acceptable and consistent with PDC 1.

For PSP seismic qualification, the PSAR states that the PSPs are designed to ASME QME-1-2023, while the PSP circuit breakers are designed to Institute of Electrical and Electronics Engineers (IEEE) 344-2013. The PSAR states partial conformance with RG 1.100, which endorses IEEE 344-2013 and ASME QME-1-2017. As noted above in SE section 7.1.2.1.6, the staff reviewed ASME QME-1-2023 and considers it appropriate if implemented consistent with the regulatory positions specified in Rev. 4 of RG 1.100.

Based on its review described above, the staff determined that the preliminary PHT design is consistent with PDC 1 because the codes and standards specified for the design and construction of the PHT SSCs are acceptable.

7.1.3.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.1.3.1.1 states that the PHT is designed to meet this requirement. The PHT is protected from design-basis hazards and the associated DBHLs discussed in PSAR section 6.1.1. The PSAR states the PHT is located within the SR reinforced concrete RXB substructure. This structure provides protection from external

²³ The staff notes this supplement contains the following statement: "[t]herefore, by reference, tube-to-tubesheet welds are permitted by ASME Section III, Division 5, with no explicit exclusions for partial penetration tube-to-tubesheet welds. HBB-3000 does not have specific rules for tube-to-tubesheet welds." Through audit, the staff confirmed the intended meaning of the statement to be: "[t]herefore, by reference, tube-to-tubesheet welds are addressed with respect to fabrication and examination requirements by ASME Section III, Division 5, with no explicit exclusions for partial penetration tube-to-tubesheet welds. However, HBB-3000 does not have specific design rules for tube-to-tubesheet welds."

flooding, while the PHT's location below grade provides protection from other natural phenomena including tornadoes and extreme climate conditions.

PSAR section 7.1.3.2.2 and table 7.1.3-4 classify the PSP and IHX components as SCS1, qualified to withstand seismic loads from the SSE. Seismic classifications, design, analysis, and qualification are discussed in PSAR section 6.4 and evaluated in SE section 6.4. The staff determined that the PHT design ensures safety-significant SSCs can perform their safety functions despite natural phenomena because the PHT is housed within the SR RXB substructure and its safety-significant SSCs are designed to withstand postulated seismic events. As such, the staff determined that the preliminary PHT design is consistent with PDC 2.

7.1.3.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.1.3.1.1 states that the PHT will be constructed with non-combustible and non-fire sustaining materials to the extent practical. The PSAR states the PHT is designed to minimize sodium leakage and the potential for fires. The risk of sodium-air reactions is reduced by using an inert cover gas over the primary coolant, as described in PSAR section 7.2.3. PSAR section 7.5.2.3 states that a fire protection program addressing safety-related SSCs will be included in the OL application. The staff determined that the preliminary PHT design is consistent with PDC 3 due to the use of low combustible materials, minimized sodium reaction risks, and the planned fire protection program.

7.1.3.1.4 Environmental and Dynamic Effects (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.1.3.1 states that safety-significant PHT SSCs and their materials are designed to handle sodium exposure, high temperatures, radiation, and the dynamic effects of fluids. The PHT, located in the SR RXB substructure, is protected from external hazards, including missiles.

The staff's evaluation of the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including PHT components, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. The staff notes that the RIM program is expected to appropriately address environmental effects on SSCs made of Alloy 800H in the IHX. Based on the identified R&D item for ensuring materials performance discussed in PSAR chapter 13 and the staff's evaluation of environmental effects on safety-significant, long-lived passive SSCs in contact with sodium or cover gas in SE section 7.1.2.1.4, the staff determined that the preliminary design information of PHT SSCs in contact with sodium or cover gas is consistent with PDC 4. Further information can reasonably be left for later consideration. The staff will evaluate the maturation of the RIM DMA and associated MANDE through the required R&D updates and the final design as part of its review of the OL application.

One particular area the staff will evaluate in the final RIM program for the PHT is the potential for corrosion-related degradation in the crevice area formed at the tube-to-tubesheet joint. In the IHX tube-to-tubesheet supplement, the applicant states that "if a crevice forms between the tube and tubesheet, the likelihood of degradation is not considered higher relative to other parts of the heat exchanger," citing the fact that the liquid sodium will be operating far from boiling conditions, unlike in LWR applications. The staff notes that boiling at a crevice would be unlikely

given margin to boiling; however, even in the absence of boiling, sodium fast reactor (SFR) and LWR operating experience indicates that crevice geometries and stagnant flow locations are more susceptible to off-normal chemistry conditions and should be considered as having a higher likelihood of degradation.

7.1.3.1.5 Reactivity Considerations (PDC 11 and 12)

PDC 11 requires that SSCs which contribute to reactivity feedback are designed so that the effect of prompt inherent nuclear feedback compensates for a rapid increase in reactivity. PDC 12 requires that the reactor core and associated SSCs are designed to ensure power oscillations that could result in exceeding SARRDLs are either not possible or can be detected and suppressed.

PSAR section 7.1.3.1.1 states that the PHT maintains a stable and uniform core inlet temperature, which ensures that inherent nuclear feedback characteristics of the core tend to counteract any rapid increases in reactivity in the power range. The section further states that the PSPs maintain the flow of coolant, which helps to ensure uniform temperature distribution within the core to prevent hot spots which could lead to power oscillations.

The steady-state nuclear and thermal hydraulic design analyses for the plant, described in PSAR sections 3.12 and 3.13 and provided in TP-LIC-RPT-0011, demonstrate that the net result of the feedback mechanisms is to compensate for a rapid increase in reactivity. In the preliminary transient analyses, core reactivity is strongly affected by thermal expansion and contraction of the lower core plate, which is a function of inlet plenum temperature. The PHT provides control over flow and temperature distributions in the inlet plenum. If a PSP fails, asymmetric flow into the lower plenum is likely to affect the core's transient reactivity response. However, this is a subject of further study as noted in the NRC-approved TR NAT-9390-A, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," Rev. 2 (ML25211A127). While the staff considers the preliminary analyses adequate for the CP based on the staff review of the methodology in the TR, the staff will further consider asymmetric effects through review of final analyses at the OL stage. The staff also concluded that even an asymmetric expansion of the lower core plate is expected to provide negative reactivity, though it would be less substantial than a symmetric expansion, and is therefore consistent with PDC 11. Core stability analysis is discussed in PSAR section 3.12.1.6, which states that there are no identified instabilities over the range of power evaluated; this was determined to be acceptable in the staff evaluation in SE sections 3.11.1.5 and 7.1.1.1.3. As such, the staff determined that the preliminary information on the PHT design is consistent with PDC 11 and 12.

7.1.3.1.6 PHT Instrumentation (PDC 13 and 10 CFR 50.34(f)(2)(xviii))

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. 10 CFR 50.34(f)(2)(xviii) requires instrumentation to the control room that provides an unambiguous indication of inadequate core cooling. PSAR section 7.1.3.1.1 states that PHT process parameters, including pump speed, sodium temperature, and sodium level, are monitored by instruments within the RV. These signals are processed by either the RPS or the nuclear island (NI) control system (NIC) depending on their safety classification. PSAR table 7.1.3-1 lists the monitored PHT parameters and their safety significance. These instruments are part of the reactor instrumentation system (RIS), discussed further in PSAR section 7.6.5 and evaluated in SE section 7.6.5.

The staff notes that the process parameters monitored for the PHT provide appropriate indications for determining the status of core cooling (e.g., pump speed, sodium temperatures, and pool levels). Accordingly, the staff determined that the preliminary PHT design includes appropriate instrumentation to monitor the integrity of the core, reactor coolant boundary, and functional containment during normal operations, AOOs, and accident conditions, consistent with PDC 13 and 10 CFR 50.34(f)(2)(xviii).

7.1.3.1.7 Primary Coolant Boundary (PDC 14, 15, 30, 31, and 32)

PDC 14 requires primary coolant boundary components to be designed, fabricated, erected, and tested to provide an extremely low probability of leakage, rapidly propagating failure, and gross rupture. PDC 15 requires the design conditions of the primary coolant boundary to not be exceeded during conditions of normal operation, including AOOs. PDC 30 requires primary coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 31 requires primary coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 32 requires primary coolant boundary components to permit periodic inspection and functional testing.

PSAR section 7.1.3.1.1 states that primary coolant boundary components of the PHT will meet the design, construction, testing, and quality assurance requirements of ASME BPVC III-5, which ensures that appropriate service level conditions and bounding load combinations are accounted for in the design. The design and material selection for PHT primary coolant boundary components consider service temperatures, service degradation of material properties, and applicable degradation mechanisms. Metallic primary coolant boundary components are constructed of austenitic stainless steel to minimize the effects of irradiation embrittlement and fracture from thermal stresses. The staff evaluation of the preliminary design information addressing environmental effects on safety-significant SSCs, including PHT primary coolant boundary SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Additionally, the staff evaluation of the RES primary coolant boundary components in SE section 7.1.2.1.6 also encompasses PHT primary coolant boundary components. Based on conformance with ASME BPVC III-5 and the staff evaluations in SE sections 7.1.2.1.4 and 7.1.2.1.6, the staff determined that the preliminary design of the PHT is consistent with PDC 14, 15, 30, and 31.

PSAR section 7.1.3.2.1 states that PSP shaft seal will use materials compatible with the service environment, including potential exposure to sodium aerosols. The PSP shaft will undergo EQ in accordance with ASME QME-1-2023. Based on the planned use of compatible materials and the qualification of the design and materials through the EQ program in accordance with ASME QME-1-2023, the staff determined that the preliminary design information for the PSP shaft seal is acceptable at the CP stage and consistent with PDC 14, 15, and 30.

The RIM program, described in PSAR chapter 8, is used to identify degradation mechanisms and appropriate test and MANDE inspection requirements for primary coolant boundary components. The staff evaluation of the RIM program is provided in SE section 8.1. Based on the proposed implementation of the RIM program for in-service inspection of primary coolant boundary components and the MANDE practices evaluated in SE section 7.1.2.1.6, the staff determined that the preliminary design of the PHT is consistent with PDC 32.

PSAR chapter 13 and section 12.1 of NAT-13478 describe an R&D item focused on assuring adequate structural materials performance in service. This R&D item also supports the RIM program through the development of performance monitoring methods, such as material surveillance programs, inspection methods and means of access, and monitoring approaches, as well as the development of PFM and LBB methodologies. The staff proposed two conditions for the KU1 CP related to updates on this R&D item as discussed in greater detail in SE chapter 13.

Based on the staff evaluation of materials compatibility of primary coolant boundary components in SE section 7.1.2.1.4, conformance with ASME BPVC III-5 and RG 1.87 for the design and construction of these components, and the proposed implementation of the RIM program and the associated R&D activities for assuring adequate material performance in service, the staff determined that the preliminary design of the PHT primary coolant boundary components is consistent with PDC 14, 15, 30, 31, and 32.

7.1.3.1.8 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. PSAR section 7.1.3.1.1 states that the PSP seals and certain IHX components are part of the primary coolant boundary, which retains radionuclides in the primary sodium. The PHT supports two safety functions related to functional containment, which are evaluated in SE section 7.1.3.1.17. The staff determined that PSP seals and IHX adequately provide radionuclide retention, consistent with PDC 16, because the PHT is primarily within the RES and the preliminary information regarding the design, fabrication, and testing of the PSP seal and IHX primary coolant boundary components was determined to be acceptable, as described above in SE section 7.1.3.1.7.

7.1.3.1.9 Primary Coolant Inventory (PDC 33)

PDC 33 requires a system to maintain primary coolant inventory to protect against small breaks in the primary coolant boundary as necessary to ensure SARRDLs are not exceeded. PDC 33 is addressed in PSAR sections 7.1.2.2 and 7.2.4.2.1, which describe the RES and SPS, respectively, and is evaluated in SE sections 7.1.2.1.6 and 7.2.4.1.7. Because the PHT is primarily within the RES and the primary coolant inventory maintenance function is addressed by the RES and SPS, as well as through the PDC 33 evaluations for these SSCs, the staff determined that the preliminary design of the PHT is consistent with PDC 33. The staff notes that any failure of the IHX would lead to an increase in primary sodium level.

7.1.3.1.10 Heat Removal (PDC 10 and 34)

PDC 10 requires that the reactor core and associated systems are designed with appropriate margin to assure SARRDLs are not exceeded during normal operation and AOOs. PDC 34 requires a system for removing residual decay heat removal during normal operations and AOOs. PSAR section 7.1.3.1.1 states that the PHT is designed to prevent exceeding SARRDLs by ensuring forced circulation of primary coolant during normal operations and natural circulation of primary coolant during off normal conditions. The section further states the PHT supports residual heat removal by providing primary coolant flow to remove decay heat from the reactor core and transfer it to the IHT via the IHXs. The heat in the IHT can then be rejected to atmosphere via the IACs or to the NI salt system (NSS) via the sodium-salt heat exchangers

(SHXs). The PSAR states that the PHT provides adequate heat transfer from the reactor core during any condition.

The ability of the PHT to provide adequate core cooling and remove residual heat during AOOs is demonstrated by the AOO analyses in PSAR chapter 3. All AOOs relying on heat removal from the PHT (either forced or natural circulation flow) maintain fuel within design limits and as such have no associated radiological releases. The staff determined that the preliminary PHT design is consistent with PDC 10 and 34 because it can remove heat from the reactor core during normal operation and AOOs, ensuring SARRDLs are not exceeded.

7.1.3.1.11 Emergency Core Cooling (PDC 35)

PDC 35 requires a system to ensure sufficient core cooling and residual heat removal during and after postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented. PSAR section 7.1.3.1.1 describes how the PHT supports emergency core cooling by maintaining sufficient primary sodium inventory if the PSPs and IHXs are unavailable. The PSAR states that the PHT provides passive flow to the core for heat removal via natural circulation. The heat from the primary sodium is then transferred to the RV wall and subsequently to the GV. Heat is removed from the GV via the RAC, described in PSAR section 7.2.1 and evaluated in SE section 7.2.1.

For natural circulation to develop, the PSAR states the PSPs must be secured. PSAR section 7.1.3.1.1 states that the PSPs will trip when excessive temperatures are reached in the primary sodium, establishing the conditions for natural circulation. Temperature differences between the hot and cold pools allow for natural circulation to develop, with the primary sodium flowing through the same flow path used when the PSPs are operating.

The ability of the PHT to provide adequate core cooling and remove residual heat following postulated accidents is demonstrated by the LBE analyses in PSAR chapter 3. Many analyzed events, particularly the DBAs, rely on natural circulation developing in the PHT for heat removal. The staff audited the LBE analyses and design basis heat removal calculations, confirming that natural circulation in the PHT can dissipate heat at a rate necessary to mitigate the LBEs which rely on it for emergency core cooling. As noted in SE section 7.1.3.1.17.2, there are two LBEs that rely on the PHT for cooling that result in fuel failure but maintain consequences within the frequency-consequence (F-C) target. These LBEs demonstrate acceptable results relative to PDC 35 because the fuel failure does not prevent continued effective core cooling. As such, the staff determined that the preliminary PHT design is consistent with PDC 35 because the design is capable of dissipating heat at a rate necessary to mitigate LBEs. The staff expects to review detailed design analyses at the OL stage demonstrating that the PSP coastdown characteristics promote the development of natural circulation flow and that the PSP impellers do not overly impede natural circulation.

7.1.3.1.12 Inspection and Testing of Emergency Core Cooling System (PDC 36 and 37)

PDC 36 and 37 address inspection and testing for systems required for emergency core cooling. PSAR section 7.2.1 describes the RAC, which provides emergency core cooling and relies on natural circulation developing within the PHT. Natural circulation develops within the PHT once the PSPs are secured. PSAR section 7.1.3.1.1 states that the PHT's natural circulation path is part of the RIM program, which assigns MANDE strategies to maintain reliability targets over the life of the plant. The PSAR also states that the PSP isolation circuit breakers are designed for periodic inspection and testing, including operability and functional

performance tests. The staff determined that the preliminary PHT design is consistent with PDC 36 and 37 because the PHT's natural circulation flow path is included in the RIM program to maintain reliability targets, and the PSP isolation circuit breakers can be inspected and tested. The staff's evaluation of the RIM program and considerations related to maintaining a natural circulation flowpath are provided in SE sections 7.1.2.1.11 and 8.1.1.4.

7.1.3.1.13 Sodium and Cover Gas Purity Control (PDC 71)

PDC 71 requires systems to maintain the purity of primary coolant sodium and cover gas within specified design limits. PSAR section 7.1.3.1.1 states that primary sodium purity and cover gas purity are managed by the SPS and SCG, respectively. Interfaces to these systems are provided by the RES. The SCG and SPS are discussed in PSAR sections 7.2.3 and 7.2.4 and evaluated by the staff in SE sections 7.2.3 and 7.2.4. Based on the evaluation of PDC 71 in SE sections 7.2.3 and 7.2.4, the staff determined that the preliminary PHT design is consistent with PDC 71.

7.1.3.1.14 Sodium Heating System (PDC 72)

PDC 72 requires heating systems for safety-significant SSCs that contain sodium. PSAR section 7.1.2.2 states that core power generation and core decay heat provide heating of the primary sodium, which is circulated through the PHT. This section further states that if sufficient decay is unavailable, heat is added to the primary sodium from the intermediate sodium pumps (ISPs) in the IHT through the IHX. The PSAR states that in the event of a single failure, a single PSP or intermediate sodium loop adds sufficient heat to the primary sodium to prevent freezing. The staff determined that the preliminary PHT design is consistent with PDC 72 because the primary sodium within the PHT can be maintained above freezing using heat generated from ISP or PSP operation.

7.1.3.1.15 Sodium-Water Reaction Prevention (PDC 74)

PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water. PSAR section 7.1.3.1.1 states that the PHT is located entirely within the RXB substructure, which does not house any water-filled systems to limit the potential for contact between primary sodium and water. PSAR section 7.1.3.2 describes the PSP shaft seals, which limit the leakage of air and moisture into the RV. Protection from environmental conditions that could lead to water intrusion into the RXB is described in PSAR section 7.8.1. The staff reviewed the location of the PHT, confirming its location in the RXB substructure away from any water-containing systems. As such, the staff determined the preliminary PHT design consistent with PDC 74.

7.1.3.1.16 Primary Coolant System Interfaces (PDC 78)

PDC 78 provides requirements for SSCs containing liquids that interface with the primary coolant system. It requires two passive barriers between primary coolant and a chemically incompatible fluid. For a chemically compatible fluid, one passive barrier is allowed if the postulated leakage at the interface does not result in failure of the intended safety functions of safety-significant SSCs or exceed SARRDLs, and if the fluid is maintained at a higher pressure than the primary coolant during normal operation, AOOs, shutdown, and accident conditions.

PSAR section 7.1.3.1.1 discusses the interfaces between the PHT and IHT as well as the PHT and the PSP lube oil system. This section states that the intermediate sodium is chemically

compatible with the primary coolant and that the IHT interface with the PHT consists of a single passive barrier within the IHX. The PSAR states that the intermediate loop is maintained at a higher pressure than the primary coolant during normal operation, AOOs, shutdown, and accident conditions. The PSAR further states that the RV cover gas space accommodates increased sodium inventory in the event of IHX leakage and that on a high primary coolant level condition, RPS will initiate a reactor scram and ISP trip to ensure SARRDLs are not exceeded. The ability of the PHT and IHT to maintain their safety functions during intermediate-to-primary leaks is demonstrated by the LBE analyses in PSAR chapter 3. The staff audited the LBE analyses, confirming the KU1 design is capable of performing its safety functions during events in which an intermediate sodium-primary sodium leak occurred within the IHX.

Given the Sodium Reactor Experiment (SRE) lube oil issues discussed in SE chapter 1, which resulted in flow blockages and fuel melt, the staff considers the design of lube oil systems for pumps in contact with primary sodium to be important at the CP stage. PSAR section 7.1.3.2.1 states that the PSP lube oil system supplies lubricant to the mechanical bearings of the PSPs and is designed with a double barrier between the lube oil and primary sodium. A PSP and ISP lube oil supplement (ML25259A175) further notes that the shaft seal and oil-lubricated PSP bearings are continuously monitored for leakage. This monitoring allows for early detection and mitigation of leaks before oil can come into contact with sodium. However, the final leak detection design has not been finalized.

The staff determined that the preliminary designs for the primary-intermediate coolant interface and primary coolant-PSP lube oil interface are consistent with PDC 78. The proposed single barrier between intermediate sodium and primary sodium in the IHX is acceptable because postulated leakage of intermediate sodium into the PHT does not lead to failure of any SSC safety functions or exceed SARRDLs, and because the intermediate sodium is maintained at a higher pressure than the primary sodium. For the PSP lube oil system, the presence of a double barrier between the primary sodium and the lube oil, along with continuous leak monitoring, supports meeting PDC 78 at the CP stage. The staff will evaluate the final leak detection methods employed for the PSP lube oil system during the OL application review.

7.1.3.1.17 Safety Functions

7.1.3.1.17.1 PSP Trip Functions

PSAR section 7.1.3.1.1 states that the PHT supports several trip functions to secure the PSPs. These are:

- DL3-HR2 – PSP Trip on High-High Sodium Temperature (SR)
- DL4-HR2 – PSP Trip Automatic Backup (NSRST)
- DL4-HR6 – Manual PSP Trip (NSRST)

The PSAR states that DL3-HR2 requires the PSP circuit breakers to trip on a combination of low neutron flux signal, existence of a scram signal, and cold pool temperature at the high-high temperature setpoint to reduce heat load created by pump operation and establish reasonable assurance that radionuclide release results in a radiological dose less than the 10 CFR 50.34 dose criteria at a safe shutdown condition. PSAR section 7.1.3.1.2 states that, during a reactor scram, PSP flow is reduced using a ramp down curve to a desired minimum flow. If the pump

fails to ramp down following a scram, the PSPs rely on DL3-HR2 to trip and coastdown. DL3-HR2 is automatically initiated by the RPS, described in PSAR section 7.6.3, when it receives a neutron flux signal from the nuclear instrumentation system (XIS) and cold pool temperature signals from the RIS. For all LBEs relying on DL3-HR2, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff audited the details of these LBE analyses, which confirmed the PSP trip described by DL3-HR2 is capable of preventing additional heat load caused by PSP operation for the LBEs relying on this trip. The staff determined that the preliminary PHT design is consistent with DL3-HR2 because all LBEs relying on this function do not result in radiological release.

The PSAR states that DL4-HR2 requires that the PSPs are automatically tripped on logic that is diverse from SR PSP trip function DL3-HR2 by providing an alternate means of preventing pump heat addition to the primary sodium if RPS fails to trip the PSPs. DL4-HR2 is automatically initiated by the coolant temperature monitoring and control system (CTC), described in PSAR section 7.6.2.2.1, which receives hot pool temperature signals from RIS. The staff determined that the preliminary PHT design is consistent with this function because the alternate PSP trip operates independently from RPS and relies on hot pool temperature, rather than cold pool temperature, for initiation.

The PSAR states DL4-HR6 allows for manual initiation of a PSP trip, providing an alternate means of securing the PSPs for DID adequacy. The staff determined the PHT design is consistent with this function because, as described in PSAR section 7.6.3, the RPS system supports the manual PSP trip with circuitry diverse from the SR automatic PSP trip.

7.1.3.1.17.2 Heat Removal Functions

PSAR section 7.1.3.1.1 states that the PHT supports several heat removal functions related to establishing passive heat removal via natural circulation. These functions are:

- DL3-HR1 – PSP Coastdown (SR)
- DL3-HR5 – Natural Circulation of Sodium in Primary System (SR)
- DL4-HR1 – IAC Passive Mode Operation (NSRST)

PSAR section 7.1.3.1.1 describes how natural circulation is established and maintained in the PHT, which is evaluated under PDC 35 in SE section 7.1.3.1.11. For all LBEs relying on DL3-HR1 and DL3-HR5, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff audited the details of these LBE analyses, which confirmed the PHT design is capable of dissipating heat at a rate necessary to mitigate those LBEs relying on natural circulation in the PHT. The staff determined the PHT design is consistent with these functions because fuel integrity is maintained for these LBEs

The PSAR also describes how the PHT supports DL4-HR1, which specifies that the IAC shall transfer heat from the IHT to atmosphere via natural convection airflow when the ISPs and RAC are unavailable for DID adequacy. This function also requires natural circulation to be established and maintained in the PHT, which allows for heat removal via the IHXs rather than the RAC. The staff determined that the preliminary PHT design is consistent with this function because it allows for an alternate means of heat removal in the event the RAC is unavailable. Heat removal via the IAC is discussed in further detail in SE section 7.2.2.

The staff notes that BDBEs DHP-LOOP-3 and 4 do not credit the DL3-HR1 coastdown function (as identified in PSAR table 5.2-2) and result in fuel failures. The staff identified in audit that these fuel failures occur during the PSP coastdown period because power decreases at a slower rate than flow, resulting in an increased power-to-flow ratio and thus increased fuel temperatures. Despite the fuel failures, these LBEs remain within the F-C target curve and thus demonstrate acceptable results under LMP.

7.1.3.1.17.3 Radionuclide Retention Functions

PSAR section 7.1.3.1.1 states that the PHT supports two functions which provide radionuclide retention as part of the primary coolant boundary. These are:

- DL3-RR1a – IHX Primary System Barrier (SR)
- DL3-RR1d – Primary Sodium Pump Seal (SR)

PSAR section 7.1.3.1.1 describes how PSP and IHX components form a portion of the primary coolant boundary. The primary coolant boundary supports radionuclide retention by acting as part of the KU1 functional containment.

Based on the staff's evaluation of the design and construction codes discussed in SE section 7.1.3.1.1; the material compatibility for primary coolant boundary components discussed in SE sections 7.1.2.1.1, 7.1.2.1.4, and 7.1.2.1.6; the staff's evaluation of applicable PDC in SE section 7.1.3.1.7; and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in service material performance, the staff determined that the preliminary PHT design is consistent with its role as a barrier to radionuclide transport and these functions.

7.1.3.1.18 Programmatic Special Treatments

Programmatic special treatments applied to the PHT are summarized in PSAR section 7.1.3.1.3 and include the QAP, D-RAP, EQ program, PITAP, CVAP, and RIM program. Aside from the CVAP program, the applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the PHT contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to the PHT. RG 1.20 states that the CVAP program should be applied to "unique or first-of-its-kind designs." PSAR section 7.1.3.1.1 states that the CVAP is applied to PHT SSCs within the RV in accordance with RG 1.20. Therefore, the staff determined these programs are appropriate to apply as special treatments to the PHT. Plant programs are evaluated further in SE chapter 8.

7.1.3.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.1.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.1.4 Intermediate Heat Transport System

PSAR section 7.1.4 describes the IHT. The IHT transports heat from the PHT to the NSS for power generation during normal operations, and to the IAC during reactor start-up and

shutdown. The IHT consists of two intermediate sodium loops, each connecting one IHX (which is considered part of the PHT) to an array of SHXs. Each loop includes an ISP, sodium-air heat exchangers (AHX, which is considered part of the IAC), and interconnecting piping. Expansion tanks in each loop accommodate thermal expansion of the intermediate coolant. All IHT components are NSRST, except for the ISP trip circuit breaker and trip actuator, which are SR. The IHT supports two SR functions and five NSRST functions, which are evaluated below.

PSAR section 7.1.4.1 states that the IHT loop piping and components are primarily located within the reactor auxiliary building (RAB), except for piping within the RXB connecting to the IHX nozzles and yard area piping connecting to the IAC AHXs. PSAR figure 7.1.4-1 provides a simplified diagram of an IHT intermediate loop.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the IHT:

- RG 1.22, "Periodic Testing of Protection system Actuation Functions," Rev. 0 (ML083300530);
- RG 1.36, Rev. 1; and
- RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Rev. 2 (ML22272A602).

PSAR sections 7.1.4 and 5.3 identify the following PDC as applicable to the IHT: 1, 2, 3, 4, 10, 34, 35, 70, 72, 73, 74, 75, 76, 77, and 78.

7.1.4.1 Technical Evaluation

7.1.4.1.1 Quality Standards and Records (PDC 1)

PSAR table 7.1.4.-2 identifies the design and construction codes and standards applicable to the IHT, including ASME BPVC section VIII, ASME B31.1-2022, and ASME B36.19M-2004. NAT-13478 specifies that the IHT expansion tanks, SHXs, and ISP tanks are designed and constructed to ASME BPVC VIII-1 and the IHT piping system is designed and constructed to ASME B31.1-2022

Appendix 14.2 of NAT-13478 identifies the materials, design and construction codes and standards, and special treatments for major NSRST SSCs in the Sodium design, including in the IHT. The following bullets summarize the design and construction special treatments identified for NSRST SSCs in the IHT designed to ASME BPVC VIII-1, excluding the SHXs. The staff notes there is greater uncertainty regarding design and construction of the SHXs due to their lower design maturity as discussed in SE section 13.1.1.4 on the associated R&D item.

- For the IHT expansion tanks and ISP tanks, the applicant stated that the ASME BPVC VIII-1 rules will be supplemented with additional analysis to account for fatigue using the fatigue evaluation procedure from ASME BPVC Section VIII, Division 2 (VIII-2).
- For the IHT expansion tank, because ASME BPVC section VIII does not explicitly account for creep effects in long-term service, components subject to non-negligible creep, as determined by screening per ASME BPVC III-5, Mandatory Appendix HCB-III,

will be evaluated using the Class A rules in ASME BPVC III-5 or ASME code case 2843-3. The staff notes that ASME code case 2843-3 provides elevated temperature rules that are similar to ASME BPVC III-5 Class A rules, including fatigue and creep-fatigue assessment.

- For the IHT expansion tank, the allowable stresses for elevated temperature are limited to ASME BPVC III-5, extended to 500,000 hours service time (see SE section 7.1.2.1.1) and the delta ferrite limits for weld material in service above 800°F will meet the requirements in ASME BPVC III-5, HCB-2433, and RG 1.31.
- For the ISP tank, the delta ferrite limits for weld material will meet the requirements in ASME BPVC III-1, NC-2433, and RG 1.31.
- All pressure boundary welds will be full penetration welds, with 100% volumetric examination to confirm weld quality. To provide assurance of material properties, all pressure boundary materials will require a CMTR.
- ASME BPVC VIII-1 rules for lethal service will be applied for liquid sodium-containing components designed to ASME BPVC VIII-1.

The following bullets summarize the design and construction special treatments identified for NSRST SSCs in the IHT designed to ASME B31.1-2022, which includes IHT piping and valves:

- The ASME B31.1-2022 construction rules will be supplemented with additional design rules in ASME BPVC III-5 for Class B piping to account for creep. For piping determined to have non-negligible creep effects, an analysis per ASME BPVC III-5 subsection HCB-3634(b) will be applied. The staff notes that subsection HCB-3634(b) provides more conservative allowable stresses for thermal expansion stresses and stress criteria for thermal cycling to account for creep effects than ASME B31.1-2022.
- The allowable stresses from ASME BPVC III-5, extended to 500,000 hours service time, will be used for the design analysis (see SE section 7.1.2.1.1) and the delta ferrite limits for weld material will meet the requirements in ASME BPVC III-5, HCB-2433, and RG 1.31.
- All pressure boundary welds will be full penetration welds, with 100% volumetric examination to confirm weld quality. To provide assurance of material properties, all pressure boundary material will require a CMTR.
- ASME B31.1-2022 toxic fluid design and construction rules will be applied to liquid sodium-containing components designed to ASME B31.1-2022.

The staff notes that the special treatments will provide increased assurance of appropriate design margin and fabrication quality through more rigorous design considering the effects of creep and fatigue and improved weld design and examination to ensure weld quality. Based on the special treatments identified above related to design and construction, the staff determined that the preliminary design of IHT SSCs classified as NSRST and constructed to ASME BPVC VIII-1 and ASME B31.1-2022 (other than the SHXs) is consistent with PDC 1. The staff will review the final design of these SSCs at the OL stage alongside plans for in-service

performance monitoring to ensure design and monitoring provide reasonable confidence that these SSCs will perform their safety functions over their lifetime.

ASME B36-19M-2004 covers the standardization of dimensions of welded and seamless wrought stainless-steel pipe and is referenced in ASME B31.1-2022 for standard piping components. The staff determined that this standard is appropriate since it is referenced in ASME B31.1-2022, which the staff determined is acceptable in SE section 6.4.1.5 and as discussed above.

The PSAR states that the IHT adheres to IEEE/International Electrotechnical Commission (IEC) 60780-323-2016, which is used for SSCs located in a harsh environment. The staff determined that this is consistent with RG 1.89. The PSAR states that IHT adheres to IEEE 308-2001. The staff determined that this is consistent with RG 1.32, which endorses the use of IEEE 308-2001. The PSAR additionally states that applicable IHT SSCs are designed to IEEE 603-1991. The staff determined that this is consistent with RG 1.153, "Criteria for Safety Systems," Rev. 1 (ML003740022), which endorses the use of IEEE 603-1991.

Based on its review described above, the staff determined that the preliminary IHT design is consistent with PDC 1 because the codes and standards specified for the design and construction of the IHT SSCs (other than the SHXs) are acceptable.

7.1.4.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.1.4.2 states that the IHT is designed to meet this requirement. IHT components are primarily within the NSRST RAB, while SR IHT components supporting the ISP trip are in the SR reinforced concrete NI control building (NCB) substructure. The NCB substructure provides protection from external flooding, while the SR IHT SSCs' location below grade provides protection from other natural phenomena including tornadoes and extreme climate conditions. Additionally, some IHT piping is within the SR RXB substructure.

PSAR table 7.1.4-1 classifies the SR ISP trip circuit breaker and ISP trip actuator as SCS1, qualified to withstand seismic loads associated with the SSE. Other IHT SSCs are NSRST and are classified as SCN1, with seismic special treatments ensure risk-significant functions or DID can be performed. Seismic classifications, design, analysis, and qualification are detailed in PSAR section 6.4 and evaluated in SE section 6.4.

The staff determined that the IHT design ensures safety-significant SSCs can perform their safety functions despite natural phenomena because NSRST components are primarily housed within the NSRST RAB, while SR components are within the SR NCB substructure, and are designed to withstand seismic events. As such, the staff determined that the preliminary IHT design is consistent with PDC 2.

7.1.4.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.1.4.2 states that the IHT will be constructed with non-combustible and non-fire sustaining materials to the extent practical. It also includes containment features in the IHT piping to detect leakage and minimize fire risks from sodium interactions. The staff evaluation of the applicant's approach to fire protection for the IHT is in

SE sections 7.5.2 and 8.3. Based on review of the information in PSAR section 7.1.4.2 and the evaluation in SE sections 7.5.2 and 8.3, the staff determined the preliminary IHT design is consistent with PDC 3.

7.1.4.1.4 Environmental and Dynamic Effects (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.1.4.2 states that sodium-containing IHT SSCs are designed to handle high temperatures and sodium exposure. Additionally, IHT SSCs in contact with salt are designed with consideration for material compatibility with the molten salt environment. The IHT is protected from certain dynamic hazards due to its SSCs being located in the RXB substructure, NSB substructure, and RAB, including missiles and discharging fluids. As such, the staff determined that the preliminary IHT design is adequately protected from external hazards and designed to accommodate environmental conditions associated with normal operations and postulated accidents, consistent with PDC 4.

The staff evaluation of the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including IHT components, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Based on the identified R&D item for ensuring materials performance and the staff's evaluation of environmental effects on safety-significant, long-lived passive SSCs in contact with sodium or cover gas in SE section 7.1.2.1.4, the staff determined that the preliminary design information of IHT SSCs in contact with sodium or cover gas (other than the SHXs) is consistent with PDC 4. Further information can reasonably be left for later consideration. The staff will evaluate the maturation of the RIM DMA and associated MANDE through the prescribed R&D updates as well as with the final design in the OL application.

One SSC within the IHT system that requires further evaluation at the OL stage is the SHXs. The SHXs facilitate the transfer of heat from the IHT sodium to the salt system and operate in both sodium and salt environments. The sodium-salt reaction and its impact on the design of the SHXs is identified as an R&D item. The staff evaluation of the SHX R&D activities, including materials performance considerations in the salt environment, is provided in SE section 13.1.1.4.

7.1.4.1.5 Heat Removal (PDC 10 and 34)

PDC 10 and 34 address adequate heat removal during normal operation and AOOs. PSAR section 7.1.4.3 states that the IHT transports heat from the IHXs to the NSS via the SHXs during normal operation. The section further states that the IHT supports heat rejection to the atmosphere by circulating intermediate sodium through the IAC AHXs during low power operating modes and reactor shutdown conditions. Three different IAC operating modes are described in detail in PSAR section 7.2.2 and evaluated in SE section 7.2.2. PSAR section 7.1.4.3 states that the IHT can use either forced flow or natural circulation of intermediate sodium to reject heat via the IAC AHXs. The PSAR states that natural circulation develops in the IHT during a loss of the ISP due to the elevation difference between the IHX and AHX, which allows heated, low-density sodium from the IHX to flow to the elevated AHX, where the intermediate sodium is cooled and becomes denser. The PSAR further states that the IHT design allows for sufficient heat removal from the core with natural circulation in both intermediate loops via the IAC without reliance on the RAC.

The AOO analyses in PSAR chapter 3 demonstrate the IAC's ability to provide adequate core cooling and remove residual heat during AOOs. All AOOs relying on heat removal from IAC (either forced or natural circulation flow) maintain fuel within design limits and as such have no associated radiological releases. The staff audited the AOO analyses and associated design basis heat removal calculations, and confirmed that the IHT design can dissipate heat at a rate necessary to mitigate AOOs relying on the IHT for heat removal. As such, the staff determined that the preliminary IHT design is consistent with PDC 10 and 34 because it can remove heat from the reactor core during normal operation and AOOs, ensuring SARRDLs are not exceeded.

7.1.4.1.6 Emergency Core Cooling (PDC 35)

PDC 35 requires sufficient core cooling during postulated accidents, which is accomplished by the RAC described in PSAR section 7.2.1. PSAR section 7.1.4.2 states that the IHT supports emergency core cooling by providing emergency shutdown of the ISPs to ensure the RAC heat removal capacity is not exceeded during postulated accidents. The section further states the ISP emergency shutdown design includes two circuit breakers in series for protection against single failure of a breaker to trip. The PSAR states that the design includes multiple automatic shutdown signals and the ability to manually secure the ISPs. These are captured by safety functions DL3-HR3, DL4-HR3, and DL4-HR7, which are evaluated below.

The ability of the emergency shutdown of the ISPs to adequately limit additional heat load on the RAC following postulated accidents is demonstrated by the LBE analyses in chapter 3 of the PSAR. The staff audited the LBE analyses and design basis heat removal calculations, confirming that the RAC can dissipate heat following the trip of the ISPs at a rate necessary to mitigate the LBEs in which it is credited. As such, the staff determined that the preliminary IHT design is consistent with PDC 35 because the design is capable of handling the heat load for the applicable events.

7.1.4.1.7 Intermediate Coolant Boundary (PDC 70, 75, 76, and 77)

PDC 70 states that the intermediate coolant system shall be designed with sufficient margin to ensure that the intermediate coolant boundary maintains integrity during normal operation and AOOs, and that the integrity of the primary coolant boundary shall be maintained during postulated accidents. PDC 75 requires intermediate coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 76 requires intermediate coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 77 requires intermediate coolant boundary components to permit periodic inspection and functional testing.

PSAR section 7.1.4.2 states that intermediate coolant boundary components of the IHT will be designed to ensure that design conditions are not exceeded during normal operation, including AOOs. NAT-13478 identifies that the IHT piping will be designed in accordance with ASME B31.1-2022, while IHT expansion tanks, SHXs, and ISP tanks will be designed to ASME BPVC VIII-1. The staff evaluated the use of industrial codes including ASME B31.1-2022 and ASME BPVC VIII-1 in SE section 6.4.1.5. The applicant identified special treatments to augment these industrial code requirements as described in SE section 7.1.4.1.1 to provide additional assurance of SSC quality and reliability. Based on the staff evaluation in SE sections 6.4.1.5 and 7.1.4.1.1, the staff determined that the preliminary design of IHT intermediate coolant boundary components is consistent with PDC 70 and 75.

The PSAR states that overpressure protection is provided via the SCG, which is described in PSAR section 7.2.3 and evaluated in SE section 7.2.3. The PSAR also states that materials for the intermediate pressure boundary are selected with consideration of environmental effects to ensure non-brittle fracture behavior. PSAR section 5.3.6.8 states that the RIM program and the DMA will evaluate the need for a materials surveillance program for the intermediate coolant boundary.

In NAT-13478, the applicant identifies leakage monitoring as a likely MANDE approach for the intermediate coolant boundary. In addition to the considerations described in SE section 7.1.2.1.6 regarding the use of PFM and leakage monitoring for the RES, which involves justification and a validated technical basis, the staff notes that the preliminary design of IHT piping and vessels does not include inerted guard enclosures or piping, unlike SSCs containing primary sodium. As a result, the reliability and effectiveness of sodium leak detection, and the potential for sodium fires (evaluated in SE sections 7.5.2 and 8.3), are more significant considerations for IHT components. PSAR section 8.2 states that leaks will be postulated from all sodium-containing piping based on a methodology to be developed for lower stress piping. The staff will review the approach to sodium leak postulation and leakage monitoring (including LBB and PFM analyses) during the OL review of the final IHT design with respect to materials performance and sodium fire (which ties to PDC 3 and 73).

The staff evaluation of the preliminary design information addressing environmental effects on safety-significant SSCs, including IHT intermediate coolant boundary components, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. The staff evaluation of the RIM program is documented in SE section 8.1. Based on the identified R&D item for ensuring materials performance discussed in SE chapter 13, the staff determined that the preliminary design information for the IHT intermediate coolant boundary components (excluding the SHXs) is consistent with PDC 76 and 77. The staff will evaluate the maturation of the RIM DMA and associated MANDE through the prescribed R&D updates as well as with the final design in the OL application.

As discussed in SE section 7.1.4.1.4, one SSC within the IHT system that requires further evaluation at the OL stage is the SHXs. The SHXs facilitate the transfer of heat from the IHT sodium to the salt system and operate in both sodium and salt environments. The sodium-salt reaction and its impact on SHX design is identified as an R&D item. The staff evaluation of SHX R&D activities, including materials performance in the salt environment, is provided in SE section 13.1.1.4.

7.1.4.1.8 Sodium Heating System (PDC 72)

PDC 72 requires heating systems for safety-significant SSCs that contain sodium. PSAR section 5.3.6.3 states that during normal operation, decay heat prevents intermediate sodium from freezing. The section further states that if decay heat is insufficient, ISP operation adds sufficient heat to the intermediate sodium to prevent freezing. PSAR section 7.1.4.2 states the ISPs can also add heat to the primary sodium via the IHXs to prevent freezing in primary sodium. The staff determined that the preliminary IHT design is consistent with PDC 72 because the IHT is designed to maintain the intermediate sodium above freezing.

7.1.4.1.9 Sodium Leakage Detection and Reaction Prevention and Mitigation (PDC 73)

PDC 73 requires a means to detect sodium leakage and to limit and control sodium-air and sodium-concrete reactions. PSAR section 7.1.4.2 states that the design of the IHT provides means to detect sodium leakage, limit and control sodium-air and sodium-concrete reactions, and mitigate effects of fires from sodium-air reactions. PSAR section 7.5.2 discusses sodium leakage detection, collection, and fire mitigation and PSAR section 8.2 describes the sodium fire protection design. The staff's evaluation of PDC 73 and the preliminary design for sodium leakage detection and reaction prevention and mitigation, which includes the IHT, is in SE sections 7.5.2 and 8.3. Based on the evaluation of PDC 73 in SE sections 7.5.2 and 8.3, the staff determined that the preliminary design of the IHT is consistent with PDC 73.

7.1.4.1.10 Sodium-Water Reaction Prevention (PDC 74)

PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water. PSAR section 8.2.1 states that there are no water-containing systems within the RXB and RAB due to the presence of sodium in these buildings. PSAR section 7.1.4.2 states that IHT piping in the yard area is protected to prevent exposure of potential sodium leakage to water, including precipitation. Based on the lack of water sources in the RXB and RAB and protection of IHT piping in the yard area, the staff finds the preliminary IHT design is consistent with PDC 74.

7.1.4.1.11 Primary-Intermediate Coolant Interface (PDC 78)

PDC 78 provides requirements for SSCs containing liquids that interface with the primary coolant system. PSAR section 7.1.4.2 states that the intermediate sodium in the IHT interfaces with the primary sodium in the PHT within the IHX. The staff evaluated this interface against PDC 78 in SE section 7.1.3. Based on the evaluation of PDC 78 in SE section 7.1.3, the staff determined that the preliminary design of the IHT is consistent with PDC 78.

7.1.4.1.12 Safety Functions

7.1.4.1.12.1 ISP Pump Trips

PSAR section 7.1.4.2 states that the IHT supports several trip functions to secure the ISPs. These functions are:

- DL3-HR3 – ISP Trip on High-High Primary Sodium Temperature (SR)
- DL3-HR12 – ISP Trip on High-High Primary Sodium Level (SR)
- DL4-HR3 – ISP Trip Automatic Backup (NSRST)
- DL2-HR2 – ISP Trip on Low IHT Level (NSRST)
- DL4-HR7 – Manual ISP Trip (NSRST)

The PSAR states that DL3-HR3 requires the ISP circuit breakers to trip on a combination of low neutron flux signal, existence of a scram signal, and cold pool temperature at the high-high temperature setpoint to reduce heat load created by pump operation and establish reasonable assurance that radionuclide release results in a radiological dose less than the 10 CFR 50.34 dose criteria at a safe shutdown condition. DL3-HR3 is automatically initiated by RPS, described

in PSAR section 7.6.3, which receives neutron flux signals from XIS and cold pool temperature signals from RIS. For all LBEs relying on DL3-HR3, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff audited the details of these LBE analyses and confirmed the ISP trip described by DL3-HR3 is capable of preventing additional heat load caused by ISP operation for the LBEs relying on this trip. The staff determined that the preliminary IHT design is consistent with DL3-HR3 because all LBEs relying on this function do not result in radiological release.

The PSAR states that DL4-HR3 requires that the ISPs are automatically tripped on logic that is diverse from SR ISP trip function DL3-HR3 by providing an alternate means of preventing pump heat addition to the primary sodium if RPS fails to trip the ISPs. DL4-HR3 is automatically initiated by the CTC, described in PSAR section 7.6.2.2.1, which receives hot pool temperature signals from RIS. The staff determined that the preliminary IHT design is consistent with this function because the alternate ISP trip operates independently from RPS and relies on hot pool temperature, rather than cold pool temperature, for initiation.

The PSAR states that DL3-HR12 requires the ISP circuit breakers to trip on a combination of low neutron flux signal, existence of a scram signal, and hot pool level at the high-high primary sodium level setpoint to prevent over-pressurization of the primary boundary, establish reasonable assurance primary inventory is maintained, and establish reasonable assurance that radionuclide release results in a radiological dose less than the 10 CFR 50.34 dose criteria at a safe shutdown condition. DL3-HR12 is automatically initiated by RPS, which receives neutron flux signals from XIS and a hot pool level signal from RIS. For all LBEs relying on DL3-HR12, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff audited the details of these LBE analyses and confirmed that the ISP trip described by DL3-HR12 is capable of preventing over-pressurization of the primary boundary and maintaining primary sodium inventory. The staff determined that the preliminary IHT design is consistent with DL3-HR12 because LBEs relying on this function do not result in radiological release.

The PSAR states that DL2-HR2 requires that the ISPs are automatically tripped on a low IHT expansion tank level to limit loss of intermediate sodium. DL2-HR2 is automatically initiated by the CTC, which receives IHT expansion tank level signals from the IHT. PSAR section 7.1.4.2 states that, for DL2-HR2, only the ISP on the same intermediate loop as the expansion tank with the low level is tripped. This section further states that this trip provides protection against RV overfill in the event of leakage within the IHX. DL2-HR2 provides DID to SR function DL3-HR12, which also provides protection against RV overfill. The staff determined that the preliminary IHT design is consistent with DL2-HR2, providing adequate DID for DL3-HR12, because this ISP trip operates independently from RPS and relies on IHT expansion tank level, rather than primary sodium level, for initiation.

The PSAR states DL4-HR7 allows for manual initiation of an ISP trip, providing an alternate means of securing the ISPs for DID adequacy. The staff determined the IHT design is consistent with this function because, as stated in PSAR section 7.6.3, the RPS system supports the manual PSP trip with circuitry diverse from the SR automatic PSP trip.

7.1.4.1.12.2 Heat Removal Function

PSAR section 7.1.4.2 states that the IHT supports NSRST function DL4-HR1, "IAC Passive Mode Operation," which requires that the IAC transfers heat from the IHT to the atmosphere via natural convection airflow for LBEs when the ISPs and RAC are unavailable. PSAR table 5.2-4 states that DL4-HR1 is required for DID adequacy. This function requires natural circulation to

be established and maintained in the IHT, which allows for heat removal from the PHT via the IHXs rather than the RAC. Heat removal via the IAC is discussed in further detail in SE section 7.2.2. The staff audited the details of these LBE analyses and confirmed the IHT design can dissipate heat at a sufficient rate to mitigate the LBEs relying on passive operation of the IAC with the ISPs and RAC unavailable. The staff determined that the preliminary IHT design is consistent with this function because it allows for alternate means of heat removal in the event the ISPs and RAC are unavailable.

7.1.4.1.12.3 Intermediate Leak Guard Piping Function

PSAR section 7.1.4.2 states the IHT supports NSRST function DL4-DID1, "Intermediate Leak Guard Piping," which requires sodium leakage detection and containment for IHT piping located within the RXB. PSAR section 7.1.4.3 describes the leak guard discussed in DL4-DID1 that encloses IHT piping within the RXB HAA as a secondary barrier for leak protection. The PSAR states that the leak guard contains and redirects sodium towards dedicated drains and containment features to prevent interaction with SR equipment within the HAA. PSAR section 8.2.1 provides additional description of the clamshell guard enclosure planned for IHT SSCs in the HAA. The staff evaluation of the clamshell guard enclosure design is provided in SE sections 7.5.2 and 8.3. Based on these evaluations, the staff determined that the preliminary IHT design is consistent with this function because the design provides a secondary barrier on the IHT piping in the HAA.

7.1.4.1.13 Programmatic Special Treatments

Programmatic special treatments applied to the IHT are summarized in PSAR section 7.1.4.4 and include the QAP, D-RAP, EQ program, PITAP, and RIM program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the IHT contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to the IHT. Plant programs are evaluated further in SE chapter 8.

7.1.4.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.1.4 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.2 Reactor Auxiliary Systems

PSAR section 7.2 describes the reactor auxiliary system SSCs, which are responsible for decay heat removal and support for key safety functions. These include the following major systems:

- RAC
- IAC
- SCG
- SPS
- CRD

7.2.1 Reactor Air Cooling System

PSAR section 7.2.1 describes the RAC. The RAC supports passive decay heat removal from the reactor through natural convection of air, which enables heat rejection to the atmosphere. The RAC operates continuously without manual or automatic actuation, allowing for long-term emergency core cooling. It includes four inlet and four outlet stacks, the CCA, and associated ducting, all of which are housed within the RXB. The CCA surrounds the GV, separating the RAC inlet and outlet air flow paths. The RAC contains instrumentation to measure inlet and outlet air temperature and outlet mass flow rate to assess heat removal performance. All RAC components are SR, except for its instrumentation, which is NSRST. The RAC supports two safety functions, which are evaluated below.

PSAR section 7.2.1.1 describes how the RAC cools the core. Reactor core heat is transferred to the primary coolant as described in PSAR section 7.1.3.1, and then from the coolant through the RV to the GV as described in PSAR section 7.1.2.3. This results in an elevated temperature on the outer surface of the GV. The CCA is heated by thermal radiation from the outer surface of the GV. As air flows through the riser annulus, it is heated via convection from both the outer surface of the GV and the inner surface of the CCA. The heated air's density decreases, causing it to rise. This draws ambient air into the system through the four inlet stacks and into a common inlet plenum within the HAA. The cool air flows through the downcomer annulus and around the bottom of the CCA. This air then flows into the riser annulus where it is heated. Heated air enters the four outlet stacks via common header ducts within the modular isolated reactor support structure (MIRSS), described in PSAR section 7.1.2.3. The heated air then exits the RAC through the outlet stacks into the atmosphere. PSAR figure 7.2.1-1 illustrates this flow path.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the RAC:

- RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Rev. 1 (ML070360253);
- RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Rev. 5 (ML18136A762);
- RG 1.100, Rev. 4;
- RG 1.243, "Safety-Related Steel Structures and Steel-Plated Composite Walls for other than Reactor Vessels and Containments," Rev. 0 (ML21089A032); and
- RG 1.140, Rev. 3.

PSAR sections 7.2.1 and 5.3 identify the following PDC as applicable to the RAC: 1, 2, 3, 4, 13, 35, 36, 37, and 64.

7.2.1.1 Technical Evaluation

7.2.1.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR table 7.2.1-2 identifies the design and construction codes and standards applicable to the RAC, including American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-18, ASME AG-1-2019, and ASME STS-1-2021. PSAR section 7.2.1.3 states the stacks, louvers, cowls, and CCA are designed to meet the requirements of ANSI/AISC N690-18 and that the RAC is in full conformance with RG 1.243, which endorses the use of ANSI/AISC N690-18 for SR steel structures other than RVs and containments.

The PSAR states that the RAC inlet and outlet stacks are additionally designed to meet ASME STS-1-2021 and that the RAC inlet and outlet ducting, louvers, and cowls are designed to meet ASME AG-1-2019. While the NRC has not endorsed ASME STS-1-2021, it is an applicable consensus standard for the design of steel stacks like those used in the RAC. ASME AG-1-2019 generally covers the design of filters, ventilation, and process gas treatment. A previous edition and addenda, ASME AG-1-2009 (including 2010 Addenda 1a and 2011 Addenda 1b), is endorsed by the NRC in RG 1.140 for the design, inspection, and testing of normal atmosphere cleanup systems for controlling releases of airborne radioactive materials to the environment in light water reactors. The staff notes that air flowing through the RAC will be subject to irradiation from the reactor, which can result in some activation of nitrogen and argon in the air. The staff reviewed ASME AG-1-2019 and considers it acceptable for design and construction of the RAC if implemented consistent with the regulatory positions specified in RG 1.140.

For seismic qualification, PSAR section 7.2.1.2 states that the RAC instrumentation is designed to IEEE 344-2020. The PSAR states partial conformance with RG 1.100, which endorses IEEE 344-2013. The staff notes that the NRC has not yet endorsed IEEE 344-2020. However, the staff reviewed IEEE 344-2020 and considers it appropriate if implemented consistent with the regulatory positions specified in Rev. 4 of RG 1.100. For post-accident monitoring (PAM), the RAC contains Type B and D variables adhering to IEEE 497-2016 and is in full conformance with RG 1.97, which endorses the use of IEEE 497-2016.

Based on its review described above, the staff determined that the preliminary RAC design is consistent with PDC 1 because the codes and standards specified for the design and construction of the RAC SSCs are acceptable.

7.2.1.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.2.1.2 states that the RAC is designed to meet this requirement and is protected from design-basis hazards and DBHLs discussed in PSAR section 6.1.1. Much of the RAC is located within the SR reinforced concrete RXB substructure. This structure provides protection from external flooding, while the below grade location of the RAC SSCs within the RXB substructure provides protection from other natural phenomena including tornadoes and extreme climate conditions. The upper portions of the SR RAC stack, outside the RXB, are designed to withstand tornadoes and extreme climates without losing functionality. PSAR section 7.2.1.3 describes the RAC inlet and outlet stack designs, including louver assemblies and chimney cowls to reduce precipitation and debris ingress. The RAC is designed with sufficient capacity to tolerate dust, sand, and other blockages without significant performance degradation and adheres to ANSI/AISC N690-18 and ASME STS-1-2021.

PSAR section 7.2.1.2 classifies RAC components as SCS1, except for RAC instrumentation, which is SCN3. The RAC is designed to withstand seismic loads from the SSE without affecting the ability to perform its safety functions. Seismic classifications, design, analysis, and qualification are discussed in PSAR section 6.4 and evaluated in SE section 6.4.1.1.

The staff determined that the preliminary RAC design is consistent with PDC 2 because the RAC components within the RXB substructure are protected from design basis external hazards, and the RAC SSCs outside the RXB are designed to withstand the most limiting natural phenomena without affecting the ability to perform the RAC's SR functions.

7.2.1.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.2.1.2 states that the RAC will be designed and located to minimize these risks and constructed with non-combustible and non-fire sustaining materials where practical. Major RAC components are made of metallic materials, as specified in PSAR table 7.2.1-3. PSAR section 7.5.2.3 states that a fire protection program for safety-related SSCs will be included in the OL application. The staff determined that the preliminary RAC design is consistent with PDC 3 due to the use of non-combustible materials, physical separation to minimize fire impact, and the planned fire protection program.

7.2.1.1.4 Environmental and Dynamic Effects (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.2.1.2 states that the RAC is designed to handle climate conditions including wind pressure drops and expected air temperature and humidity ranges. RAC SSCs in the RXB substructure are protected from external hazards, including missiles. The RAC fully conforms with RG 1.76, with its stacks designed to withstand wind generated missiles. Based on the information provided in PSAR table 7.2.1-3, the staff notes that the RAC operates in mild environmental conditions (temperature range and chemistry) for the preliminary materials of construction. As such, the staff determined that the preliminary RAC design is adequately protected from external hazards and can accommodate environmental conditions associated with normal operations and postulated accidents, consistent with PDC 4.

7.2.1.1.5 RAC Instrumentation (PDC 13)

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and for accident conditions. PSAR section 7.2.1.2 states that RAC inlet temperature, outlet temperature, and outlet flow rate are monitored to assess system performance. The PSAR states that the RAC is in full compliance with RG 1.97 and contains Type B and D variables with redundant instrumentation adhering to IEEE 497-2016. The staff determined that the preliminary RAC design is consistent with PDC 13 because monitoring air flow rate, inlet, and outlet temperatures allow for continually accessing heat removal efficacy of the RAC. Additionally, the staff determined that the PSAR information regarding the PAM system is sufficient because it indicates full compliance with RG 1.97 and follows IEEE 497-2016.

7.2.1.1.6 Emergency Core Cooling (PDC 35)

PDC 35 requires sufficient core cooling during postulated accidents. PSAR section 7.2.1.3 describes how the RAC removes heat from the PHT following a loss of forced flow in the PHT and IHT. In this scenario, natural convection develops in the PHT, transferring heat to the RV wall. Heat is then transferred from the RV wall to the GV via thermal radiation, causing the GV temperature to rise. As the GV temperature rises, more heat is transferred to the CCA via thermal radiation, causing the CCA temperature to rise. Increased wall temperatures for both the GV and CCA enhance heat transfer to air via convection in the riser annulus, increasing the air's temperature and flow rate, leading to greater heat rejection to the atmosphere. PSAR section 7.2.1.3 states that the RAC inlet and outlet stacks, ducting, and other portions of the airflow path are designed to provide sufficient airflow to remove the design basis heat load under the limiting environmental conditions specified in PSAR section 2.1.

The RAC flow path includes design features to ensure airflow is maintained. PSAR section 7.2.1.1 states that the inner ducts are insulated and that the CCA contains insulation between its inner and outer cylinders, both of which limit the heating of incoming air to minimize the effect on natural circulation. The CCA also prevents bypassing between the inlet and outlet flow paths. PSAR section 7.1.2 describes the HAA seal and well seal, both SR, which minimize air leakage to the HAA and maintain the integrity of the RAC airflow path. The HAA and well seal are evaluated in SE section 7.1.2.

The PSAR also states that common air regions, such as the area in the vicinity of the GV, are open and interconnected to compensate for potential blockages. As previously discussed, the RAC includes four separate inlet stacks that meet at a common header at the inlet to the downcomer annulus. From there, air flows through a single common flow path into the riser annulus, which then splits into four separate paths after exiting the vicinity of the reactor. These paths lead to the four outlet stacks. The staff audited applicant calculations which confirmed the RAC's capability to maintain adequate performance even with significant blockages.

The staff is aware of experience at Argonne National Laboratory Natural Convection Shutdown Heat Removal Test Facility (NSTF), which showed that wind conditions can significantly impact natural convection flow in natural convection heat removal systems, resulting in flow reversal and substantially reduced performance (Lisowski, Darius D., et al.). The staff audited analyses regarding RAC performance in adverse wind conditions and confirmed that RAC performance is maintained, largely due to the design and orientation of the inlet louver assemblies and the design of the chimney cowl.

The ability of the RAC to provide adequate core cooling and remove residual heat following postulated accidents is demonstrated by the LBE analyses in chapter 3 of the PSAR. Many of the analyzed events, particularly DBAs, rely on the RAC for heat removal. The staff audited the LBE analyses and design basis heat removal calculations, confirming that the RAC design can dissipate heat at sufficient rate to mitigate the LBEs relying on the RAC for emergency core cooling. The staff determined that the preliminary RAC design is consistent with PDC 35 because it provides adequate core cooling capability for applicable events. The staff notes that RAC heat transfer performance is the subject of an R&D item as discussed in SE chapter 13; the staff will confirm RAC performance at the OL stage.

7.2.1.1.7 Inspection and Testing of Emergency Core Cooling System (PDC 36 and 37)

PDC 36 and 37 address inspection and testing for systems required for emergency core cooling. PSAR section 7.2.1.2 states that RAC ducts, stacks, and the riser annulus between the GV and the CCA are accessible for personnel inspection. Areas inaccessible to personnel, such

as the downcomer annulus, are accessible for remote inspection and debris removal. Inspections of the RAC flow passages are performed under the RIM program to ensure integrity and capability. Additionally, the PSAR states that the RAC's instrumentation allows for continuous monitoring of air temperature and flow, ensuring proper function during normal operation. The staff determined that the preliminary RAC design is consistent with PDC 36 and 37 because it allows for periodic inspections and continuous monitoring to ensure the system's integrity and capability to perform its SR function.

7.2.1.1.8 Monitoring Radioactivity Releases (PDC 64)

PDC 64 requires monitoring of effluent discharge paths for radioactivity that may be released from normal operations and postulated accidents. PSAR section 7.2.1.1 states that means are provided for monitoring radioactivity releases from the RAC, specifically identifying N-16 and Ar-41 as gaseous effluents. PSAR section 9.1 states that an Offsite Dose Calculation Manual (ODCM) will be provided at the OL stage and will include details on effluent and environmental radiation monitoring. PSAR section 9.1.1 confirms expected releases of N-16 and Ar-41 from the RAC. During the audit, the staff requested details on how the applicant plans to address RAC monitoring requirements. The applicant clarified information on the ODCM in PSAR section 9.1, indicating that releases from the RAC will be quantified by periodic sampling of radionuclide concentrations. The details and sampling methods will be provided in the ODCM at the OL stage.

The staff determined that the preliminary RAC design is consistent with PDC 64 because PSAR section 7.2.1.1 outlines plans to monitor gaseous effluents for N-16 and Ar-41 with additional details being provided in the ODCM with the OL application as specified in PSAR section 9.1.

7.2.1.1.9 Safety Functions

7.2.1.1.9.1 Heat Removal Function

PSAR section 7.2.1.2 states the RAC has one SR function, DL3-HR4, "Inherent – RAC Operation," which specifies that the RAC continuously transfers heat to the atmosphere via natural circulation during accident conditions. PSAR section 7.2.1.3 describes how the RAC removes heat from the PHT, which is evaluated under PDC 35. For all LBEs relying on DL3-HR4, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff audited these LBE analyses and confirmed that the RAC design can dissipate heat at rate sufficient to mitigate the LBEs relying on the RAC for emergency core cooling. The staff determined that the preliminary RAC design is consistent with DL3-HR4 because all LBEs relying on RAC heat removal maintain fuel integrity and do not result in radiological release.

7.2.1.1.9.2 PAM Function

PSAR section 7.2.1.2 states the RAC supports NSRST function, DL5-PAM1, "Post-Accident Monitoring," which specifies that RAC performance variables are monitored to provide data for PAM. PSAR section 7.2.1 describes the RAC performance variables monitored for PAM, evaluated under PDC 13 (see SE section 7.2.1.1.5). The PSAR states that the RAC is in full compliance with RG 1.97 and contains Type B and D variables with redundant instrumentation adhering to IEEE 497-2016. The staff determined that the preliminary RAC design is consistent with DL5-PAM1 based on meeting PDC 13 and because the facility states full compliance with RG 1.97 and follows IEEE 497-2016.

7.2.1.1.10 Programmatic Special Treatments

Programmatic special treatments applied to the RAC are summarized in PSAR section 7.2.1.4 and include the QAP, D-RAP, EQ program, PITAP, CVAP, and RIM program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the RAC contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to RAC. Plant programs are evaluated further in SE chapter 8.

7.2.1.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.2.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.2.2 Intermediate Air Cooling System

PSAR section 7.2.2 describes the IAC. The IAC transfers heat from the IHT to the atmosphere and is the primary means of reactor heat removal during low power and shutdown conditions. The IAC consists of two trains, each containing an AHX, chimney structure, air blower, dampers, and air heater housed within an air stack structures and equipment (ASE). The AHX is connected in series within each intermediate loop. Intermediate sodium flows on the tube-side of the AHX, with air flowing on the shell-side. Each ASE contains a blower which provides forced air flow and remote operated dampers. These allow for the control of air flow in the AHXs and thus control of heat transfer to atmosphere. All components within the IAC are NSRST. The IAC supports one NSRST function, evaluated below.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the IAC:

- RG 1.31, Rev. 4;

PSAR sections 7.2.2 and 5.3 identify the following PDC as applicable to the IAC: 1, 2, 3, 4, 10, 34, 70, 72, 73, 74, 75, 76, and 77.

7.2.2.1 Technical Evaluation

7.2.2.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR table 7.2.2-2 identifies the design and construction codes and standards applicable to the IAC, including ASME BPVC VIII, ASME STS-1-2021, and Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) 1108, 2008.

Appendix 14.2 of NAT-13478 identifies the materials, design, construction codes and standards, and special treatments for major NSRST SSCs, including in the IAC. The major NSRST SSC identified for the IAC is the AHX which is designed and constructed to ASME BPVC VIII-2. The applicant identifies the following design and construction special treatments for the AHX:

- Creep effects will be evaluated using the Class A rules in ASME BPVC III-5 or ASME code case 2843-3. The staff notes that code case 2843-3 provides elevated temperature

rules that are similar to ASME BPVC III-5 Class A rules, including fatigue and creep-fatigue assessment.

- The allowable stresses for elevated temperature are limited to ASME BPVC III-5, extended to 500,000 hours service time (see SE section 7.1.2.1.1) and the delta ferrite limits for weld material in service above 800°F will meet the requirements in ASME BPVC III-5, HCB-2433, and RG 1.31.
- All pressure boundary welds will be full penetration welds with 100% volumetric examination to confirm weld quality. To provide assurance of material properties, all pressure boundary materials will require a CMTR.

Aside from the application of lethal service requirements, which do not exist in ASME BPVC VIII-2, these special treatments are comparable to those proposed for other portions of the IHT. The staff notes that the special treatments will provide increased assurance of appropriate design margin and fabrication quality through more rigorous design considering the effects of creep and fatigue and improved weld design and examination to ensure weld quality. Based on the special treatments identified above related to design and construction, the staff determined that the preliminary design of IAC AHX classified as NSRST and constructed to ASME BPVC VIII-2 is consistent with PDC 1.

ASME STS-1-2021 covers the design and construction of steel stacks. The staff notes that the NRC has not endorsed ASME STS-1-2021. However, the staff reviewed ASME STS-1-2021 and considers it acceptable for use in the IAC because it is a consensus standard applicable to the steel stacks that are used in the IAC. SMACNA 1108 is an industrial standard for the fabrication and installation of ducts. The staff considers it appropriate for use in IAC because it is a consensus standard applicable to design and construction of ducting.

Based on its review described above, the staff determined that the preliminary IAC design is consistent with PDC 1 because the codes and standards specified for the design and construction of the IAC SSCs are acceptable.

7.2.2.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.2.2.2 states that IAC components are primarily within the NSRST ASEs. PSAR table 7.2.2-1 states that NSRST IAC components are classified as SCN1, which are subject to seismic special treatments to provide assurance that risk-significant functions or DID can be performed. Seismic classifications, design, analysis, and qualification are discussed in PSAR section 6.4 and evaluated in SE section 6.4. Additional design basis hazards considered for safety-significant SSCs are discussed in SE section 6.1.

The staff determined that the preliminary IAC design is consistent with PDC 2 because the ASEs provide protection for safety-significant SSCs, and its safety-significant SSCs are designed to withstand postulated seismic events. The staff will review the detailed final design at the OL stage to confirm the ASEs adequately protect against natural phenomena, such as tornado missiles, consistent with PDC 2.

7.2.2.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.2.2.2 states that the IAC will be constructed with non-combustible and non-fire sustaining materials to the extent practical. The section also notes that ASEs are located away from other structures to reduce fire risks and that the IAC design includes features to minimize sodium leakage. Sodium leakage from the AHX is directed to collection points to minimize the surface area exposed to air. The staff's evaluation of the applicant's approach to fire protection for the IAC is provided in SE sections 7.5.2 and 8.3. Based on review of the information in PSAR section 7.2.2.2 and the evaluation in SE sections 7.5.2 and 8.3, the staff determined the preliminary IAC design is consistent with PDC 3.

7.2.2.1.4 Environmental and Dynamic Effects (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. The staff determined that the preliminary IAC design is adequately protected from external hazards, consistent with PDC 4, because The ASE protects the IAC from external hazards, such as transportation incidents.

PSAR section 7.1.4.2 states that IAC materials are compatible with a sodium environment, including sodium aerosols, vapors, and oxidation products, throughout the design lifetime. The staff's evaluation of the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including IAC SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Based on the identified R&D item for ensuring materials performance discussed in chapter 13 and the staff's evaluation of environmental effects on safety-significant, long-lived passive SSCs in contact with sodium or cover gas in SE section 7.1.2.1.4, the staff determined that the preliminary design information of IAC SSCs in contact with sodium or cover gas is consistent with PDC 4 with respect to being designed to accommodate environmental conditions associated with normal operations and postulated accidents. The staff will evaluate the maturation of the RIM DMA and associated MANDE at the OL stage.

7.2.2.1.5 Heat Removal (PDC 10 and 34)

PDC 10 and 34 address adequate heat removal during normal operation and AOOs. PSAR section 7.2.2.3 describes three IAC modes for removing core decay heat:

1. IAC active mode: Provides normal operation cooling to the intermediate loop with forced circulation of intermediate sodium and air via the ISPs and IAC blowers.
2. IAC blower mode: Provides off-normal operation cooling with natural circulation of intermediate sodium and forced air flow via the IAC blowers when ISPs are unavailable.
3. IAC passive mode: Provides emergency cooling with natural circulation of intermediate sodium in the IHT and natural convection of air.

The staff notes that the LBE analyses in PSAR chapter 3 refer to both active and blower modes as non-passive IAC.

PSAR section 7.2.2.3 states that one IAC train is sufficient for the limiting decay heat load in both active and blower modes, while passive mode requires both IAC trains. PSAR section 7.2.2.3 indicates that no automatic actuations or manual actions are needed for IAC to

perform its safety function in passive mode. Upon power loss, natural circulation develops in the IHT, and IAC dampers fail open, allowing natural convection airflow. Heated air exits the IAC exhaust, drawing ambient air into the low-elevation intake.

The AOO analyses in PSAR chapter 3 demonstrate the IAC's ability to provide adequate core cooling and remove residual heat during AOOs. All AOOs relying on heat removal from IAC (either forced or natural circulation flow) maintain fuel within design limits and as such have no associated radiological releases. The staff audited the LBE analyses and design basis heat removal calculations, and confirmed the IAC design can appropriately dissipate heat during normal operations and AOOs, consistent with PDC 10 and 34.

7.2.2.1.6 Intermediate Coolant Boundary (PDC 70, 75, 76, and 77)

PDC 70 requires the intermediate coolant system to be designed with sufficient margin to ensure that the intermediate coolant boundary maintains integrity during normal operation and AOOs, and that the integrity of the primary coolant boundary shall be maintained during postulated accidents. PDC 75 requires intermediate coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 76 requires intermediate coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 77 requires intermediate coolant boundary components to permit periodic inspection and functional testing.

PSAR section 7.2.2.2 states that intermediate coolant boundary components of the IAC will be designed to ensure that design conditions are not exceeded during normal operation, including AOOs. PSAR section 7.2.2.3 states that the AHX assembly will be designed and analyzed in accordance with ASME BPVC section VIII. The staff evaluated the use of industrial codes including ASME BPVC section VIII in SE section 6.4.1.5. The applicant identified special treatments to ASME BPVC section VIII requirements as described in SE section 7.2.2.1.1 to provide additional assurance of adequate SSC quality. Based on the information in PSAR section 7.2.2.2 and 7.2.2.3 and the evaluation in SE sections 6.4.1.5 and 7.2.2.1.1, the staff determined that the preliminary IAC design, and specifically the information related to the design and construction of AHX, is consistent with PDC 70 and 75.

PSAR section 7.2.3 states that overpressure protection is provided via the SCG and is evaluated in SE section 7.2.3. Materials for the intermediate pressure boundary are selected in consideration of exposure to high temperature liquid sodium, ambient air, and outdoor weather conditions. Based on the information in PSAR section 7.2.3 and the staff evaluation in SE section 7.1.2.1.4 on the preliminary design information addressing environmental effects on safety-significant SSCs, including IAC intermediate coolant boundary SSCs, in contact with sodium or cover gas containing sodium aerosols, the staff determined the preliminary IAC design is consistent with PDC 76 and 77.

The RIM program described in chapter 8 of the PSAR is used to identify degradation mechanisms, and appropriate test and MANDE inspection requirements for intermediate coolant boundary components. The staff evaluation of the RIM program is documented in SE section 8.1. Based on the inclusion of IAC intermediate coolant boundary components in the RIM program, the staff evaluation of the RIM program, and the supporting R&D activities as documented in SE sections 7.1.2.1.4 and 8.1, the staff determined that the preliminary information for the design of these components is consistent with PDC 76 and 77. The staff will

evaluate the maturation of the RIM DMA and associated MANDE through the prescribed R&D updates as well as with the final design in the OL application review.

7.2.2.1.7 Sodium Heating System (PDC 72)

PDC 72 requires heating systems for safety-significant SSCs that contain sodium. PSAR section 7.2.2.2 states that the ASEs contain heaters to prevent sodium freezing in the AHX, the only component containing sodium. The section further states that heating systems and controls ensure the temperature distribution and the rate of change of temperature in the intermediate sodium loops are maintained within their design limits. The staff determined that the preliminary IAC design is consistent with PDC 72 because the temperature of the intermediate sodium within the IAC can be maintained within necessary limits.

7.2.2.1.8 Sodium Leakage (PDC 73)

PDC 73 requires a means to detect sodium leakage and to limit and control sodium-air and sodium-concrete reactions. PSAR section 7.2.2.2 states that leak detection and catch pans are provided within the ASEs to identify and contain potential sodium leakage from the IAC AHXs. Based on the information in PSAR section 7.2.2.2 and the staff's evaluation of the preliminary design for sodium leakage detection and reaction prevention and mitigation, which includes the IAC, in SE sections 7.5.2 and 8.3, the staff determined the preliminary IAC design is consistent with PDC 73.

7.2.2.1.9 Sodium-Water Reaction Prevention and Mitigation (PDC 74)

PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water. PSAR section 7.2.2.2 states that the IAC is designed and located to prevent sodium-water contact. ASEs are not near water-containing systems, and their air intakes and exhausts have weather caps to limit water entry. The ASE also includes sodium containment features to limit potential leakage to the environment. If an AHX sodium leak is detected, dampers will automatically close on the affected ASE. Additionally, ASE catch pans have detectors to notify operators of water intrusion. Based on these design features limiting the potential for water to come in contact with potential leaked sodium, the staff determined that the preliminary design of the IAC is consistent with PDC 74.

7.2.2.1.10 Safety Functions

7.2.2.1.10.1 Heat Removal Function

PSAR section 7.2.2.2 states that the IAC supports NSRST function, DL4-HR1, "IAC Passive Mode Operation," which requires that the IAC transfers heat from the IHT to the atmosphere via natural convection airflow for LBEs when the ISPs and RAC are unavailable. PSAR table 5.2-4 states that DL4-HR1 is required for DID adequacy. PSAR section 7.2.2.3 describes how the IAC passive mode removes heat from the PHT, which is evaluated in SE section 7.2.2.1.5. For all LBEs relying on DL4-HR1, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff audited the details of these LBE analyses and confirmed the IAC design can dissipate heat at a rate necessary to mitigate the LBEs relying on passive operation of the IAC with the ISPs and RAC unavailable. The staff determined that the preliminary IAC design is consistent with this function because it allows for an alternate means of heat removal in the event the ISPs and RAC are unavailable.

7.2.2.1.11 Programmatic Special Treatments

Programmatic special treatments applied to the IAC are summarized in PSAR section 7.2.2.4 and include the QAP, D-RAP, EQ program, PITAP, and RIM program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the IAC contains NSRST SSCs, the staff determined that these programs are thus appropriate to apply to IAC. Plant programs are evaluated further in SE chapter 8.

7.2.2.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.2.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.2.3 Sodium Cover Gas System

PSAR section 7.2.3 describes the SCG, which consists of both the primary SCG (SCG-P) and the intermediate SCG (SCG-I). The primary SCG is responsible for supplying and maintaining argon gas to various systems and components within the RXB and fuel handling building (FHB). It receives high-pressure, purified argon from the NI argon gas distribution and storage system (NGA) and distributes it to the RV, RV-GV annulus, reactor head seals, PSP lube oil and gas barrier sealing, and other systems. Argon continually flows through the cover gas space in the RV and exhausts through a vapor trap cell wall into the primary filter skid, where it undergoes preheating, vapor condensation, and aerosol filtering to remove sodium vapors and aerosols. If cesium is present, the gas is routed through a cesium vapor filter before being exhausted to the gaseous radwaste processing system (RWG) in the FHB. Gas in the RV-GV annulus is generally recycled in a closed loop and only exhausts to relieve high pressure, either directly to RWG or nuclear island ventilation system (NHV), depending on activity. The primary SCG additionally monitors for impurities, radiation, and tag gas in the primary cover gas, RV-GV annulus, and primary SCG enclosures.

The intermediate SCG supplies argon gas to the IHT SSCs within the RAB. It also receives purified argon from the NGA and distributes it to the ISP tanks, IHT drain tanks, ISP gas barriers, and other related systems. The exhaust gas from the ISP tanks and gas barriers passes through expansion tanks, drain tanks, and filters where aerosols are removed before being discharged through an oil bubbler to the NHV. The primary SCG contains SR, NSRST, and NST SSCs, while the intermediate SCG contains NSRST and NST SSCs, as illustrated in PSAR figures 7.2.3-1 and 7.2.3-2, respectively. Additionally, the SCG performs several SR and NSRST functions that are described in the sections below.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the SCG:

- 10 CFR 50.34(f)(2)(x);
- RG 1.11, "Instrument Lines Penetrating the Primary Reactor Containment," Rev. 1 (ML100250396);
- RG 1.36, Rev. 1;

- RG 1.44, Rev. 1;
- RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," Rev. 1 (ML073200271);
- RG 1.61, Rev. 1;
- RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," Rev. 1 (ML12160A047);
- RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Rev. 39 (ML13339A515);
- RG 1.87, Rev. 2;
- RG 1.97, Rev. 5;
- RG 1.100, Rev. 4;
- RG 1.124, Rev. 3;
- RG 1.153, Rev. 1; and
- RG 1.199, "Anchoring Components and Structural Supports in Concrete," Rev. 1 (ML19336A079).

PSAR sections 7.2.3 and 5.3 identify the following PDC as applicable to the SCG: 1, 2, 3, 4, 10, 13, 14, 15, 16, 30, 31, 32, 60, 63, 70, 71, 72, 73, 74, 75, 76, 77, 78, and 79.

7.2.3.1 Technical Evaluation

7.2.3.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.2.3.4 identifies the design and construction codes and standards applicable to the SCG. PSAR section 7.2.3.2.1 states that the portion of the SCG from the RV up to and including the first isolation valve is SR, part of the primary coolant boundary, and designed and constructed to ASME BPVC III-5. Based on the endorsement of ASME BPVC III-5 in RG 1.87, the staff determined that the use of ASME BPVC III-5 is acceptable and is consistent with PDC 1.

Appendix 14.2 of NAT-13478 identifies the materials, design and construction codes and standards, and special treatments for major NSRST SSCs, including those in the SCG-P. The applicant specifies that the design and construction code for SCG-P piping is ASME B31.3-2020. The applicant identifies the following special treatments for NSRST SSCs in the SCG-P designed to ASME B31.3-2020, including SCG-P piping and valves:

- For piping determined to have non-negligible creep effects, an analysis per ASME BPVC III-5, subsection HCB-3634(b), will be applied. The staff notes that HCB-3634(b)

provides more conservative allowable thermal expansion stresses and stress criteria for thermal cycling to account for creep effects than ASME B31.3-2020.

- Allowable stresses for elevated temperature are limited to ASME BPVC III-5, extended to 500,000 hours service time (see SE section 7.1.2.1.1) and the delta ferrite limits for weld material will meet the requirements in ASME BPVC III-5, HCB-2433, for service above 800°F (ASME BPVC III-1, NC-2433, for service below 800°F) and RG 1.31.
- All pressure boundary welds will be full penetration welds with 100% volumetric examination to confirm weld quality. To provide assurance of material properties, all pressure boundary material will require a CMTR.
- Category M fluid classification will be applied. The staff notes that Category M fluid applies to piping systems containing highly toxic fluids and requires more stringent design, material selection, fabrication, and leak testing than normal fluid service.

The applicant specifies that the SCG-P vapor trap cell condenser and the primary aerosol and cesium filters will be designed and constructed to ASME BPVC VIII-1. The applicant identifies the following special treatments for NSRST SSCs in the SCG-P designed to ASME BPVC VIII-1:

For the SCG-P vapor trap cell condenser, because ASME BPVC VIII-1 does not explicitly account for creep effects in long-term service, components subject to non-negligible creep, as determined by screening per ASME BPVC III-5, mandatory appendix HCB-III, will be evaluated using the Class A rules in ASME BPVC III-5 or ASME code case 2843-3. The staff notes that code case 2843-3 provides elevated temperature rules that are similar to ASME BPVC III-5 Class A rules, including fatigue and creep-fatigue assessment.

- For SCG-P vapor trap cell condenser, the allowable stresses for elevated temperature are limited to ASME BPVC III-5, extended to 500,000 hours service time (see SE section 7.1.2.1.1) and the delta ferrite limits for weld material in service above 800°F will meet the requirements in ASME BPVC III-5, HCB-2433, and RG 1.31.
- For the SCG-P primary aerosol and cesium filters, the delta ferrite for weld material in service below 800°F will meet the requirements in ASME BPVC III-1, NC-2433, and RG 1.31.
- For SCG-P vapor trap cell condenser and SCG-P primary aerosol and cesium filters, fatigue analysis will be performed using the rules in ASME BPVC VIII-2. All pressure boundary welds will be full penetration welds with 100% volumetric examination to confirm weld quality. To provide assurance of material properties, all pressure boundary materials will require a CMTR.

The staff notes that the identified special treatments will provide increased assurance of appropriate design margin and fabrication quality. These treatments incorporate more rigorous design considerations for creep and fatigue effects, as well as improved weld design and examination to ensure weld quality. Based on these identified special treatments, the staff determined that the preliminary design of SCG SSCs classified as NSRST and constructed to ASME B31.3-2020 and ASME BPVC VIII-1 is consistent with PDC 1. While appendix 14.2 of NAT-13478 does not address all NSRST SSCs in the SCG, such as the vapor trap cell barrier

and those in the SCG-I, the staff will review the design and construction codes and standards for all NSRST SSCs with the final design at the OL stage. The staff expects a similar approach will be applied for special treatments of NSRST SSCs not addressed in appendix 14.2 of NAT-13478. As such the staff focused its review on the most safety significant NSRST SSCs and determined that further information can reasonably be left for later consideration at the OL stage.

ASME AG-1-2019 generally covers the design of filters, ventilation, and process gas treatment. A previous edition and addenda, ASME AG-1-2009 (including 2010 Addenda 1a and 2011 Addenda 1b), is endorsed by the NRC in RG 1.140 for the design, inspection, and testing of normal atmosphere cleanup systems for controlling releases of airborne radioactive materials to the environment in light water reactors. The staff reviewed ASME AG-1-2019 and determined it is acceptable for the SCG. SCG instruments that support PAM adhere to IEEE 497-2016 and are in full conformance with RG 1.97, which endorses the use of IEEE 497-2016. The PSAR identifies IEEE 382-2006 for qualification of applicable SR actuators, which is endorsed for use by RG 1.73.

For seismic qualification, PSAR section 7.2.3.2 states that applicable SR SCG SSCs are designed to ASME QME-1-2023 and IEEE 344-2020. The PSAR states partial conformance with RG 1.100, which endorses IEEE 344-2013 and QME-1-2017. The staff notes that the NRC has not yet endorsed ASME QME-1-2023 or IEEE 344-2020. However, the staff reviewed ASME QME-1-2023 and IEEE 344-2020 and considers them appropriate if implemented consistent with the regulatory positions specified in Rev. 4 of RG 1.100. The PSAR further states that these codes will be applied to NSRST SCG SSCs in accordance with their safety significance. Furthermore, the PSAR identifies ANSI/AISC N690-18, which is endorsed by RG 1.243, for guidance on SR steel structures.

The PSAR additionally states that applicable SCG instruments are designed to IEEE 603-2018. The staff notes that the NRC has not yet endorsed IEEE 603-2018, However, the staff reviewed IEEE 603-2018 and considers it acceptable, noting that IEEE 603-1991 is endorsed in RG 1.153.

Based on its review described above, the staff determined that the preliminary SCG design is consistent with PDC 1 because the codes and standards specified for the design and construction of the SCG SSCs are acceptable.

7.2.3.1.2 Design Bases for Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.2.3.2.1 states that safety-significant SCG SSCs are primarily below grade in the RXB or RAB, with SR SCG SSCs located in the SR reinforced concrete RXB substructure. The RXB substructure provides protection from external flooding, while the SR SCG SSCs' location below grade provides protection from other natural phenomena including tornadoes and extreme climate conditions. Safety-significant SCG SSCs that are above grade are designed to a hazard level commensurate with their safety-significance. PSAR section 7.2.3.2.1 classifies SR SCG SSCs as SCS1, qualified to withstand seismic loads from the SSE. NSRST SSCs are classified as either SCN1 or SCN3, with seismic special treatments to ensure risk-significant functions or DID can be performed. Seismic classifications, design, analysis, and qualification are detailed in PSAR section 6.4 and evaluated in SE section 6.4.1.1.

The staff determined that the SCG design ensures safety-significant SSCs can perform their safety functions despite natural phenomena. NSRST components are primarily housed below grade in the RXB or RAB, while SR components are within the SR NCB substructure; all are designed to withstand seismic events consistent with their safety significance. As such, the staff determined that the preliminary SCG design is consistent with PDC 2.

7.2.3.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.2.3.2.1 states that all SCG SSCs are designed with non-combustible, fire-resistant materials. The section further states that safety-significant SCG SSCs are located to prevent damage from fires or explosions, and that sodium containing components are either located in inert environments or enclosed to mitigate sodium leakage. The staff's evaluation of the applicant's approach to fire protection for the SCG is provided in SE sections 7.5.2 and 8.3. Based on the information in PSAR section 7.2.3.2.1 and the evaluation in SE sections 7.5.2 and 8.3, the staff determined the preliminary SCG design is consistent with PDC 3.

7.2.3.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.2.3.2.1 states that safety-significant SSCs in the SCG are protected from environmental conditions, dynamic transients, and chemical accidents based on their location within the RXB and RAB. As such, the staff determined that the preliminary SCG design is adequately protected from external hazards, consistent with PDC 4.

The PSAR further states that the SCG is designed to handle exposure to liquid sodium, sodium aerosols, and sodium oxidation products. The staff's evaluation of the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including SCG SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Based on the identified R&D item for ensuring materials performance discussed in chapter 13 and the staff's evaluation of environmental effects on safety-significant, long-lived passive SSCs in contact with sodium or cover gas in SE section 7.1.2.1.4, the staff determined that the preliminary design information of SCG SSCs in contact with sodium or cover gas is consistent with PDC 4 with respect to being designed to accommodate environmental conditions associated with normal operations and postulated accidents. The staff will evaluate the maturation of the RIM DMA and associated MANDE at the OL stage.

7.2.3.1.5 SCG Instrumentation (PDC 13 and 63)

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and for accident conditions. PDC 63 requires that appropriate systems are provided for fuel storage and radioactive waste systems to (1) detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) initiate appropriate safety actions. PSAR section 7.2.3.2.1 states that the SCG provides instrumentation to monitor the RV and GV pressures, primary coolant boundary and SCG passive barrier isolation valve positions, RV relief valve positions, and cover gas radiation levels. These variables are monitored for PAM, with SCG instruments in full compliance with RG 1.97 and IEEE 497-2016. This section further states that the SCG interfaces with the

radiation monitoring system (RMS) to monitor radiation levels and detect excessive levels in the primary cover gas, indicating potential issues with fuel stored in the IVS.

The staff determined that the preliminary SCG design is consistent with PDC 13 and 63 because it contains instruments that can monitor RV and GV pressures, SCG valve positions, and radiation levels in the cover gas as illustrated in PSAR figure 7.2.3-1 that shows the SCG connections that enable these monitoring functions.

7.2.3.1.6 Reactor Design (PDC 10)

PDC 10 requires that the reactor core and associated systems are designed with appropriate margin to ensure SARRDLs are not exceeded during normal operation and AOOs. PSAR section 7.2.3.2.1 states that the SCG provides pressure to the RV and GV, and supplies argon gas to reactor head seals to prevent leakage to the environment. The SCG does not play a direct role in controlling heat generation or heat removal from the reactor core; therefore, the staff considers its role in PDC 10 to be primarily supporting the appropriate function of the PHT and CRD, as discussed in SE sections 7.1.3 and 7.2.5, as well as maintaining the primary coolant boundary.

The staff considered a potential scenario in which the SCG could affect the core by supplying inert gas in such a way that allows it to become entrained into the primary coolant. This could result in gas bubbles flowing through the core, potentially causing positive reactivity insertion depending on the location of bubble introduction. Since cover gas from the SCG is injected through the reactor head, gas entrainment is only possible at the cover gas-sodium interface; therefore, turbulence or wave formation resulting from high velocity cover gas flow would be necessary for entrainment to occur. The staff confirmed in audit that the cover gas velocity is low enough that minimal gas entrainment would be expected.

The staff also expects the SCG to have an associated SARRDL because gaseous radionuclides released from fuel during operation will flow through the primary coolant and enter the SCG. The SCG radionuclide monitoring capabilities of the SCG, discussed in SE sections 7.2.3.1.6 and 7.2.3.1.9, provide the necessary indication for this.

Because the SCG maintains a low leakage rate, provides appropriate support for SSCs involved in transient analyses, and provides adequate instrumentation to monitor SARRDLs and ensure they are not exceeded, the staff considers the preliminary design of SCG to be consistent with PDC 10.

7.2.3.1.7 Primary Coolant Boundary (PDC 14, 15, 30, 31, and 32)

PDC 14 requires primary coolant boundary components to be designed, fabricated, erected, and tested to provide an extremely low probability of leakage, rapidly propagating failure, and gross rupture. PDC 15 requires the design conditions of the primary coolant boundary to not be exceeded during conditions of normal operation, including AOOs. PDC 30 requires primary coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 31 requires primary coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 32 requires primary coolant boundary components to permit periodic inspection and functional testing.

PSAR section 7.2.3.2.1 describes the portions of the SCG that comprise the primary coolant boundary and states that these components will be constructed to ASME BPVC III-5 to ensure a low probability of leakage or rupture and to maintain structural integrity during all postulated accidents. The SCG primary coolant boundary SSCs are designed to withstand environmental effects, including sodium exposure, thermal conditions, radiation fluence, and dynamic fluid effects during all normal operation conditions and postulated accidents. The RIM program, described in PSAR chapter 8, is used to assess degradation mechanisms and to identify appropriate MANDE requirements for SCG primary coolant boundary components.

PSAR section 7.2.3.3 states the primary SCG provides overpressure protection for the RV, retains overpressure discharge, and a boundary to primary coolant leakage. RV overpressure protection is evaluated in SE section 7.2.3.1.15.

The staff's evaluation of the preliminary design information addressing environmental effects on safety-significant SSCs, including SCG primary coolant boundary SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Additionally, the staff's evaluation of the RES primary coolant boundary components in SE section 7.1.2.1.6 also encompasses SCG primary coolant boundary components. The staff evaluation of the RIM program is provided in SE section 8.1.

PSAR chapter 13 and section 12.1 of NAT-13478 describe an R&D item focused on assuring adequate structural materials performance in service. This R&D item also supports the RIM program through the development of performance monitoring methods, such as material surveillance programs, inspection methods and means of access, and monitoring approaches, as well as the development of PFM and LBB methodologies. The staff proposed two conditions for the KU1 CP related to updates on this R&D item as discussed in greater detail in SE chapter 13.

Based on the staff evaluation of material compatibility of primary coolant boundary components in SE section 7.1.2.1.4, conformance with ASME BPVC III-5 and RG 1.87 for the design and construction of these components, and the proposed implementation of the RIM program and the associated R&D activities for assuring adequate material performance in service, the staff determined that the preliminary design of the SCG primary coolant boundary components is consistent with PDC 14, 15, 30, 31, and 32.

7.2.3.1.8 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. PSAR section 7.2.3.1.1 states that the SCG supports multiple layers of functional containment to control the release of radioactivity to the environment and ensure that functional containment design conditions are not exceeded. As noted in SE section 7.2.3.1.7, SCG primary coolant boundary components are designed to ASME BPVC III-5. The SCG also supports functional containment through:

- The vapor trap cell which acts as a sealed barrier around exhaust system components. It contains a steel liner with a concrete shield, with sealed penetrations to minimize leakage.
- SCG exhaust gas monitoring cabinets outside of the HAA that are housed in secondary enclosures and held at a slight vacuum to contain leaks to prevent exceeding SARRDLs.

- The SCG provides argon gas to support the primary barrier purge and barrier gas seals in the RES to enable radionuclide retention.
- The SCG provides overpressure protection which prevents rupture of the primary coolant boundary.
- The SCG provides the ability to isolate the SCG following a leak.

The staff notes that PSAR table 5.2-3 identifies a SR SCG isolation function, DL3-RR10, “SCG Isolation,” which is associated with multiple DBEs and BDBEs. Note 1 to PSAR table 5.2-3 states that this function will be fail safe with SR detection and actuation signals, but it “has not yet been integrated into the design or analysis and is therefore not discussed further” in the PSAR, including in chapter 7. The staff will review DL3-RR10 as part of the final design in the OL application. Further discussion of this function is provided in SE sections 5.4 and 7.2.3.1.15. The staff notes that functional containment performance is assessed through the mechanistic source term analyses described in PSAR section 3.2 and SE section 3.2, which at the CP stage rely on assumptions regarding leakage rates across barriers. The staff will review leakage rate justifications at the OL stage relative to the design of the SSCs comprising each functional containment boundary.

The staff determined that the preliminary SCG design is consistent with PDC 16 because it includes multiple barriers to minimize leakage and support leakage monitoring, includes a SR SCG isolation function, and there are plans to justify leakage assumptions in the final design.

7.2.3.1.9 Cover Gas Inventory, Purity Control, and Radioactive Release (PDC 60, 71, and 79)

PDC 60 requires, in part, a means to control the release of radioactive materials in gaseous and liquid effluents during normal reactor operation, including AOOs. PDC 71, in part, requires systems to maintain cover gas purity within specified limits based on radionuclide concentrations and air or moisture ingress as a result of a leak of cover gas. PDC 79 requires a system to maintain cover gas inventory, ensuring primary coolant sodium design limits are not exceeded as a result of cover gas loss due to small leaks or ruptures in the primary coolant boundary.

PSAR sections 7.2.3.1.1 and 7.2.3.2.1 describe the once through flow path and filtering process for the primary SCG. Pure argon is supplied by the NGA to the RV at a positive pressure. The PSAR states that a flow control valve maintains a minimum flow throughout the cover gas region to remove contaminants, supplemented with makeup argon if a rapid depressurization of the RV occurred. Argon accumulators are located downstream of the NGA supply in the event of disruption. Exhaust gas from the RV passes through a vapor trap and aerosol filter to remove sodium vapors and aerosols. If cesium is present, the gas is routed through a cesium vapor filter before being exhausted to the gaseous RWG. SCG gas from the RV-GV annulus exhausts to the NHV unless high concentrations of radiological effluents exist, in which case it discharges to the RWG. In an overpressure condition, SCG relief valves discharge to a holdup volume with sufficient capacity to hold the radiological effluent. PSAR section 7.2.3.3 describes how the primary SCG monitors impurities, radiation, and tag gas in the cover gas; and oxygen, radiation, and filter differential in the RV-GV annulus gas. Impurities in the cover gas, such as oxygen, indicate potential leaks in the primary coolant boundary, while tag gas indicates failed fuel. Monitoring RV-GV annulus gas allows for detection of RV leaks to the GV annulus. PSAR

figure 7.2.3-1 illustrates the connections for collecting and analyzing cover gas and RV-GV annulus samples.

The staff determined that the preliminary SCG design is consistent with PDC 60 because primary SCG exhaust is monitored for radiation and fission products and it discharges to a holdup volume on a high pressure condition, ensuring that discharge to the RWG can be controlled to ensure normal effluent values are not exceeded. The staff determined that the preliminary SCG design is consistent with PDC 71 because the primary SCG is designed as a once through system with exhaust processing and can monitor for radiation and any air or moisture ingress. The staff determined that the preliminary SCG design is consistent with PDC 79 based on the continuous argon flow from the NGA and the presence of local accumulators.

7.2.3.1.10 Intermediate Coolant Boundary (PDC 70, 75, 76, and 77)

PDC 70 requires the intermediate coolant system to be designed with sufficient margin to ensure that the intermediate coolant boundary maintains integrity during normal operation and AOOs, and that the integrity of the primary coolant boundary shall be maintained during postulated accidents. PDC 75 requires intermediate coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 76 requires intermediate coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 77 requires intermediate coolant boundary components to permit periodic inspection and functional testing.

PSAR section 7.2.3.4 states that SCG intermediate coolant boundary SSCs are designed and constructed in accordance with ASME B31.3-2020 and ASME BPVC VIII-1. The staff evaluated the use of industrial codes including ASME B31.3-2020 and ASME BPVC VIII-1 in SE section 6.4.1.5. Based on the staff evaluation in SE section 6.4.1.5 and the evaluation of codes and standards for SCG SSCs in SE section 7.2.3.1.1, the staff determined that the preliminary design information for SCG intermediate coolant boundary components is consistent with PDC 70 and 75. While appendix 14.2 of NAT-13478 does not address all NSRST SSCs in the SCG, such as those in the SCG-I, the staff will review the design and construction codes and standards for all NSRST SSCs with the final design. The staff expects a similar approach will be applied for special treatments of NSRST SSCs not addressed in appendix 14.2 of NAT-13478. As such the staff focused its review on the most safety significant NSRST SSCs and determined that further information can reasonably be left for later consideration.

Table 5.2 of NAT-13478 states that SCG intermediate coolant boundary components are constructed of austenitic stainless steel, either 304H or 316H with ER16-8-2 weld metal at high temperatures or 304 or 316 for low temperature service. The staff evaluation of the preliminary design information addressing environmental effects on safety-significant SSCs, including SCG intermediate coolant boundary SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. The RIM program, described in PSAR chapter 8, is used to assess degradation mechanisms and identify appropriate MANDE requirements for SCG intermediate coolant boundary components. The staff evaluation of the RIM program is documented in SE section 8.1. Based on the inclusion of SCG intermediate coolant boundary components in the RIM program, the staff's evaluation of the RIM program and the supporting R&D activities documented in SE sections 7.1.2.1.4 and 8.1, the staff determined that the preliminary information for the design of these components is consistent with PDC 76 and 77.

The staff will evaluate the maturation of the RIM DMA and associated MANDE through the prescribed R&D updates as well as with the final design in the OL application.

7.2.3.1.11 Sodium Heating Systems (PDC 72)

PDC 72 requires heating systems for safety-significant SSCs that contain sodium. PSAR section 7.2.3.2.1 states that safety-significant portions of the SCG that contain, or could contain, sodium are equipped with heating systems to prevent freezing. This includes heat tracing in the primary SCG exhaust line, where sodium is expected during normal operations, and on the primary SCG supply line to prevent sodium diffusion back toward the argon-supplying SSCs from freezing in the event of a gas flow interruption.

PSAR figure 7.2.3-1 depicts the location of the SR primary overpressure protection skid within the primary SCG. PSAR section 7.2.3.2.1 states that the SCG pressure relief devices for internal overpressure protection are located off the main feed line supplying argon to the RV. The applicant states that a minimum argon flow is maintained in the main feed line at all times to ensure sufficient gas velocity, thereby preventing sodium diffusion back into the line, which could impair the overpressure protection function. In the event of a loss of argon flow, the use of the heaters may be required to prevent freezing and to avoid compromising the overpressure protection function.

Because safety-significant sodium-containing SCG SSCs are equipped with heaters to prevent freezing, the staff determined that the preliminary SCG design is consistent with PDC 72. The staff will review the final design, technical specifications, and operating procedures at the OL stage to ensure that PDC 72 is met.

7.2.3.1.12 Sodium Leakage Detection and Reaction Prevention and Mitigation (PDC 73)

PDC 73 requires a means to detect sodium leakage and to limit and control sodium-air and sodium-concrete reactions. PSAR section 7.2.3.2.1 describes the primary SCG and intermediate SCG sodium leakage detection capabilities. The RV-GV annulus is supplied with argon gas to create an inert environment that mitigates the effects of a sodium leak. This annulus gas is recirculated and monitored by the primary SCG for indications of sodium leakage, including changes in absolute filter differential pressure, oxygen concentration, and radiation levels. The vapor trap cell, which houses the primary filter skid and includes an inner steel liner assembly, is inerted with nitrogen containing a low level of oxygen and maintained under slight vacuum conditions. These measures mitigate potential sodium leaks, limit sodium-air and sodium-concrete reactions, and reduce the consequences of leakage from SSCs within the cell. The intermediate SCG monitors for impurities that may indicate system degradation or the presence of leakage pathways. Additional discussion on sodium leakage detection and reaction prevention and mitigation is provided in PSAR sections 7.5.2 and 8.2 and evaluated in SE sections 7.5.2 and 8.3.

The staff determined that the preliminary SCG design is consistent with PDC 73 because it includes a means for detecting and containing sodium leakage and mitigating the effects of sodium-air and sodium-concrete reactions.

7.2.3.1.13 Sodium-Water Reaction Prevention and Mitigation (PDC 74)

PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water. PSAR section 7.2.3.2.1 states that sodium-containing portions of the SCG

are separated from water sources to prevent water-sodium interactions. PSAR section 8.2.1 states that there are no water-containing systems within the RXB and RAB due to the presence of sodium in these buildings. Based on the lack of water sources in the RXB and RAB, the staff finds the preliminary design of the SCG is consistent with PDC 74.

7.2.3.1.14 Primary Coolant System Interfaces (PDC 78)

PDC 78 outlines requirements for SSCs containing fluids that interface with the primary coolant system. Argon in the SCG directly interfaces with the primary coolant. Based on the staff's evaluations of the argon-sodium interface described in SE section 7.2.3.1.9 above, the staff determined that the preliminary design of the SCG is consistent with PDC 78.

7.2.3.1.15 Safety Functions

7.2.3.1.15.1 Radionuclide Retention

PSAR section 7.2.3.2.2 and table 5.2-3 state that the SCG supports seven functions which provide radionuclide retention. These are:

- DL3-RR1c, SCG Primary System Barrier (SR)
- DL3-RR10, SCG Isolation (SR)
- DL2-RR10, Primary SCG Barrier (NSRST)
- DL4-RR4a, Automatically Close SCG Isolation Valves on Leak Detection (NSRST)
- DL4-RR4, SCG Cell Barriers (NSRST)
- DL4-RR4c, Vapor Trap Cell Isolation on Overpressure (NSRST)

Note 1 to PSAR table 5.2-3 states that DL3-RR10 mitigates radionuclide release from SCG pathways following certain LBEs. This will include a SR SCG isolation, with SR detection and actuation signals. The applicant states that this function has not yet been integrated into the design and is not discussed further in the CP application. The staff audited the LBEs that rely on this function and determined that a SR SCG isolation function is appropriate for radionuclide retention. Based on the information in note 1 to PSAR table 5.2-3 and the additional staff evaluation on the SR SCG isolation in section 5.4, the staff determined the preliminary SR SCG design is consistent with this function. The staff will review the integration of this function into the final design during the OL review.

PSAR section 7.2.3.2.2 states that DL4-RR4a requires the SCG isolation valves to close upon leak detection. This trip is automatically initiated by the NSRST auxiliary monitoring and control system (AMC), which receives radioactivity indication signals from the radiation monitoring system (RMS). AMC and RMS are discussed in PSAR sections 7.6.2 and 7.6.6 and evaluated by staff in SE section 7.6.2 and 7.6.6, respectively. Upon detection of a SCG leak, AMC automatically closes the SCG reactor vessel exhaust isolation valves downstream of the vapor cell. PSAR table 5.2-4 states that DL4-RR4a is required for DID adequacy. The staff determined that the preliminary SCG design is consistent with this function because it operates

independently from the SR SCG isolation (DL3-RR10) and relies on alternate indications. The staff will review the independence of this function from DL3-RR10 during the OL review.

DL3-RR1c specifies that the SCG provides a portion of the primary coolant boundary and acts as a low leakage barrier to support radionuclide retention. PSAR section 7.2.3.2.1 states that this portion consists of the primary SCG piping from the RES to the SCG primary isolation valves, excluding the closure of these valves. Based on the inclusion of a SR SCG isolation in DL3-RR10, the evaluation of material compatibility of the primary coolant boundary components in SE section 7.1.2.1.4, the evaluation of the PDC related to the primary coolant boundary in SE section 7.2.3.1.7, and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in-service material performance, the staff determined that the preliminary SCG design is consistent with this function.

DL2-RR10 specifies that the primary SCG provides a radionuclide barrier for SSCs beyond the primary coolant boundary isolation valves. Based on the inclusion of a SR SCG isolation in DL3-RR10, the evaluation of material compatibility of safety-significant SSCs, including SCG SSCs, in contact with sodium or cover gas containing sodium aerosols in SE section 7.1.2.1.4, the evaluation of the PDC related to the primary coolant boundary in SE section 7.2.3.1.7, the evaluation of special treatments for SCG NSRST SSCs in SE section 7.2.3.1.1, and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in-service material performance, the staff determined that the preliminary SCG design is consistent with this function.

DL4-RR4 specifies that the SCG vapor trap cell provides a low leakage barrier for postulated SCG releases. PSAR section 7.2.3.2.1 states that the cell barrier utilizes a steel liner assembly that is sealed to prevent leakage and is installed within a concrete biological shield. NAT-13478 states that the vapor trap cell will be designed, constructed, and analyzed to ASME BPVC VIII-1. Based on the mild normal operating conditions for this assembly (temperature and inert gas), its inclusion in the RIM program, and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in-service material performance, the staff determined that the preliminary SCG design is consistent with this function. The staff will review the design and construction codes and standards for all NSRST SSCs with the final design.

DL4-RR4c requires the primary SCG vapor trap cell to be isolated upon detection of a primary coolant overpressure relief valve open indication. PSAR section 7.2.3.2.1 states that the primary coolant overpressure relief valves discharge into the vapor trap cell, which is then isolated to act as a low leakage barrier. This function is initiated by the NSRST utility monitoring and control system (UMC), described in PSAR section 7.6.3 and evaluated in SE section 7.6.2. Upon detection of an overpressure relief valve open indication, UMC automatically closes the applicable NHV dampers, isolating the SCG vapor trap cell. This function reduces the dose consequences of an overpressure event in which cover gas is discharged into the vapor trap cell. The staff notes that performance of this function in the PSAR LBE analyses is based on assumptions about leakage rates across the HAA boundary, operation and isolation of NHV systems, and performance of NHV filtration. Based on the information in PSAR section 7.2.3.2.1 and PSAR section 7.6.3 as evaluated in SE section 7.6.2, the staff determined the preliminary design of the SCG is consistent with the ability to perform this function. The staff will verify assumptions and evaluate the final design's performance during the OL application review.

7.2.3.1.15.2 RV Overpressure Protection

PSAR section 7.2.3.2.2 states that the SCG supports SR function, DL3-RR7, “RES Pressure Relief,” which specifies that the SCG provides both internal and external overpressure protection to the RV, with external protection provided within the RV-GV annulus and the CRDM housings. PSAR table 5.2-3 does not list any associated LBEs for DL3-RR7. During audit discussion, the applicant stated that the initial cause(s) of a primary overpressure event have not been defined at the preliminary design stage.

PSAR figure 7.2.3-1 depicts the location of the SR primary overpressure protection skid in the primary SCG. PSAR section 7.2.3.2.1 states that the SCG pressure relief devices for internal overpressure protection are off the main feed line supplying argon to the RV. The applicant states that a minimum argon flow is maintained in the main feed line at all times to ensure sufficient gas velocity, thereby preventing sodium diffusion back into the line, which could impair the overpressure protection function. External overpressure protection is provided by pressure relief devices in the RV-GV annulus and CRDM.

The staff determined that the preliminary SCG design is consistent with DL3-RR7, as it incorporates overpressure protection provisions in accordance with ASME BPVC III-5, which is endorsed in RG 1.87. This determination is based on the location of the pressure relief devices and the assurance that continuous argon flow will prevent sodium interference with operation. The staff will evaluate the final SCG design, including argon flow rate and the overpressure protection setpoints, during the OL application review.

Paragraph 10 CFR 50.34(f)(2)(x) requires a test program and the performance of tests to qualify reactor coolant system relief and safety valves. PSAR table 1.4-4 states the design will qualify reactor coolant system relief and safety valves for argon containing sodium vapors and aerosols at expected conditions during identified plant events using analysis and testing. The staff determined that the preliminary SCG design meets 10 CFR 50.34(f)(2)(x) because analysis and tests will be performed for the primary pressure relief valves. The staff determined that further information on testing of the SCG pressure relief valves can be reasonably left for the OL stage.

7.2.3.1.15.3 Heat Removal Function

PSAR section 7.2.3.2.2 states that the SCG supports NSRST function, DL4-HR1, “IAC Passive Mode Operation,” which requires the IAC to transfer heat from the IHT to the atmosphere via natural convection airflow for LBEs when the ISP and RAC are unavailable. PSAR table 5.2-4 states that DL4-HR1 is required for DID adequacy. This function requires the SCG-I to maintain its pressure-retaining functions during all modes and LBEs when IAC operability is needed. The SCG-I also provides overpressure protection for the IAC. During the audit, the applicant clarified that the SCG-I does not perform any heat removal functions itself but supports this function solely by maintaining its boundary integrity. The staff evaluated the SCG-I’s ability to maintain boundary integrity and therefore support IAC heat removal functions. Based on the evaluation of the PDC related to the intermediate coolant boundary in SE section 7.2.3.1.10 and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in-service material performance, the staff determined that the preliminary SCG design is consistent with this function.

7.2.3.1.15.4 PAM Function

PSAR section 7.2.3.2.2 states the SCG supports NSRST function, DL5-PAM1, “Post-Accident Monitoring,” which specifies that SCG variables are monitored to provide data for PAM. The PSAR states that SCG PAM variables consist of RV and GV pressure, SCG isolation valve and

SCG relief valve positions, and cover gas radiation. The PSAR states that these SCG instruments are in full compliance with RG 1.97, adhering to IEEE 497-2016. The staff determined that the preliminary RAC design is consistent with DL5-PAM1 because the facility states full compliance with RG 1.97 and follows IEEE 497-2016.

7.2.3.1.16 Programmatic Special Treatments

Programmatic special treatments applied to the SCG are summarized in PSAR section 7.2.3.4 and include the QAP, D-RAP, EQ program, IST, PITAP, and RIM program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the SCG contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to SCG. Plant programs are evaluated further in SE chapter 8.

7.2.3.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.2.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.2.4 Sodium Processing System

PSAR section 7.2.4 describes the SPS, which monitors and removes impurities from primary, intermediate, and ex-vessel storage tank (EVST) sodium. The SPS consists of three loops: the primary SPS (SPS-P), intermediate SPS (SPS-I) and EVST SPS (SPS-E). The SPS-P draws primary sodium from the RV, the SPS-I draws intermediate sodium from the IHT or IHT drain tanks, and the SPS-E draws sodium from the EVST. Each loop contains submersible electromagnetic pumps to deliver sodium to the monitoring and purification equipment. The purification equipment consists of cold traps which remove impurities, and for the SPS-P, a cesium trap to remove cesium. Monitoring equipment consists of the plugging temperature indicators (PTIs), multi-purpose samplers (MPSs), and hydrogen sensors. After sampling and purification, the sodium is returned to its respective loop. The SPS-P has components located in both the RXB and RAB, while the SPS-I is located entirely within the RAB. The SPS-E is located in the FHB. The SPS supports both SR and NSRST functions which are evaluated below.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the SPS:

- 10 CFR 50.34(f)(1)(xii),
- 10 CFR 50.34(f)(2)(viii),
- RG 1.22, Rev. 0;
- RG 1.31, Rev. 4;
- RG 1.34, "Control of Electroslag Weld Properties," Rev. 1 (ML101670357);
- RG 1.36, Rev. 1;
- RG 1.44, Rev. 1;

- RG 1.45, Rev. 1;
- RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," Rev. 1 (ML101870612);
- RG 1.61, Rev. 1;
- RG 1.84, Rev. 39;
- RG 1.87, Rev. 2;
- RG 1.100, Rev. 4;
- RG 1.153, Rev. 1;
- RG 1.189, Rev. 5;
- RG 1.193, "ASME Code Cases Not Approved for Use," Rev. 7 (ML21181A224);
- RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," Rev. 0 (ML080500187); and
- RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants -- Design Stage Man-Rem Estimates," Rev. 1 (ML003739550).

PSAR sections 7.2.4 and 5.3 identify the following PDC as applicable to the SPS: 1, 2, 3, 4, 14, 15, 16, 30, 31, 32, 33, 61, 63, 70, 71, 72, 73, 74, 75, 76, 77, and 78.

7.2.4.1 Technical Evaluation

7.2.4.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.2.4.4 identifies the design and construction codes and standards applicable to the SPS. PSAR section 7.2.4.2.1 states that the portion of the SPS-P piping from the RV up to and including the first isolation valve is SR, part of the primary coolant boundary, and designed and constructed to ASME BPVC III-5. Similarly, the SPS-E piping from the EVST up to the first isolation valve is SR according to PSAR table 7.2.4-1 and will be designed to ASME BPVC III-5 according to PSAR section 7.3.2.3. Based on the endorsement of ASME BPVC III-5 in RG 1.87, the staff determined that the use of ASME BPVC III-5 is acceptable and consistent with PDC 1.

PSAR section 7.2.4.2.1 states that NSRST SPS piping system SSCs, including SPS-P SSCs outside of the primary coolant boundary, are designed in accordance with ASME B31.1-2022. Appendix 14.2 of NAT-13478 identifies the materials, design and construction codes and standards, and special treatments for major NSRST SSCs, including in the SPS. The applicant identifies the following special treatments for NSRST SSCs in the SPS-P designed to ASME B31.1-2022, including SPS-P piping and valves:

- Delta ferrite limits for weld material in service below 800°F will meet the requirements in NC-2433 from ASME BPVC III-1 and RG 1.31.
- All pressure boundary welds will be full penetration welds with 100% volumetric examination to confirm weld quality. To provide assurance of material properties, all pressure boundary material will require a CMTR.
- ASME B31.1-2022 toxic fluid requirements will be applied to liquid sodium-containing components designed to ASME B31.1-2022.
- ASME B16.34-2020 section 8 will be applied for special class valves.

The applicant specifies that the SPS-P cold trap, SPS-P cesium trap, and SPS-P MPS modules pressure boundaries will be designed and constructed to ASME BPVC VIII-1. The applicant identifies the following special treatments for NSRST SSCs in the SPS designed to ASME BPVC VIII-1:

- For SPS-P MPS modules pressure boundaries, because ASME BPVC VIII-1 does not explicitly account for creep effects in long-term service, components subject to non-negligible creep, as determined by screening per ASME BPVC III-5, mandatory appendix HCB-III, will be evaluated using the Class A rules in ASME BPVC III-5 or ASME code case 2843-3. The staff notes that code case 2843-3 provides elevated temperature rules similar to ASME BPVC III-5 Class A rules, including fatigue and creep-fatigue assessment.
- For SPS-P MPS modules pressure boundaries, allowable stresses for elevated temperature are limited to ASME BPVC III-5, extended to 500,000 hours service time (see SE section 7.1.2.1.1) and the delta ferrite limits for weld material in service above 800°F will meet the requirements in ASME BPVC III-5, HCB-2433, and RG 1.31.
- For the SPS-P cold trap and the SPS-P cesium trap, delta ferrite limits for weld material in service below 800°F will meet the requirements in ASME BPVC III-1, NC-2433, and RG 1.31.
- For SPS-P MPS module pressure boundaries, the SPS-P cold trap, and the SPS-P cesium trap, fatigue analysis will be performed using the rules in ASME BPVC VIII-2. All pressure boundary welds will be full penetration welds with 100% volumetric examination to confirm weld quality. To provide assurance of material properties, all pressure boundary materials will require a CMTR.
- ASME BPVC VIII-1 rules for lethal service will be applied for liquid sodium-containing components designed to ASME BPVC VIII-1.

The staff notes that the identified special treatments will provide increased assurance of appropriate design margin and fabrication quality. These treatments incorporate more rigorous design considerations for creep and fatigue effects, as well as improved weld design and examination to ensure weld quality. Based on the identified special treatments, the staff determined that the preliminary design of SPS SSCs classified as NSRST and constructed to ASME B31.1-2022 and ASME BPVC VIII-1 is consistent with PDC 1. While appendix 14.2 of NAT-13478 does not address all NSRST SSCs in the SPS, such as other SPS-P SSCs in PSAR table 7.2.4-1 and SPS-I and SPS-E SSCs, the staff will review the design and

construction codes and standards for all NSRST SSCs with the final design. The staff expects a similar approach will be applied for special treatments of NSRST SSCs not addressed in appendix 14.2 of NAT-13478. As such, the staff focused its review on the most safety significant NSRST SSCs and determined that further information can reasonably be left for consideration at the OL stage.

For SPS-P seismic qualification, the PSAR states that the SPS-P pump breakers are designed to ASME QME-1-2023 and IEEE 344-2020. The PSAR states partial conformance with RG 1.100, which endorses IEEE 344-2013 and QME-1-2017. The staff notes that the NRC has not yet endorsed ASME QME-1-2023 or IEEE 344-2020. However, the staff reviewed ASME QME-1-2023 and IEEE 344-2020 and considers them appropriate if implemented consistent with the regulatory positions specified in Rev. 4 of RG 1.100. The PSAR further states that these codes will be applied to NSRST SPS SSCs in accordance with their safety significance. The PSAR additionally states that the SPS-P pump breakers are designed to IEEE 603-1991, stating full conformance with RG 1.153, which endorses the use of IEEE 603-1991.

Based on its review described above, the staff determined that the preliminary SPS design is consistent with PDC 1 because the codes and standards specified for the design and construction of the SPS SSCs are acceptable.

7.2.4.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.2.4.2.1 states that the SPS is designed to meet this PDC. Safety-significant SPS SSCs are designed to a hazard level commensurate with their safety-significance. PSAR section 7.2.4.1 states that the SPS is protected from design-basis hazards by the RAB, RXB, and FHB. These structures provide protection from external flooding, while the safety-significant SPS SSCs' location below grade provides protection from other natural phenomena including tornadoes and extreme climate conditions. The RXB and FHB substructures are SR, while the RAB substructure is NSRST.

PSAR section 7.2.4.2.2 classifies the SR SPS SSCs as SCS1, qualified to withstand seismic loads from the SSE. NSRST SSCs are classified as SCN1 and have seismic special treatments to ensure risk-significant functions or DID can be performed. Seismic classifications, design, analysis, and qualification are described in PSAR section 6.4 and evaluated in SE section 6.4. Additionally, PSAR section 6.1.3.1 will outline the methodology to fulfill seismic interaction requirements relating to all sodium-containing piping in the SPS, including NST portions of the system.

The staff determined that the SPS design ensures safety-significant SSCs can perform their safety functions despite natural phenomena because safety-significant components are housed below grade in the RXB, RAB, and FHB and are designed to withstand seismic events. As such, the staff determined that the preliminary SCG design is consistent with PDC 2.

7.2.4.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.2.4.2.1 states that all sodium-containing SPS SSCs are designed using materials that are non-combustible and non-fire sustaining. Based on

the information in PSAR section 7.2.4.2.1 and the staff's evaluation of the applicant's approach to fire protection for the SPS in SE sections 7.5.2 and 8.3, the staff determined the preliminary SPS design is consistent with PDC 3.

7.2.4.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.2.4.2.1 states that safety-significant SPS SSCs are protected from environmental conditions, dynamic transients, and chemical accidents based on their location within the RXB, RAB, and FHB. As such, the staff determined that the preliminary SPS design is adequately protected from external hazards, consistent with PDC 4.

The staff's evaluation of the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including SPS SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Based on the identified R&D item for ensuring materials performance and the staff's evaluation of environmental effects on safety-significant, long-lived passive SSCs in contact with sodium or cover gas in SE section 7.1.2.1.4, the staff determined that the preliminary design information of SPS SSCs in contact with sodium or cover gas is consistent with PDC 4 with respect to being designed to accommodate environmental conditions associated with normal operations and postulated accidents. The staff will evaluate the maturation of the RIM DMA and associated MANDE at the OL stage.

7.2.4.1.5 Primary Coolant Boundary (PDC 14, 15, 30, 31, and 32)

PDC 14 requires primary coolant boundary components to be designed, fabricated, erected, and tested to provide an extremely low probability of leakage, rapidly propagating failure, and gross rupture. PDC 15 requires the design conditions of the primary coolant boundary to not be exceeded during conditions of normal operation, including AOOs. PDC 30 requires primary coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 31 requires primary coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 32 requires primary coolant boundary components to permit periodic inspection and functional testing.

PSAR section 7.2.4.2.1 describes the portions of the SPS comprising the primary coolant boundary and states that these components will be designed and constructed to ASME BPVC III-5. The RIM program, described in PSAR chapter 8, is used to assess degradation mechanisms and to identify appropriate MANDE requirements for SPS primary coolant boundary components.

The staff's evaluation of the preliminary design information addressing environmental effects on safety-significant SSCs, including SPS primary coolant boundary SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Additionally, the staff's evaluation of the RES primary coolant boundary components in SE section 7.1.2.1.6 also encompasses SPS primary coolant boundary components. The staff evaluation of the RIM program is provided in SE section 8.1.

PSAR chapter 13 and section 12.1 of NAT-13478 describe an R&D item focused on assuring adequate structural materials performance in service. This R&D item also supports the RIM program through the development of performance monitoring methods, such as material surveillance programs, inspection methods and means of access, and monitoring approaches, as well as the development of PFM and LBB methodologies. The staff proposed two conditions for the KU1 CP related to updates on this R&D item as discussed in greater detail in SE chapter 13. These proposed permit conditions are discussed further in SE chapter 13.

Based on the staff evaluation of material compatibility of primary coolant boundary components in SE section 7.1.2.1.4, conformance with ASME BPVC III-5 and RG 1.87 for the design and construction of these components, and the proposed implementation of the RIM program and the associated R&D activities for assuring adequate material performance in service, the staff determined that the preliminary design of the SPS primary coolant boundary components is consistent with PDC 14, 15, 30, 31, and 32.

7.2.4.1.6 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. PSAR section 7.2.4.2.1 describes the applicant's functional containment strategy for the SPS. The PSAR states that SPS-P primary coolant boundary components are designed to ASME BPVC III-5 and all SPS-P piping up to the first isolation valve utilize secondary containments. The SR means to prevent continuous sodium release from the SPS is the SR SPS pump trip and the SPS siphon break, which are evaluated in SE sections 7.2.4.1.14 and 7.2.4.1.7.

The PSAR also identifies NSRST functions for DID to limit sodium release, including providing for the isolation of the SPS-P supply and tripping of the SPS pumps following a leak. PSAR section 7.2.4.2.1 states that the SPS is monitored for leakage, and PSAR section 8.2.1.8 further states that leak detection is provided at multiple points between pipe supports along all normally sodium-containing systems. PSAR section 7.2.4.2.1 also states that barriers around the SPS-P cells and SPS-I cold traps are part of functional containment while the SPS-E piping up to and including its first isolation valves are part of the EVST radionuclide barrier. These safety-significant functions to support functional containment are discussed in greater detail in SE section 7.2.4.1.14.

The staff notes that functional containment performance is assessed through the mechanistic source term analyses described in PSAR section 3.2 and SE section 3.2, which at the CP stage rely on assumptions regarding leakage rates across barriers. The staff will review leakage rate justifications at the OL stage relative to the design of the SSCs comprising each functional containment boundary.

The staff determined that the preliminary SPS design is consistent with PDC 16 because it includes multiple barriers and isolation functions to minimize leakage and there are plans for leakage monitoring that can justify leakage assumptions in the final design.

7.2.4.1.7 Primary Coolant Inventory (PDC 33)

PDC 33 requires a system to maintain primary coolant inventory to protect against small breaks in the primary coolant boundary as necessary to ensure SARRDLs are not exceeded. PSAR section 7.2.4.2.1 states that the safety-significant sections of the SPS-P are designed to ASME

BPVC III-5 and are included in the EQ program. The PSAR further states that coolant inventory is protected by the SR RPS and the NSRST AMC, which can both trip the SPS-P pump independently. The PSAR states that tripping the SPS-P pumps is sufficient to stop the sodium leak, noting that the SPS-P is located above the primary coolant level to prevent siphoning that could lower reactor level below a safe threshold. Based on SPS design features such as the SPS-P trips and the siphon break, the staff determined that the preliminary SPS design is consistent with PDC 33.

7.2.4.1.8 Sodium and Cover Gas Purity Control (PDC 71, 10 CFR 50.34(f)(1)(xii), 10 CFR 50.34(f)(2)(viii))

PDC 71 requires systems to maintain the purity of primary coolant sodium within specified design limits. PSAR table 1.4-4 states that the KU1 design is in partial compliance with 50.34(f)(2)(viii) and full compliance with 50.34(f)(1)(xii).

PSAR section 7.2.4.3 describes how the SPS can be operated continuously to monitor and remove contaminants from sodium. Sodium purity is monitored using the PTI and MPS. The PTI monitors impurities based on a correlation between the sodium temperature and the solubility of oxygen and oxides in sodium. The MPS is used to periodically sample primary sodium and measure its purity, allowing comparison against the PTI. The SPS-P also has installed meters to measure hydrogen. The PSAR states that the SPS contains cold traps, which remove impurities, including oxygen, hydrogen, and tritium. The SPS-P additionally contains cesium traps to remove cesium and particulate filters to remove insoluble particulates. The staff notes that these design features collectively provide an ability to monitor and maintain sodium coolant purity.

Table 4-5 of NAT-13478 identifies sodium chemistry impurity targets intended to limit corrosion. These targets cover various elements including oxygen, hydrogen, carbon, potassium, calcium, silicon, chlorides and bromides, lithium, and boron. NAT-13478 states that oxygen concentration and sodium temperature are the primary drivers of corrosion rates in austenitic stainless steels. Accordingly, the KU1 design targets a low oxygen environment of less than 2 parts per million by weight (ppmw) during operating, shutdown, and refueling conditions. The staff notes that table 7-1 of NAT-13478 provides preliminary screening criteria for corrosion in a sodium environment, including maintaining oxygen below 2 ppmw. The staff also notes that the carburization and decarburization mechanisms are significantly impacted by carbon levels in the system. The staff will review the final KU1 design, including the SPS for sodium impurity controls and the RIM program for ensuring passive SSC performance, at the OL stage. Additional evaluation of the RIM DMA is in SE section 7.1.2.1.4.

The staff determined the preliminary design of the SPS is consistent with PDC 71 because the SPS provides the ability to monitor and maintain sodium coolant purity within specified design limits.

Paragraph 50.34(f)(2)(viii) of 10 CFR requires the capability to obtain and analyze reactor coolant and containment samples for key radionuclides and chemical indicators of core damage such as noble gases, iodine, cesium, hydrogen, chloride, and boron, without exposing any individual to more than 5 rem whole-body or 50 rem extremity radiation doses. Consistent with PSAR table 1.4-4, the SPS enables samples to be obtained and analyzed within these exposure limits. PSAR table 1.4-4 also indicates that the KU1 design is in partial compliance with 10 CFR 50.34(f)(2)(viii) since the design will not obtain and analyze for hydrogen in the

containment atmosphere or dissolved gases or chloride in the coolant as these are not applicable to the KU1 design.

The staff determined that the SPS can monitor required core damage indicators through the process radiation monitors described in PSAR section 7.2.4.3.3 and the MPS. The process radiation monitors consist of two gamma spectrometers to identify and measure radioactive isotopes in the primary sodium while the MPS enables coolant sampling. The staff notes that the SCG also allows for cover gas to be monitored for radionuclides. Hydrogen buildup in the spaces which make up portions of the functional containment is non-credible, as sodium does not react with the fuel or cladding to produce hydrogen (unlike high temperature zircaloy-water reactions in LWRs) and there are no other credible sources of significant amounts of hydrogen. As such, the staff determined that the preliminary design is consistent with 50.34(f)(2)(viii).

Paragraph 50.34(f)(1)(xii) of 10 CFR requires an evaluation of alternative hydrogen control systems. This requirement is focused on mitigating hydrogen build-up following a loss of coolant accident in a LWR plant. As stated previously, the staff determined that the build-up of hydrogen is not credible for the KU1 design. The SPS-P allows for the measurement and removal of hydrogen. As such, the staff determined the preliminary design is consistent with 10 CFR 50.34(f)(1)(xii).

7.2.4.1.9 Intermediate Coolant Boundary (PDC 70, 75, 76, and 77)

PDC 70 requires the intermediate coolant system to be designed with sufficient margin to ensure that the intermediate coolant boundary maintains integrity during normal operation and AOOs, and that the integrity of the primary coolant boundary shall be maintained during postulated accidents. PDC 75 requires intermediate coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 76 requires intermediate coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 77 requires intermediate coolant boundary components to permit periodic inspection and functional testing.

PSAR section 7.2.4.2.1 states that the portions of the SPS-I piping between the isolation valves and the IHT form part of the intermediate coolant boundary and is designed to ensure that design conditions are not exceeded during normal operation, including AOOs. PSAR section 7.2.4.4 states that SPS intermediate coolant boundary SSCs are designed and constructed in accordance with ASME B31.1-2022 and ASME BPVC VIII. The staff evaluated the use of industrial codes including ASME B31.3-2020 and ASME BPVC VIII in SE section 6.4.1.5. Based on the staff evaluation in SE sections 6.4.1.5 and 7.2.4.1.1, the staff determined that the use of ASME B31.1-2022 and ASME BPVC VIII for the design and construction of SPS intermediate coolant boundary components is consistent with PDC 70 and 75. While appendix 14.2 of NAT-13478 does not address all NSRST SSCs in the SPS, such as those in the SPS-I, the staff will review the design and construction codes and standards for all NSRST SSCs with the final design. The staff expects a similar approach will be applied for special treatments of NSRST SSCs not addressed in appendix 14.2 of NAT-13478. As such, the staff focused its review on the most safety significant NSRST SSCs and determined that further information can reasonably be left for later consideration.

Table 5.2 of NAT-13478 states that SPS intermediate coolant boundary components are constructed from austenitic stainless steel, either 304H or 316H with ER16-8-2 weld metal for high temperatures applications, or 304 or 316 for low temperature service. The staff's evaluation

of the preliminary design information addressing environmental effects on safety-significant SSCs, including SPS intermediate coolant boundary SSCs, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. The RIM program, described in PSAR chapter 8, is used to assess degradation mechanisms and identify appropriate MANDE inspection requirements for SPS intermediate coolant boundary components. The staff evaluation of the RIM program is documented in SE section 8.1. Based on the inclusion of SPS intermediate coolant boundary components in the RIM program and the staff's evaluation of the RIM program and the supporting R&D activities documented in SE sections 7.1.2.1.4 and 8.1, the staff determined that the preliminary information for the design of SPS intermediate coolant boundary components is consistent with PDC 76 and 77. The staff will evaluate the maturation of the RIM DMA and associated MANDE through the prescribed R&D updates as well as with the final design at the OL stage.

7.2.4.1.10 Sodium Heating System (PDC 72)

PDC 72 requires heating systems for safety-significant SSCs that contain sodium. PSAR section 7.2.4.2.1 states that electric heat tracing is used to maintain SPS sodium temperature within design limits, with sufficient margin to prevent exceeding the thermal design limits of the SR portions of the SPS in the event of a single failure. The PSAR further states that some portions of SPS-P and SPS-E piping will have limited access and therefore use redundant heat trace to improve reliability. The applicant also states that sodium freezing in this piping will not damage the primary coolant boundary and thus the heat tracing in this section is not SR. Based on the inclusion of electric heat tracing in the SPS piping and the consideration of how potential freezing could impact the primary coolant boundary, the staff determined that the preliminary SPS design is consistent with PDC 72.

7.2.4.1.11 Sodium Leakage Detection and Reaction Prevention and Mitigation (PDC 73)

PDC 73 requires a means to detect sodium leakage and to limit and control sodium-air and sodium-concrete reactions. PSAR section 7.2.4.2 states that the design of the SPS provides means to detect sodium leakage, limit and control sodium-air and sodium-concrete reactions, and mitigate effects of fires from sodium-air reactions. PSAR section 7.5.2 discusses sodium leakage detection, collection, and fire mitigation and PSAR section 8.2 describes the sodium fire protection design. Based on the information in PSAR section 7.2.4.2 and the staff evaluation of PSAR sections 7.5.2 and 8.2 in SE sections 7.5.2 and 8.3, respectively, the staff determined that the preliminary design for sodium leakage detection and reaction prevention and mitigation, which includes the SPS, is consistent with PDC 73.

7.2.4.1.12 Sodium-Water Reaction Prevention (PDC 74)

PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water. PSAR section 8.2.1 states that there are no water-containing systems within the RXB and RAB due to the presence of sodium in these buildings and that there are no fire water suppression systems in areas of the FHB with sodium-containing systems. Additionally, PSAR section 7.2.4.2 states that major components of SPS-P are located in inerted vaults and inert cover gas limits contact between the sodium and the moisture from the air. Based on the lack of water sources in the RXB, RAB, and areas of the FHB with sodium-containing systems, the staff determined that the preliminary design of the SPS is consistent with PDC 74.

7.2.4.1.13 Primary Coolant System Interfaces (PDC 78)

PDC 78 provides requirements for SSCs containing liquids which interface with the primary coolant system. PDC 78 requires two passive barriers between primary coolant and a chemically incompatible fluid. For a fluid that is chemically compatible with the primary coolant, one passive barrier is allowed if the postulated leakage at the interface does not result in failure of the intended safety functions of safety-significant SSCs or result in exceeding SARRDLs and if the fluid is maintained at a higher pressure than the primary coolant during normal operation, AOOs, shutdown, and accident conditions.

PSAR section 7.2.4.2.1 states that the SPS-P cold traps and cells act as single passive barriers with nitrogen gas as the interfacing fluid. The section further states that nitrogen is chemically compatible with the sodium coolant. The PSAR states that the cold traps and cells are isolable from the primary coolant system to limit the loss of primary system inventory resulting from postulated leakage. The PSAR further states that a pressure differential between the nitrogen and sodium is not required as leakage across the interface does not result in failure of SPS SR and NSRST functions. The SPS cold traps and cells that incorporate a passive barrier with a compatible fluid (nitrogen gas) are isolable to limit inventory loss from leakage and are designed such that leakage does not result in failure of intended safety-significant functions. Because of this, the staff determined that the preliminary design of the SPS is consistent with PDC 78.

7.2.4.1.14 Safety Functions

7.2.4.1.14.1 SPS Pump Trip Functions

PSAR section 7.2.4.2.2 states that the SPS supports several trip functions to secure the SPS pumps. These are:

- DL3-HR11, SPS Pump Trip on Low Primary Sodium Level (SR)
- DL4-RR3b, SPS Pump Trip on Leak Detection (NSRST)
- DL4-RR8, Manual SPS Pump Trip on Low Primary Sodium Level (NSRST)

PSAR section 7.2.4.2.2 states that DL3-HR11 requires that the SPS-P pump trips on low primary sodium level in the hot pool. PSAR table 5.2-2 states this trip maintains primary sodium level in the RV high enough to provide reasonable assurance that natural circulation can develop in the PHT. DL3-HR11 is automatically initiated by the RPS, described in PSAR section 7.6.3 and evaluated in SE section 7.6.3, which receives hot pool level signals from the RIS. The PSAR states that tripping the SPS-P pumps is sufficient to stop the sodium leak, noting that the SPS-P is located above the primary coolant level to prevent siphoning that could lower reactor level below a safe threshold. The staff determined that the preliminary SPS design is consistent with this function as it relies on SR instrumentation in the RIS and the SR RPS for initiation, and it prevents the siphoning of primary sodium from the reactor. However, the staff notes that the setpoint for this trip is yet to be determined and will be evaluated by the staff during the OL application review to ensure it supports the development of natural circulation.

The PSAR states that DL4-RR3b requires that the SPS-P pumps trip on detection of a sodium leak within the SPS-P to limit the loss of primary sodium and the severity of the leak. This trip is automatically initiated by the NSRST AMC, described in PSAR section 7.6.3 and evaluated in SE section 7.6.2, that receives leak detection signals from the NSRST sodium leak detection, collection, and containment system (NNA). The staff's evaluation of the sodium leak detection

methods is provided in SE section 8.3. PSAR table 5.2-4 states that DL4-RR3b is required for DID adequacy. The staff determined that the preliminary SPS design is consistent with this function as this SPS-P trip operates independently from RPS and relies on alternate indications compared to DL3-HR11.

PSAR section 7.2.4.2.2 also states that DL4-RR8 allows for manual initiation of an SPS-P pump trip, providing an alternate means to DL3-HR11 for securing the SPS-P pumps and maintaining primary sodium level in the RV. PSAR section 7.6.3 describes how the RPS system supports this trip with circuitry that is diverse from the SR automatic SPS-P trip. As such, the staff determined the SPS design is consistent with this function.

7.2.4.1.14.2 SPS Leak Isolation Functions

PSAR section 7.2.4.2.2 states that the SPS supports two leak detection functions that preserve the flow of sodium and preserve sodium inventory. Those functions are:

- DL4-RR3a, SPS Supply Valve Isolation on Leak Detection (NSRST)
- DL4-RR3c, SPS Cell Barrier Isolation on Leak Detection (NSRST)

PSAR section 7.2.4.2.2 states that DL4-RR3a requires the SPS-P supply isolation valve to automatically close upon leak detection. This function is initiated by the NSRST AMC, which receives SPS-P leak detection signals from the NSRST NNA. The PSAR states that this function mitigates the severity of an SPS-P leak by limiting flow from the PHT. The PSAR further states DL4-RR3c requires automatic isolation of SPS-P cell ventilation upon leak detection. This function is initiated by the NSRST UMC, described in PSAR section 7.6.3 and evaluated in SE section 7.6.2, that receives leak detection signals from the NSRST NNA. Upon detection of a sodium leak, UMC automatically closes the NHV dampers, isolating the SPS-P cell. The staff's evaluation of the design's sodium leak detection methods is provided in SE section 8.3. PSAR table 5.2-4 states that DL4-RR3a and DL4-RR3c are required for DID adequacy.

The staff determined that the preliminary SPS design is consistent with these functions because they operate independently from the RPS and rely on alternate indications compared to DL3-HR11, the SR means of stopping a SPS-P leak.

7.2.4.1.14.3 Radionuclide Retention Functions

PSAR section 7.2.4.2.2 states that the SPS supports five functions that provide radionuclide retention as part of functional containment. These are:

- DL3-RR1b, SPS Primary System Barrier (SR)
- DL2-RR7, Primary SPS Barrier (NSRST)
- DL4-RR3, SPS Cells Barrier (NSRST)
- DL2-RR8, Intermediate Cold Trap SPS Barrier (NSRST)
- DL3-RR4, EVST Barrier (SR)

DL3-RR1b specifies that the SPS provides a portion of the primary coolant boundary and acts as a low leakage barrier to support radionuclide retention. PSAR section 7.2.4.2.2 states that this section consists of the SPS-P from the RES to the SPS primary isolation valves, excluding the closure of these valves. The staff notes that the SR means of stopping an SPS-P leak is through the securing of the SPS-P pumps (DL3-HR11) rather than through isolation valve closure. Based on the staff's evaluation of the SPS-P pump trips, the evaluation of material compatibility of the primary coolant boundary components in SE section 7.1.2.1.4, the evaluation of the PDC related to the primary coolant boundary in SE section 7.2.4.1.5, and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in-service material performance, the staff determined that the preliminary SPS design is consistent with this function.

PSAR table 5.2-4 states that DL2-RR7 and DL2-RR8 are required for DID adequacy, with no associated LBEs.

DL2-RR7 requires that the SPS-P provides a passive barrier for portions of the SPS that contain primary sodium but are not part of the primary coolant boundary. SE section 7.2.4.1.1 evaluates the design and construction codes, standards, and special treatments applied to SPS-P SSCs located outside the primary coolant boundary. SE section 7.1.2.1.4 evaluates the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including SPS SSCs, in contact with sodium or cover gas containing sodium aerosols. Based on the evaluations in SE section 7.2.4.1.1 and 7.1.2.1.4 regarding the design, construction, and environmental considerations for portions of the SPS-P that contain primary sodium but are not part of the primary coolant boundary, the staff determined that the preliminary SPS design is consistent with this function.

DL2-RR8 requires that the SPS-I cold trap acts as a barrier to retain sodium. SE section 7.2.4.1.1 evaluates the design and construction codes and standards and special treatments applied to SPS SSCs and notes that information on special treatments for the SPS-I cold trap was not provided at the CP stage. While appendix 14.2 of NAT-13478 does not address all NSRST SSCs in the SPS, such as those in the SPS-I, the staff will review the design and construction codes and standards for all NSRST SSCs during the review of the final design. The staff expects a similar approach will be applied for special treatments of NSRST SSCs not addressed in appendix 14.2 of NAT-13478. As such the staff focused its review on the most safety significant NSRST SSCs and determined that further information can reasonably be left for later consideration. SE section 7.1.2.1.4 evaluates the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including the SPS-I cold trap, in contact with sodium or cover gas containing sodium aerosols. Based on the evaluations in SE sections 7.2.4.1.1 and 7.1.2.1.4 regarding the design, construction, and environmental considerations that apply to the SPS-I cold trap, the staff determined that the preliminary SPS design is consistent with this function.

DL4-RR3 requires that the SPS-P cell enclosures act as a low leakage barrier that contains radionuclides in the event of a leak. The preliminary design includes limited design information on the SPS cell enclosures, which are identified as being part of the RAB substructure in PSAR section 7.8 and discussed in SE section 7.8. The staff will review the final design of the SPS cells at the OL stage. The DL4-RR3 function and its associated LBE are discussed in greater detail in SE sections 3.2.1, 3.6.3.3, 3.8.3.4, and 5.5. Based on the evaluation of this function in these sections, the staff determined that the preliminary SPS design is consistent with DL4-RR3 and further information can reasonably be left for later consideration.

DL3-RR4 specifies that the SPS-E supports radionuclide retention in the EVST via providing a barrier through its SSCs. This includes SSCs exiting the EVST, including isolation valves, but does not include isolation valve closure. The staff notes that the SPS-E EVST piping and isolation valves support this function and will be designed to ASME BPVC III-5 and in the scope of the RIM program. The staff will review the final design to confirm the final design and operation of these SSCs ensures this function is met. Further discussion of the DL3-RR4 can be found in SE section 7.3.2.

Based on the evaluation above, the staff determined that the preliminary SPS design is consistent with DL3-RR4, DL3-RR1b, DL2-RR7, DL4-RR3, and DL2-RR8.

7.2.4.1.14.4 SPS Heat Removal Function

PSAR section 7.2.4.2.2 states that the SPS supports NSRST function, DL4-HR1, "IAC Passive Mode Operation," which specifies that the IAC shall transfer heat from the IHT to the atmosphere via natural convection airflow for LBEs when the ISP and RAC are unavailable. The PSAR states that the SPS-I piping between the IHT and the SPS-I isolation valves are designed to prevent leakage or flow disruptions to ensure IAC heat removal remains available. Based on the staff's evaluation of the PDC related to the intermediate coolant boundary in SE section 7.2.4.1.9 and the proposed implementation of the RIM program and associated R&D activities to ensure adequate in-service material performance, the staff determined that the preliminary SPS design is consistent with this function.

While appendix 14.2 of NAT-13478 does not address all NSRST SSCs in the SPS, such as those in the SPS-I, the staff will review the design and construction codes and standards for all NSRST SSCs with the final design. The staff expects a similar approach will be applied for special treatments of NSRST SSCs not addressed in appendix 14.2 of NAT-13478. As such the staff focused its review on the most safety significant NSRST SSCs and determined that further information can reasonably be left for later consideration.

7.2.4.1.14.5 PAM Function

PSAR section 7.2.4.2.2 states the SPS supports NSRST function, DL5-PAM1, "Post-Accident Monitoring." The applicant states that the SPS contributes to PAM by providing valve position indication for the SPS-P isolation valves and breaker position indication for the SPS-P pump breakers. The staff notes that these variables enable an operator to determine whether a leak has been secured, either through the tripping of the SPS-P pumps or the isolation of the system. PSAR section 7.6.3 further describes the PAM design, stating that it partially conforms to IEEE 497-2016, which is endorsed in RG 1.97. Based on adherence to IEEE 497-2016 and the selection of variables that enable the operator to determine whether a leak has been isolated, the staff determined that the preliminary SPS-P design is consistent with this function. The staff will review selected PAM variables and the extent of conformance with RG 1.97 at the OL stage.

7.2.4.1.15 Programmatic Special Treatments

Programmatic special treatments applied to the SPS are summarized in PSAR section 7.2.4.4 and include the QAP, D-RAP, EQ program, IST, PITAP, and RIM program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the SPS contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to SPS. Plant programs are evaluated further in SE chapter 8.

7.2.4.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.2.4 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.2.5 Control Rod Drive System

PSAR section 7.2.5 describes the CRD. The primary purpose of the CRD is to support normal reactivity control and reactor scram functions by positioning CRAs within the core. Portions of the CRD also form part of the primary coolant boundary.

There are thirteen control rod drives, all located on top of the reactor head, in the HAA. Each drive is connected to a control rod driveline, which in turn connects to a CRA through a pneumatically-controlled latching mechanism. This mechanism enables remote coupling and decoupling of the CRA from the driveline. The pressure holding the gripper engaged with the CRA is provided by a scram valve and cylinder assembly located within the CRDM housing, which is pressurized by the SCG. The SCG also provides overpressure protection for the CRDM housing. Actuation of the scram valve discharges the scram cylinder contents into the CRDM housing, which is kept at a significantly lower pressure. This pressure differential disengages the latch, allowing the CRA to fall into the core by gravity.

For normal control of reactivity, each CRDM uses a motor coupled to a ball screw assembly to move the driveline vertically. This motion raises or lowers the connected CRA within a duct and guide tube in the core. The control assembly duct is similar to the fuel ducts described in SE section 7.1.1. The CRA design, including both primary and secondary CRAs with differing geometric arrangements, is also discussed in SE section 7.1.1 and in NAT-2806-A, which is incorporated by reference into the PSAR in section 3.11.

The scram valves are actuated by the RPS, as discussed in SE section 7.6.3, upon receipt of a scram signal. CRD motor control is provided by the rod monitoring and control system (RMC), a subset of the NIC described in SE section 7.6.2. Each CRA is controlled individually and software interlocks within the control system ensure that only one CRA can be withdrawn at a time. A scram signal also initiates a motor-driven scram follow, during which the CRDM pushes the CRAs into the core at maximum speed.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the CRD:

- RG 1.20, Rev. 4;
- RG 1.31, Rev. 4;
- RG 1.44, Rev. 1;
- RG 1.87, Rev. 2; and
- RG 1.100, Rev. 4.

PSAR sections 7.2.5 and 5.3 identify the following PDC as applicable to the CRD: 1, 2, 3, 4, 12, 14, 15, 16, 26, 28, 29, 30, 31, 32, and 80.

7.2.5.1 Technical Evaluation

7.2.5.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR table 7.2.5-2 identifies the design and construction codes applicable to the CRD. PSAR section 7.2.5.2 identifies the CRD system's conformance to associated regulatory requirements and guidance.

The CRD primary coolant boundary components are designed to the requirements of ASME BPVC III-5, which is endorsed in RG 1.87 and is therefore acceptable for high-temperature metallic components. PSAR section 7.2.5.2 states that the CRD is in full conformance with RG 1.87.

PSAR table 7.2.5-4 and table 5-2 of NAT-13478 identify alloy 718 as a material of construction for the CRD driveline assembly, which is not expected to require welding. The staff notes that alloy 718 is an approved material in ASME BPVC III-5 for bolting applications only, which does not include its planned use in the CRD driveline assembly. Though the staff does not consider the failure of a gripper to be a major safety concern because it would be most likely to result in inadvertent insertion of the attached control rod assembly, the staff notes that the reactivity control function of the CRD is safety-related and must be performed reliably. Given that alloy 718 is already approved in ASME BPVC III-5 for bolting applications, the staff determined it is reasonable to leave for later consideration the development of a detailed design and analysis methodology for alloy 718 under ASME BPVC III-5 for its use in the final CRD driveline assembly design. The staff will review further details on the applicant's approach to the design and construction of the CRD driveline assembly using alloy 718 consistent with ASME BPVC III-5 at the OL stage.

For EQ, ASME QME-1-2023 or IEEE 323-2003 are applied to the SR mechanical and electrical CRDM and CRD driveline assemblies. PSAR table 1.4-1 indicates that IEEE 323-2003 is used to qualify electrical equipment in a mild environment, as endorsed by RG 1.209. QME-1 is used to qualify active mechanical equipment. The staff notes that the NRC has not yet endorsed ASME QME-1-2023. However, the staff reviewed ASME QME-1-2023 and considers it appropriate if implemented consistent with the regulatory positions specified in Rev. 4 of RG 1.100. Because these standards are endorsed by the NRC and are being used in accordance with the terms of their endorsement, the NRC determined that they are acceptable.

Based on its review described above, the staff determined that the preliminary CRD design is consistent with PDC 1 because the codes and standards specified for the design and construction of the CRD SSCs are acceptable.

7.2.5.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.2.5.2 states that the CRD is designed to withstand natural phenomena without loss of capability to

perform its safety-significant functions. Design basis hazards are discussed further in SE chapter 2, while methods applied to evaluate these hazards are discussed in chapter 6.

The CRD is located in the RXB substructure, except for the CRDM motor controllers, which are located in the NCB substructure. These SR structures provide protection from external flooding, while the CRD's location below grade provides protection from other natural phenomena including tornadoes and extreme climate conditions. The CRD is designed to withstand earthquakes, consistent with the methods described in PSAR section 6.4.1. Seismic classification of the CRD components is provided in PSAR section 7.2.5.2, which states that SR components are designated SCS1, while NSRST components are designated SCN1. PSAR table 7.2.5-1 indicates that the motor drive assemblies and controllers are NSRST.

The staff determined that the CRD system is consistent with PDC 2 because it is protected from external hazards by the RXB and NCB substructures and is appropriately designed to withstand seismic events while maintaining its capability to perform safety-significant functions.

7.2.5.1.3 Fire Protection (PDC 3)

PDC 3 requires that safety-significant SSCs are designed and located to minimize the probability and effects of fires and explosions. The PSAR states that the CRD is designed to use non-combustible or fire-resistant materials to the extent practical. The staff determined that this approach is reasonable for a CP application, when detailed design information is not available. However, more detailed information on CRD materials should be provided at the OL stage to ensure the CRD is appropriately resistant to fire.

One potential source of fire associated with the CRD is sodium-air interaction. The applicant states that this risk is mitigated through use of bellows, leak detection, and inert gas supplied by the SCG. The SCG, evaluated in SE section 7.2.3, provides a continuous purge flow directed towards the cover gas space, as discussed in PSAR section 7.2.5.1. Bellows serve to limit the ingress of sodium cover gas, aerosolized sodium, and radionuclides into the CRDM housing as described in PSAR section 7.2.5.3. PSAR section 7.2.5.1 also states that SCG flow is monitored to identify leakage.

Additional fire protection measures, including those used in the HAA where the CRDMs are located, are evaluated in SE sections 7.5.2 and 8.3. Because the CRD contains appropriate measures to minimize the probability and effect of fires and explosions, including fires resulting from sodium-air interactions, the staff determined that the preliminary design information on the CRD is consistent with PDC 3.

7.2.5.1.4 Environmental and Dynamic Effects (PDC 4)

PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, AOOs, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. PDC 4 also requires chemical consequences of sodium leaks to be considered.

The CRD contains components located on the RH and inside the RV that are exposed to high temperatures, liquid sodium, sodium aerosols, and potentially oxidation products. The PSAR states that materials selected for CRD primary coolant boundary components are consistent with ASME BPVC III-5, ensuring that these components are designed to accommodate their

operating conditions. CRD components are also qualified for their environments through the EQ program described in PSAR chapter 8. The staff's evaluation of the preliminary design information addressing environmental effects on safety-significant, long-lived passive SSCs, including CRD components, in contact with sodium or cover gas containing sodium aerosols is provided in SE section 7.1.2.1.4. Based on the identified R&D item for ensuring materials performance and the staff's evaluation of environmental effects on safety-significant, long-lived passive SSCs in contact with sodium or cover gas in SE section 7.1.2.1.4, the staff determined that the preliminary design information of CRD SSCs in contact with sodium or cover gas is consistent with PDC 4. The staff will evaluate the maturation of the RIM DMA and associated MANDE at the OL stage. The staff notes that the RIM program is expected to appropriately address environmental effects, including irradiation, on SSCs made of Alloy 718 in the CRD driveline assembly.

The RXB and NCB substructures provide protection from certain industrial and transportation hazards, described in PSAR section 2.3 and evaluated using the methods described in PSAR section 6.1.1. However, they do not provide protection from sodium fires that may occur in the HAA, which are instead addressed under other PDC and in PSAR sections 7.5.2 and 8.2.

The staff determined that the preliminary design of CRD is consistent with PDC 4 because the applicant uses appropriate design codes, EQ, and materials of construction to ensure the SSC can accommodate applicable environmental conditions, including high temperature, sodium aerosols, and oxidation products.

7.2.5.1.5 Suppression of Reactor Power Oscillations (PDC 12)

PDC 12 requires power oscillations that can result in conditions exceeding SARRDLs to either not be possible or be reliably and readily detected and suppressed. PSAR section 7.2.5.2 indicates that the role played by CRD for PDC 12 is to position CRAs to control oscillations or scram the reactor, as necessary. Preliminary stability analyses, discussed in PSAR section 3.12, indicate that no regions of instability were identified.

The staff determined that the CRD system is consistent with PDC 12 because it has the identified function to scram the reactor before setpoints are exceeded. The capability of the reactivity control system to perform this function is discussed further in SE section 7.2.5.1.11.

7.2.5.1.6 Primary Coolant Boundary (PDC 14, 15, 30, 31, and 32)

PDC 14 requires primary coolant boundary components to be designed, fabricated, erected, and tested to provide an extremely low probability of leakage, rapidly propagating failure, and gross rupture. PDC 15 requires the design conditions of the primary coolant boundary to not be exceeded during conditions of normal operation, including AOOs. PDC 30 requires primary coolant boundary components to be designed, fabricated, erected, and tested in accordance with quality standards commensurate with their safety significance. PDC 31 requires primary coolant boundary components to be designed with sufficient margin to ensure the boundary behaves in a non-brittle manner such that the probability of rapidly propagating fracture is minimized. PDC 32 requires primary coolant boundary components to permit periodic inspection and functional testing.

The applicant states that the primary coolant boundary components of the CRD will meet the design, construction, testing, and quality assurance requirements of ASME BPVC III-5, which ensures that appropriate service level conditions and bounding load combinations are

accounted for in design. Materials of the CRD primary coolant pressure boundary are selected for compatibility with a sodium environment with consideration given to service temperatures, service degradation of material properties, and other applicable material degradation mechanisms. Materials of construction for CRD components are provided in PSAR table 7.2.5-4. Material qualification considerations for primary coolant boundary components and other materials in contact with primary sodium are discussed in more detail in SE section 7.1.2. The RIM program, described in PSAR chapter 8, is used to identify degradation mechanisms and to define appropriate test and MANDE inspection requirements for primary coolant boundary components.

The staff determined the design of CRD is consistent with PDC 14, 15, 30, 31, and 32 because the design codes are appropriately selected based on the safety-significance of these components and the high temperature environment, the materials are selected consistent with the design code, material qualification is appropriately addressed as discussed in SE section 7.1.2, and appropriate test and inspection requirements are identified using the RIM program.

7.2.5.1.7 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. As part of the primary coolant boundary, the CRD forms a portion of the functional containment as discussed in PSAR section 1.3.2.1. Additionally, CRD components are designed to reduce radiation release from the RV into the HAA. A continuous argon purge flow from the SCG through the CRD prevents the flow of radionuclides up through the CRDM penetration.

The staff determined that the CRD design is consistent with PDC 16 because it is appropriately designed to mitigate release of radionuclides, consistent with its role as part of the functional containment. The primary coolant boundary aspects of the CRD are discussed in SE section 7.2.5.1.6 and the SCG is discussed in SE section 7.2.3.

7.2.5.1.8 Reactivity Control Systems (PDC 26)

PDC 26 requires a minimum of two reactivity control systems or means to provide the capability to: (1) insert negative reactivity at sufficient rate and amount to ensure SARRDLs are not exceeded and safe shutdown is achieved and maintained for AOOs; (2) control the rate of reactivity changes during planned, normal power changes to ensure SARRDLs are not exceeded; (3) cool the core, shut down the reactor, and maintain safe shutdown following a postulated accident; and (4) maintain the reactor shutdown during fuel loading, inspection and repair activities²⁴. Capability (2) is required to be independent and diverse from the others.

Compliance with PDC 26 is discussed in PSAR section 5.3.3. The PSAR states that the two separate means of inserting the CRAs (and thus performing the required reactivity control capabilities) are (a) the de-latching of the CRAs to allow them to insert by gravity, and (b) the control rod motor drive insertion. The PSAR notes that there are no common-cause failures between the de-latching system and the motor drive. The staff confirmed this for the preliminary design by auditing detailed design documents and PRA system models for the CRD.

²⁴ Equivalent to "cold shutdown" in operating LWRs.

However, a potential common cause mechanism that could cause CRAs to fail to insert into the core is binding between the CRA and its associated duct, which could result from seismic motion or significant duct or CRA distortions. This mechanism is mitigated by the use of two separate CRA designs, referred to as primary and secondary CRAs, with different geometric configurations. The two types of CRAs are described in further detail in SE section 7.1.1, and detailed descriptions and drawings were provided in TP-LIC-RPT-0011 and TR NAT-2806-A. The secondary CRAs are designed to mitigate the effects of rod-to-duct binding relative to the primary CRAs. Analyses demonstrating that deformations and seismic motions can be accommodated by the secondary CRAs were not provided, which is consistent with the design information expected to be available at the CP stage. The staff expects to evaluate these analyses at the OL stage when the design is complete and the analysis methodologies have been validated.

PSAR section 5.3.3 describes how each of the capabilities required by PDC 26 is provided.

- Capability (1) is provided by de-latching the primary and secondary CRAs, allowing them to fall into the core. Malfunctions are considered, as required by the PDC, including scenarios where the highest worth rod is stuck out of the core or where either the entire set of primary or secondary control rods fails to insert while the other set successfully inserts.
- Capability (2) is provided by the motor drive control, which can insert control rods at a rate that prevents the SARRDLs from being exceeded. Diversity is provided by the two separate sets of CRAs. PSAR section 7.1.1 indicates that secondary control rods also provide a regulating function to manage small reactivity changes.
- Capability (3) is provided by the same means as capability (1).
- Capability (4) is provided by the primary and secondary CRAs, as well as administrative controls.

The staff determined that the preliminary CRD design is consistent with PDC 26 because it provides separate gravity scram and motor drive-in functions and uses different primary and secondary CRA designs to mitigate common-cause failure mechanism and ensure sufficient independence and diversity. Additionally, past SFR operating experience shows that similar reactor designs have shut down with limited or no control rod insertion, even for DBA-type scenarios, due to strong inherent reactivity feedback mechanisms. While these mechanisms were not credited to comply with PDC 26, they provide additional confidence that final details of the reactivity control system can be reasonably left for later consideration.

The staff also considered potential failures in the protection system that could prevent control rod insertion. This is mitigated by the CRD driveline scram follow function (DL4-RC3), which is actuated by a separate signal from RPS than the scram, as well as by the alternative shunt trip (DL4-RC6), which provides an alternate means of tripping the reactor trip breakers. The adequacy of the protection and control systems to perform these functions is addressed in SE section 7.6.

7.2.5.1.9 Reactivity Limits (PDC 28)

PDC 28 requires limits on the amount and rate of reactivity increase to ensure the primary coolant boundary and core are not damaged. PSAR section 7.2.5.2 states that the CRD motor is sized to limit the maximum withdrawal speed and thus reactivity insertion rate. A control interlock also prevents the removal of more than one control rod at a time.

The staff audited details of the preliminary CRD design and the analyses involving control rod withdrawal presented in PSAR chapter 3, confirming that the assumed reactivity insertion rate is consistent with the physical design of the CRD motor. The chapter 3 analyses (those in the initiating event family RPD-CW1ACS) demonstrate that fuel acceptance criteria are maintained, thus confirming that the core (and therefore also the primary coolant boundary) are not damaged. Because of this, and because the CRD includes both physical and control system features that limit the amount and rate of reactivity increase, the staff determined that the preliminary CRD design is consistent with PDC 28.

7.2.5.1.10 Reactor Vessel and Reactor System Structural Design Basis (PDC 80)

PDC 80 requires, in part, that the RV and reactor system are designed to maintain their integrity during postulated accidents to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown. PSAR section 7.2.5.2 states that the integrity of the CRD is maintained during accidents to ensure that CRAs can be inserted into the reactor core. The CRDMs are mounted to and structurally supported by the rotatable plug assembly, as described in SE section 7.1.2. The integrity and insertability of the CRAs are discussed in further detail in SE section 7.1.1. For the rest of the CRD, ASME BPVC III-5 is identified as the applicable design code in PSAR table 7.2.5-2. Based on the use of an appropriate design code and the evaluations of CRA insertability, the staff determined that the design of CRD is consistent with PDC 80.

7.2.5.1.11 Safety Functions

To support the review of the safety-significant PRA safety functions in the context of the safety analysis discussed below, the staff audited the PRA and deterministic DBA models. The staff determined that the CRD models are consistent with their descriptions in the PSAR, with appropriate assumptions considering the preliminary state of the design.

7.2.5.1.11.1 Reactivity Control Functions

PSAR section 7.2.5.2 states that the CRD supports three reactivity control functions. These are:

- DL3-RC1 – Scram - Gravity Driven Absorber Insertion by Latch Release (SR)
- DL3-RC2 – Reactor Scram on Loss of Power (SR)
- DL4-RC3 – Control Rod Drive System Driveline Scram Follow (NSRST).

As described in PSAR table 5.2-1, DL3-RC1 is the function to release the latching mechanism and drop the control rods. The CRD supports DL3-RC1 by ensuring timely release of the CRAs to establish reasonable assurance that the radionuclide release from an accident is less than the 10 CFR 50.34 dose criteria at a safe shutdown condition. This function is a key aspect of the CRD design as discussed in this section. The supporting instrumentation and control (I&C) functions that initiate the scram at the appropriate time are addressed in SE section 7.6.

PSAR table 5.2-1 states that DL3-RC2 is the function that provides a reactor scram on loss of power. While the table uses the same language to describe the CRD's role in both DL3-RC1 and DL3-RC2, the staff considers DL3-RC2 to be more directly related to the design of the scram cylinder and valve assembly to provide the scram function on a loss of power. PSAR section 7.2.5.3 states that the CRD scram is a "fail-safe function that is performed by de-energizing the scram valves, which vent the scram cylinder and release the CRAs to insert by gravity into the reactor core." As discussed in PSAR section 7.6.3.2.2, the RPS maintains voltage on the reactor trip breaker (RTB) undervoltage coils to keep the RTBs closed, thereby maintaining power to the CRD solenoid operated valves. On a loss of power to the RPS, voltage to the RTB undervoltage coils is lost, causing the undervoltage release mechanism to open the RTBs, interrupting power to the CRD solenoid operated valve coils and allowing the scram cylinders to vent. RPS is discussed further in SE section 7.6.3.

All AOOs and DBEs discussed in PSAR chapter 3 that need reactivity control, except for those occurring from a low power or shutdown condition, rely on the gravity scram function to shut down the reactor. The local core faults DBE relies on a manual scram actuation, but CRA insertion is still gravity-driven. Many BDBEs also rely on the gravity scram function, although several assume its failure to demonstrate the ability of the scram motor drive-in function (DL4-RC3). DBAs rely entirely on the gravity scram function, as the motor drive-in function is NSRST.

The LBEs span an appropriate range of initiating conditions and event sequences to demonstrate the ability of the gravity scram function to protect the reactor. This includes loss of flow transients, loss of heat sink transients, reactivity transients (including spurious control rod withdrawal and seismic core compaction), and intermediate-to-primary leak scenarios. Transients that rely on the gravity scram do not exceed the 10 CFR 50.34 dose criteria or the F-C target curve. Therefore, the staff determined that the preliminary CRD design is consistent with DL3-RC1 and DL3-RC2 because LBEs relying on the gravity scram functions maintain dose consequences below acceptable limits.

PSAR section 7.2.5.2 describes how the CRD supports DL4-RC3 by providing motor-driven CRA insertion on a driveline scram follow demand insertion, as well as providing indication of full CRA insertion. Performance evaluation of DL4-RC3 is provided through the LBEs described in PSAR chapter 3. Due to the reliability of the gravity scram function, no AOOs or DBEs rely on the motor-driven scram function, and because it is NSRST, it is not credited for the mitigation of any DBAs.

However, several BDBEs do rely on the motor-driven scram function (e.g., PSAR section 3.8.1.2). Only one of the events, the loss of offsite power with scram motor drive-in, results in the fuel performance acceptance criteria being exceeded. The source term analysis indicates that two-thirds of the fuel fails due to creep rupture. As a loss of offsite power would not typically be expected to result in a severe plant transient, the staff audited this analysis and found it to be based on conservative assumptions that bound the expected event sequence. Additionally, consistent with acceptable LBE analyses under the NEI 18-04 process, the dose consequences remain well below the 10 CFR 50.34 dose criteria and the F-C target curve. Based on these results, the staff determined that the preliminary CRD design is consistent with DL4-RC3.

7.2.5.1.11.2 Radionuclide Retention Functions

PSAR section 7.2.5.2 states that the CRD supports DL3-RR1, “Primary Coolant Boundary,” which is SR. DL3-RR1 is the SR function to provide radionuclide retention as part of the primary coolant boundary. As discussed in SE section 7.2.5.1, the CRDM housing forms a portion of the primary coolant boundary. This function is primarily achieved through the design and materials selection of the SR CRDM housing, which, as discussed in PSAR section 7.2.5.3, is performed according to ASME BPVC III-5.

PSAR table 7.2.5-4 states that the CRDM housing is composed of type 316 stainless steel. PSAR table 7.2.5-3 indicates that the design temperature ranges from 150°F at the top end of the CRDM housing, located inside the HAA, to a design temperature still under development at the interface where the CRDM housing connects to the RPA (the staff expects this temperature to be closer to the RPA design temperature of 800°F). The design pressure varies from 30 pounds per square inch gauge (psig) to 60 psig, depending on proximity to the reactor.

Based on the use of an appropriate design code consistent with the expected temperature environment, and the selection of materials compatible with both the code and the sodium environment as discussed in SE section 7.2.5.1.6, the staff determined that the preliminary CRD design is consistent with DL3-RR1.

7.2.5.1.12 Programmatic Special Treatments

Programmatic special treatments applied to the CRD are summarized in PSAR section 7.2.5.4 and include the QAP, D-RAP, EQ program, PITAP, CVAP, and RIM program. Aside from the CVAP program, the applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the CRD contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to CRD. RG 1.20 states that the CVAP program should be applied to “unique or first-of-its-kind designs.” PSAR section 7.2.5.2 states that the CVAP is applied to the CRD lower guide tube in accordance with RG 1.20. The staff therefore determined these programs are appropriate to apply as special treatments to the CRD. Plant programs are evaluated further in SE chapter 8.

7.2.5.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.2.5 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.3 Fuel Handling Systems

PSAR section 7.3 describes the SSCs that make up the fuel handling systems. The following systems encompass all fuel handling system operations:

- Water pool fueling handling system (FHP)
- Ex-vessel fuel handling system (FHE)
- In-vessel fuel handling system (FHI)

7.3.1 Water Pool Fuel Handling System

PSAR section 7.3.1 describes the FHP, which provides a water-based environment for the intermediate-term storage of spent fuel and is also used for the storage of non-fuel irradiated core assemblies prior to disposal. The FHP provides core assembly cooling, spent fuel reactivity control, fission product containment, and radiation shielding. The FHP is divided into four subsystems: the pool immersion cell (PIC), spent fuel pool (SFP), fuel pool cooling (FPC), and fuel pool purification (FPP). The PIC facilitates the transition of fuel from a sodium environment in the FHE to a water environment in the SFP by cleaning sodium off fuel assemblies. The SFP contains equipment for handling, storing, and transferring fuel assemblies and other irradiated components. The SFP provides physical protection for stored core assemblies and maintains a large inventory of water under all conditions to support cooling, shielding, and protection of the stored core assemblies. The SFP also includes core assembly storage racks that support natural circulation cooling of core assemblies, maintain the core assemblies in a substantially subcritical arrangement, and contribute to physical protection of the core assemblies. The pool handling machine (PHM) supports the movement of core assemblies within the SFP. The FPC and FPP maintain the SFP water temperature and chemistry, respectively, within prescribed limits during normal operation.

The bottom loading transfer cask (BLTC), which is part of the FHE, serves as the interface between the FHE and FHP by delivering spent fuel assemblies to the PIC.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the FHP:

- 10 CFR 50.68, "Criticality accident requirements;"
- 10 CFR 50.155(e), "Spent fuel pool monitoring;"
- RG 1.13, "Spent Fuel Storage Facility Design Basis," Rev. 2 (ML070310035);
- RG 1.140, Rev. 3;
- RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," Rev. 3 (ML20141L613);
- RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Rev. 6 (ML21232A142);
- RG 1.240, "Fresh and Spent Fuel Pool Criticality Analysis," Rev. 0 (ML20356A127);
- RG 1.243, Rev. 0;
- RG 1.244, "Control of Heavy Loads at Nuclear Facilities," Rev. 0 (ML21006A346); and
- RG 1.97, Rev. 5.

PSAR sections 7.3.1 and 5.3 identify the following PDC as applicable to the FHP: 1, 2, 3, 4, 61, 62, 63, and 74.

7.3.1.1 Technical Evaluation

7.3.1.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.3.1.3 identifies the codes and standards selected for the design and construction of FHP components. The staff reviewed these standards for applicability to the specified uses. The selection of American Concrete Institute (ACI) 349-13 for SR concrete structures is consistent with the guidance in RG 1.142, which endorses this standard. Similarly, the application of the AISC N690-18 for design and construction of the SR steel SFP liner and PIC structural supports is consistent with RG 1.243, which endorses ANSI/AISC N690-18.

PSAR section 7.3.1.3 states that the SR FFC pressure vessel would be designed to ASME BPVC III-1, subsection NB; the SR spent fuel storage racks would be designed to ASME BPVC III-1, subsection NF; and the NSRST FHP emergency makeup piping would be designed to ASME B31.1-2022. Although PSAR section 8.1.1 states that KU1 uses NEI 18-04 as the basis for plant design SSC classifications rather than RG 1.26, the staff determined that the design standards for these components are acceptable because they are identified in RG 1.26 for components with comparable safety significance. The selection of ASME material handling standards (i.e., ASME NOG-1-2020, ASME NUM-1-2016, and ASME BTH-1-2017) for components performing a temporary support function is consistent with RG 1.244.

Instrumentation provided to monitor fuel storage conditions in the SFP is designed to IEEE 497-2016, consistent with RG 1.97. Therefore, the staff determined that these codes and standards are appropriate for the FHP based on its preliminary design and safety functions.

PSAR section 7.3.1.3 indicates that the PIC receiver, cleaning vessel, and portions of the processing skid are designed to the applicable requirements of ASME BPVC VIII-1. The PSAR states that the piping connecting the PIC cleaning vessel to the PIC processing skid will be designed to ASME B31.3-2020. The staff notes that these components are preliminarily designated as SR and would therefore generally be expected to be designed to ASME BPVC III, consistent with RG 1.87. As noted in SE section 7.3.1.1.9.3, the PRA modelling of the fuel assembly wash process in the PIC and attached BLTC does not fully align with the design described in the PSAR. A supplement dated September 17, 2025 (ML25260A002), provides additional information to clarify the evaluation of LBEs involving an excessive sodium-water reaction (ESWR) within the PIC. In this PIC supplement letter, the applicant states that a process hazard analysis will be conducted to inform updated PRA event trees and source terms. The resulting frequency and consequence information will be used to revise special treatments, including the codes and standards, which will be provided with the OL application. The staff determined that the information provided in the PSAR and PIC supplement for the PIC and associated processing skid is sufficient for the CP stage as discussed in SE section 7.3.1.1.9.3; as such, their detailed design may reasonably be left for consideration at the OL stage. Accordingly, evaluation of the standards applied to the PIC components will be performed at the OL stage.

ASME AG-1-2009 (including 2010 Addenda 1a and 2011 Addenda 1b), is endorsed by the NRC in RG 1.140 for the design, inspection, and testing of normal atmosphere cleanup systems for controlling releases of airborne radioactive materials to the environment in light water reactors. The staff reviewed ASME AG-1-2019 and determined it is acceptable for the FHP.

The PSAR identifies additional design standards relevant to the safe storage of irradiated fuel in pools related to the guidance in RG 1.13. Specifically, the PSAR identifies partial conformance with RG 1.13, with exceptions related to the heat removal capability of the FPC system and the seismic qualification of the normal SFP makeup water system. The applicant justified these limited exceptions on the basis that the SFP can be passively cooled by its contained water inventory for over seven days due to the low design-basis heat load. The staff determined that the level of FHP conformance with RG 1.13 to be appropriate based on its preliminary design basis with extensive fuel storage in both the IVS and EVST occurring prior to placement in the FHP.

Based on its review described above, the staff determined that the preliminary FHP design is consistent with PDC 1 because the codes and standards specified for the design and construction of the FHP SSCs are acceptable.

7.3.1.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

PSAR section 7.3.1.2 states that the FHP SSCs are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain the capability to perform safety-significant functions. The SR FHB substructure, described in PSAR section 7.8.2, which includes the SFP structure, provides protection against external missile and flooding hazards. The FHB substructure, the design of the NSRST FHB superstructure main structural members to absorb the energy of wind-driven missiles, the water inventory of the SFP, and the core assembly racks collectively provide acceptable protection to the core assemblies and other SR SSCs from high energy wind-driven missiles.

PSAR table 7.3.1-1 lists the SFP liner, the FFC, and both the intact and failed fuel racks with SR seismic classifications of SCS1. Post-accident SFP monitoring instrumentation and pool handling equipment are classified as NSRST and SCN3. PIC components that contain postulated radionuclide releases are listed as SR with a seismic classification of SCS2, consistent with the preliminary SR function for the BLTC transfer barrier. This is a reasonable preliminary seismic classification; however, as discussed below in SE section 7.3.1.1.9.3, the staff determined that detailed design of the PIC and associated processing skid may reasonably be left for consideration at the OL stage.

The staff determined that the preliminary FHP design is consistent with PDC 2 because taken together, the FHB and FHP SSCs ensure that safety-significant FHP SSCs retain the capability to perform their safety functions considering the effects of natural phenomena. The FHB and FHP SSCs provide barriers against the effects of most natural phenomena, and safety-significant FHP SSCs are designed to withstand the effects of postulated seismic events.

7.3.1.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.3.1.2 states that the FHP core assembly handling equipment and cleaning process are designed to contain assembly residual sodium to limit the potential for fires. The FHP is located within the FHB in dedicated spaces for core assembly handling and storage activities to minimize the potential impact on other safety-

significant SSCs in the event of a fire. FHP equipment is constructed of noncombustible and non-flammable materials to the extent practical.

The staff determined that the preliminary FHP design is consistent with PDC 3 because the design and location of FHP SSCs appropriately minimize the probability and effects of fires and explosions. The FHP provides for the controlled removal of small amounts of residual sodium during the transfer of core assemblies from the FHE within the PIC enclosure.

7.3.1.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions.

PSAR section 7.3.1.2 states that FHP SSCs are designed to accommodate the effects of, and to be compatible with, environmental conditions associated with spent fuel decay heat and radiation, as well as exposure to liquid sodium and its aerosols, vapors, and oxidation products. Materials used in FHP SSCs are selected to withstand the operating environments to which they are exposed. As noted in PSAR section 7.3.1.3.1, the FHB substructure described in PSAR section 7.8.2, which includes the SFP structure and its contained water inventory, provides protection against external hazards, including missiles. Additionally, PSAR section 1.1.4.3.3 states that a single permanent concrete barrier separates the cask loading pit from the fuel storage area, which helps protect against a complete loss of coolant from the storage area in the event of a cask handling accident over the cask pit.

USO preliminarily identified the prevention of heavy load drops from the FHB crane over the SFP area as a necessary measure for DID adequacy. As discussed in SE section 5.1, USO provided a supplement to its PSAR dated September 10, 2025 (ML25253A386), that defined a process to identify a minimum set of design features or functions that prevent postulated initiating events from occurring and to classify those features or functions as NSRST. The staff will review the results of this process related to prevention of heavy load drops in the FHB at the OL stage.

The staff determined that the preliminary design of the FHP system is consistent with PDC 4 because the safety-significant portions of the FHP system are appropriately protected from external hazards and are designed to accommodate the effects of environmental conditions associated with both normal operations and postulated accidents, consistent with the guidance in RG 1.13.

7.3.1.1.5 Fuel Storage and Handling and Radioactivity Control (PDC 61)

PDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of safety-significant components, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the safety significance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.

PSAR section 7.3.1.3 describes the SFP as a stainless-steel-lined concrete structure with a leak chase to identify and monitor liner leakage; the PIC is in a corner of the SFP. Free-standing storage racks provide storage for both intact fuel and fuel contained within failed fuel canisters (FFCs). PSAR section 7.3.1.2 states that the FHP is designed to permit periodic inspection and testing to assure safety-significant functional capability. The staff determined that the design and arrangement of the SFP, the PIC, and the storage racks support leakage monitoring and periodic inspection of safety-significant functions consistent with PDC 61.

PSAR section 7.3.1.2 states that the FHP provides adequate SFP water inventory for residual heat removal from stored fuel and provides suitable shielding for radiation protection under both normal and postulated accident conditions. Intact fuel and FFCs within the SFP are passively cooled by natural circulation of the SFP water. Within the PIC, fuel cleaning vessels and FFCs are designed to passively transfer heat from the contained core assemblies to the SFP water. Under normal conditions, the water is cooled by the FPC system. Under accident conditions, cooling is provided by evaporation of the water inventory.

PSAR section 7.3.1.3 describes that vertical motion limits on the PHM ensure sufficient water for shielding above the core assemblies during movement. The PSAR also states that biological shields, physical access controls, or administrative boundaries are used to protect workers where water shielding is not practical. Drawings reviewed during the audit confirmed that the SFP can maintain adequate water inventory above the fuel for shielding, including during fuel movement. Therefore, the staff determined the FHP preliminary design provides reliable residual heat removal and shielding for fuel assemblies within the PIC, during fuel movement, within an FFC, and when located within the storage racks, consistent with PDC 61.

PSAR section 7.3.1.3 describes FHP design features that prevent a significant reduction in fuel storage cooling under accident conditions. The pool water inventory is sufficient to cool the fuel in a fully-loaded SFP by evaporation for at least seven days without makeup water. As previously noted, a stainless-steel liner prevents leakage of pool water. Makeup water is normally available from the NI water system; under accident conditions it is available from offsite sources via an external fill pipe. The pool wall has no penetrations below the top of stored fuel. PSAR section 1.1.4.3.3 states that there are no hydraulic penetrations or pipes extending below the minimum required water level for fuel rack shielding and that a single permanent concrete barrier separates the fuel storage area from the cask loading pit.

PSAR section 7.3.1.2 describes that the FPC and FPP suction and discharge lines are located such that a leak or rupture would not drain the SFP below a minimum safe level. PSAR section 7.3.1.2 further states that the FHP conforms with RG 1.13 with two exceptions: the normal makeup system is not seismically qualified and the design heat load is lower than specified. The staff determined that these exceptions are justified based on the preliminary FHP design, which assumes pool storage only after significant decay, and the analysis indicating that the SFP can be passively cooled for more than seven days before operator action is required. This seven-day period is based in part on preliminary operational considerations regarding the decay of fuel within the FHE prior to transfer to the FHP, which will be assessed as part of the OL application review. Therefore, the staff determined that the FHP preliminary design provides appropriate design features to prevent a significant reduction in fuel cooling under accident conditions, consistent with PDC 61.

PSAR section 7.3.1.3 describes that, in conjunction with the BLTC, the PIC provides a containment barrier function and a controlled filtration path for processing releases from the PIC cleaning vessel during normal operations and for radionuclides in the event of an accidental

drop of a core assembly. Confinement for other postulated radionuclide releases in the FHB, such as a core assembly drop within the SFP, is provided through the facility's functional containment design, which is supported by the FHB and NHV, and is addressed in SE section 7.5.1. Therefore, the preliminary design of the FHP SSCs provides appropriate containment, confinement, and filtering capabilities, consistent with PDC 61.

7.3.1.1.6 Prevention of Criticality in Fuel Storage and Handling (PDC 62)

PDC 62 requires that criticality in the fuel storage and handling system be prevented through physical systems or processes, preferably by using geometrically safe configurations. PSAR section 7.3.1.2 states that the storage racks maintain spent fuel subcritical by controlling the relative spacing within the racks (geometry control) and by installing fixed borated neutron absorbers within the assembly storage cells. The storage racks do not rely on specific zoning of fuel assemblies to maintain subcriticality.

The staff determined that the preliminary FHP design is consistent with PDC 62 and provides an acceptable means of criticality prevention based on the evaluation of the supporting criticality analysis in SE section 3.13.

7.3.1.1.7 Monitoring Fuel and Waste Storage (PDC 63)

PDC 63 requires that appropriate systems be provided in fuel storage and radioactive waste systems, and associated handling areas, to (1) detect conditions that may result in loss of residual heat removal capability and excessive radiation levels, and (2) initiate appropriate safety actions.

PSAR section 7.3.1.5 states that SFP water level sensors initiate SFP refill upon detection of low water level during normal operation and also serve to monitor SFP water level under adverse conditions to maintain passive cooling and shielding of the spent fuel. SFP temperature instruments also allow for PAM. PSAR section 7.3.1.2 states that FHB radiation monitoring provides indication of high radiation levels to support operator initiation of appropriate safety actions.

The staff determined the preliminary FHP design is consistent with PDC 63 because the monitoring capabilities of the SFP and FHB allow for detection of conditions which could result in a loss of heat removal and support initiation of appropriate safety actions. The RMS is evaluated in SE section 7.6.6.

7.3.1.1.8 Sodium-Water Reaction Prevention and Mitigation (PDC 74)

PDC 74 requires that SSCs containing sodium be designed and located to avoid contact between sodium and water. PSAR section 7.5.1.3 states that core assemblies are loaded into a cleaning vessel within the PIC, which provides a controlled environment to chemically clean and remove residual sodium coolant using a humidified inert gas stream. Effluents from the cleaning process are collected and separated into gaseous and liquid streams for further processing. The PIC cleaning vessel isolates the core assembly from the SFP water until the assembly is ready to be flooded. The cleaning vessel is locked to the fixed PIC receiver until an interlock confirms that residual sodium has been removed. The cleaning vessel, containing the core assembly, then lowers into the SFP.

Core assemblies with damaged fuel cladding are placed directly into an FFC, which is preloaded into the PIC, rather than into a cleaning vessel. Once the FFC loading and sealing processes are complete, the core assembly is isolated from the external environment, thereby preventing any sodium-water interaction.

Because the PIC allows for the isolation of sodium from water for both normal and damaged core assemblies, either through controlled sodium removal in a cleaning vessel or direct placement into a sealed FFC, the staff determined that the preliminary FHP design effectively prevents sodium-water contact during transfer into the SFP, consistent with PDC 74.

7.3.1.1.9 Safety Functions

7.3.1.1.9.1 Reactivity Control Functions

PSAR section 7.3.1.1 states that the SFP includes storage racks that maintain stored spent fuel subcritical by controlling the relative spacing of fuel within the racks (geometry control) and by including fixed neutron absorbers in the racks. These reactivity control functions were not modeled in the PRA; instead, PSAR section 3.13 describes analyses demonstrating the prevention of criticality in fuel storage and handling. PSAR section 7.3.1.2 states that the intact and failed fuel racks provide a SR structural support function for core assemblies. Additionally, PSAR section 7.3.1.3 states that a neutron absorber material monitoring program will be provided at the OL stage.

As discussed in SE section 7.3.1.1.2, the intact and failed fuel storage racks, along with the supporting SFP structure, are designed to quality standards appropriate for SR structural support functions and are subject to programmatic special treatments. Therefore, the staff determined that the preliminary FHP design, and specifically, the intact and failed fuel racks, is consistent with reactivity control functions. Availability of the neutron absorber material monitoring program at the OL stage is consistent with the requirements of 10 CFR 50.34 and 10 CFR 50.35 because it is an operational program.

7.3.1.1.9.2 Heat Removal Functions

PSAR section 7.1.3.2 identifies FHP heat removal functions. The FHP provides SR heat removal functions DL3-HR10, "Passive Heat Removal in the FHP," which includes the natural circulation of water through the fuel storage racks and around the PIC cleaning vessel, and DL3-HR13, "Passive Heat Removal in the Failed Fuel Canister." The FHP supports these SR functions by providing a natural circulation flow path through the storage racks and around the PIC to dissipate decay heat, and by facilitating heat conduction from the fuel assembly within the FFC to the surrounding SFP water.

The FHP also provides NSRST function DL4-HR4, "FHP Emergency Makeup," which consists of piping that provides an unobstructed flow path to the SFP from an external water fill connection located outside the FHB.

The staff determined that the preliminary FHP design is consistent with these functions because it includes provisions to ensure passive cooling of all potential fuel locations within a large pool and reliable transfer of the collected heat to the environment.

7.3.1.1.9.3 Radionuclide Retention Functions

PSAR section 7.1.3.2 identifies the following radionuclide retention functions associated with the FHP:

- DL3-RR8 – FFC Barrier (SR)
- DL3-RR5a – BLTC Transfer Barrier (SR)
- DL2-RR2 – PIC Radionuclide Boundary (NSRST), which consists of the PIC receiver, cleaning vessel, and process skid.

The FFC barrier (DL3-RR8) forms a radionuclide confinement boundary around an enclosed fuel assembly once sealed and also prevents sodium-water interactions. The BLTC transfer barrier (DL3-RR5a) is established when the BLTC is mated with the PIC. This barrier includes the BLTC, the PIC receiver, the cleaning vessel, the process skid, and the fueling floor valve (FFV) attached to the PIC. It provides a confinement boundary around a fuel assembly dropped into the PIC or a core assembly experiencing ESWR in the PIC with the BLTC attached to the open FFV.

PSAR section 3.7.4.6 provides the results of a preliminary DBE evaluation that credits the DL3-RR5a boundary for mitigation. PSAR section 3.9.5.3 describes the associated DBA evaluation. These LBEs considered source terms selected from a limited set of evaluated configurations not specific to LBEs involving the PIC. PSAR section 3.2 describes the methodology used to develop the preliminary source terms, and the associated staff evaluation of these source terms is provided in SE section 3.2.1.3.

DL2-RR2 refers to the barrier established when the PIC is isolated from the BLTC. This function was not associated with any existing LBE because the PRA model described in the PSAR did not consider an ESWR as an initiating event when the BLTC is not mated to the PIC.

A supplement dated September 17, 2025, provides clarification regarding the evaluation of LBEs involving an ESWR within the PIC. The supplement indicates that the PRA model used to inform the PSAR, specifically for the PIC and BLTC configuration during the fuel assembly wash process, differs from the preliminary design. Furthermore, the iterative design process resulted in changes that reduced the likelihood of ESWR in the PIC. The supplement states that, prior to submitting the KU1 OL application, a process hazard analysis will be performed to inform updated PRA event trees and source terms. The resulting frequency and consequence information will be used to define system LBEs, establish SSC classifications, and revise special treatments, consistent with NEI 18-04.

The staff reviewed the FHP functions credited for radionuclide retention. The FFC supports radionuclide retention by providing a sealed barrier around failed fuel that also prevents uncontrolled sodium-water reactions. The BLTC transfer barrier supports radionuclide retention by directing any releases through the PIC processing skid, where they are filtered before release to the NHV. Radionuclide retention provided by the filter in the BLTC transfer barrier, along with the SFP water inventory for fuel handling accidents within the SFP, is considered when determining the source term.

The PIC supplement states that the radionuclide retention barriers associated with the PIC will be reevaluated as part of the development of the OL application. Although some postulated events within the BLTC transfer barrier result in significant consequences, the staff expects a reassessment of the process hazards associated with fuel cleaning in the PIC to be presented at the OL stage, along with design changes intended to reduce the risk of PIC operations. The staff determined that the information provided for the PIC in the PSAR and the PIC supplement is sufficient at the CP stage because the PIC radionuclide retention barrier SSCs are already considered safety-significant (SR through DL3-RR5a and NSRST through DL2-RR2) and because the additional preventative controls will reduce the LBE frequencies and provide additional margin to the F-C target curve. Further detailed design information for the PIC and associated processing skid may reasonably be left to the OL stage because their functional design requirements are established by the preliminary design and because the PIC is separable from (i.e., would not be expected to affect) the design of the rest of the FHP. Therefore, the staff determined that the preliminary FHP design is consistent with the identified radionuclide retention functions.

7.3.1.1.9.4 Additional Functions

PSAR section 7.3.1.2 identifies the provision of structural support to SR SSCs, action as a temporary load path, external hazard protection, and PAM (DL5-PAM1) as additional safety significant functions of FHP SSCs.

The structural support function primarily applies to the spent fuel storage racks, which the staff previously addressed as a reactivity control function. The NSRST temporary load path function applies to the PHM and the PIC, each of which supports a single core assembly for a duration. The temporary support function serves to prevent mechanical damage to the fuel cladding radionuclide retention barrier during fuel movement. The SFP water inventory and the storage racks provide protection from wind-driven missiles for SR fuel stored in the storage racks. The PAM function identifies SFP water level and temperature as providing instrument signals for monitoring of SFP conditions. PSAR section 7.3.1.2 states partial conformance to RG 1.97, which endorses IEEE 497-2016 for PAM. The staff will review selected PAM variables and the extent of conformance with RG 1.97 at the OL stage.

The staff determined that the preliminary FHP design is consistent with these functions because the FHP includes design features that prevent challenges to FSFs and provide monitoring capability that supports early detection of developing challenges.

7.3.1.1.10 Programmatic Special Treatments

Programmatic special treatments preliminarily identified for application to the FHP are summarized in PSAR section 7.3.1.4 and include the QAP, D-RAP, RIM program, EQ program, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the FHP contains SR and NSRST SSCs, the staff determined that these programs are appropriate to apply to the FHP. Plant programs are evaluated further in SE chapter 8.

7.3.1.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.3.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.3.2 Ex-Vessel Fuel Handling System

PSAR section 7.3.2 describes the FHE as the equipment that facilitates the receipt, inspection, conditioning, and storage of core assemblies, as well as their removal from the core. The FHE interfaces with the FHI, as described in PSAR section 7.3.3, for in-vessel handling operations, and with the FHP, as described in PSAR section 7.3.1, for fuel pool handling operations.

The FHE is designed to convey and store a core assembly in an inert environment from the time it is placed in the new assembly preconditioning station (NAPS) until it is transferred to the RV, and subsequently from the time a core assembly is removed from the RV until it is placed in the PIC and lowered into the SFP. The FHE provides radiation shielding, radionuclide retention, and decay heat removal for irradiated fuel assemblies.

Major components of the FHE include the ex-vessel handling machine (EVHM), the EVST, and the BLTC. The EVHM and BLTC are rail-mounted, shielded, sealable, vessel transport systems that support core assembly handling operations. The EVST is a liquid-sodium-filled tank with a rotatable carousel capable of temporarily storing up to approximately 100 core assemblies. Additional components of the FHE include the NAPS, the pin removal cell (PRC), the core assembly transfer tube (CATT), the fuel transfer adapters (FTAs), and FFVs.

The EVHM is used to transfer core assemblies, whether non-fuel, irradiated fuel, or non-irradiated fuel, between the RV in the RXB and the EVST in the FHB. Transferred assemblies are contained within a core assembly pot (CAP) filled with liquid sodium, which is transported inside a shielded cask assembly. The EVHM uses a grapple and hoist assembly to move the CAP into and out of the EVHM.

The EVST provides a controlled environment for the temporary storage of new components in route to the reactor and irradiated core assemblies removed from the reactor. The EVST supports active and passive decay heat removal, maintains a containment barrier, and provides radiation shielding. The EVST GV provides containment if the primary barrier leaks.

The BLTC transfers core assemblies in a shielded cask assembly between the NAPS, EVST, and PIC within the FHB. The BLTC also uses a grapple and hoist assembly for loading and unloading core assemblies. The BLTC supports active and passive decay heat removal, maintains a containment barrier, and provides temporary structural support for core assemblies.

The PRC is a below grade hot cell located along the EVHM travel path. It supports the extraction and packaging of removable pins taken from LTAs and LDAs after irradiation in the RV. The PRC maintains an inert atmosphere and provides cooling to ensure acceptable temperatures. The EVHM transfers LTAs and LDAs, each within a CAP, from the RV to the PRC and from the PRC to the EVST. The PRC supports passive decay heat removal, maintains a containment barrier, and provides temporary structural support for an LTA or LDA. The pin extraction machine within the PRC temporarily supports removed fuel pins.

The NAPS provides an inert environment for preconditioning new core assemblies. The geometric arrangement of core assemblies within the NAPS ensures a subcritical configuration. The NAPS holds only unirradiated core assemblies.

The CATT connects the fuel transfer lift (FTL), installed in the RV during refueling, to the FFV installed on the operating floor above the RV. It maintains an inert environment and provides cooling and shielding to a core assembly during transit in the HAA above the RV. The CATT

contributes to maintaining containment. The FTL is part of the FHI and serves as the interface between the FHI and FHE.

An FTA connects each EVST fuel transfer port to the FFV installed on the operating floor above the EVST. It maintains an inert environment during core assembly transfers when the transfer port plug is removed from the EVST. The FTA contributes to maintaining EVST confinement.

The FFV maintains an inert environment around core assemblies and serves as an adapter for connecting the EVHM or BLTC to the CATT, FTA, PRC, PIC, or NAPS. The FFV also contributes to maintaining containment.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the FHE:

- RG 1.87, Rev. 2;
- RG 1.97, Rev. 5;
- RG 1.243, Rev. 0;
- RG 1.244, Rev. 0.

PSAR sections 7.3.2 and 5.3 identify the following PDC as applicable to the FHE: 1, 2, 3, 4, 16, 61, 62, 63, 72, 73, and 74.

7.3.2.1 Technical Evaluation

7.3.2.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.3.2.3 identifies the codes and standards applied to the design and construction of safety-significant FHE components. The staff reviewed the selected standards for applicability to their specified uses.

Consistent with the guidance of RG 1.87, PSAR section 7.3.2.3 states that the applicable requirements of ASME BPVC III-5 are applied to the design of the following components:

- EVHM and BLTC shielded cask assemblies;
- EVST main pressure vessel, GV, rotating carousel, support structure, and associated piping up to the attached systems isolation to the vessel;
- PRC liquid sodium holding tank; and
- CATT, the FTA, and the FFV.

The EVST striker plate, which protects the EVST's functional and confinement capabilities against external loads, is designed to ANSI/AISC N690-18, which is endorsed by RG 1.243 for

the construction of safety-significant steel structures. The selection of ASME NOG-1-2020 for the design and construction of the NSRST EVHM and BLTC transporters is appropriate based on their rail mounted construction and is consistent with the guidance in RG 1.244. Similarly, the selection of ASME NOG-1-2020 and ASME BTH-1-2017 for the design and construction of the EVHM and BLTC hoists and grapples, respectively, is consistent with the guidance in RG 1.244. Instrumentation used to monitor fuel storage conditions in the EVST is designed to IEEE 497-2016, consistent with the guidance in RG 1.97.

Based on its review described above, the staff determined that the preliminary FHE design is consistent with PDC 1 because the codes and standards specified for the design and construction of the FHE SSCs are acceptable.

7.3.2.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

PSAR section 7.3.2.2 states that FHE SSCs are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain the capability to perform safety-significant functions. The SR RXB and FHB substructures, described in PSAR sections 7.8.1.1 and 7.8.2, respectively, provide protection against external missile and flooding hazards for the CATT, the EVST, and the PRC. The capability of FHE components to perform safety-significant functions will be established using the methods described in PSAR section 6.1.1 to evaluate natural phenomena and determine associated DBHLs.

PSAR table 7.3.2-1 states that the EVHM shielded cask assembly, BLTC shielded cask assembly, EVST pressure vessel and supports, FFV, CATT, PRC and CAP are classified as SCS1. This classification supports their SR functions, such as radionuclide retention and reactivity control. Temporary support functions associated with the EVHM and BLTC transporters, including maintaining the shielded cask assembly in a vertical orientation and reducing the frequency of core assembly handling events, are classified as SCN1 to ensure functional capability during and after a seismic event. The NSRST EVST GV, which ensures cooling in the event of an EVST main vessel leak, and the NSRST pin extraction machine, which provides temporary structural support for an extracted fuel pin, are classified as SCN3. The staff determined that this classification is appropriate given the very low frequency of these conditions.

The staff determined that the preliminary FHE design is consistent with PDC 2 because, taken together, the FHB, RXB, and FHE SSCs ensure that safety-significant FHE SSCs retain the capability to perform their safety functions considering the effects of natural phenomena, the likelihood of the safety function being needed in combination with the effects of natural phenomena, and the safety significance of the function.

7.3.2.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.3.2.2 states that FHE core assembly handling equipment has design features that minimize the potential for liquid sodium leakage to its surroundings, thereby limiting the potential for fires. FHE equipment is constructed of noncombustible and non-flammable materials to the extent practical.

The staff determined the preliminary FHE design is consistent with PDC 3 because the design appropriately minimizes the probability and effects of fires and explosions; specifically, FHE SSCs minimize the potential for sodium leakage during the transfer of core assemblies between major components of the FHE.

7.3.2.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions.

PSAR section 7.3.2.2 states that FHE SSCs are designed to accommodate the effects of, and be compatible with, environmental conditions associated with spent fuel decay heat and radiation, as well as exposure to liquid sodium and its aerosols, vapors, and oxidation products. The EVST striker plate protects the EVST containment function and stored fuel assemblies from the effects of postulated dropped loads. Materials used in FHE SSCs are selected to withstand the operating environments to which they are exposed. As noted in SE section 7.3.2.1.2, the FHB and the RXB substructures described in PSAR section 7.8 provide protection against external hazards, including missiles.

The staff determined that the preliminary design of the FHE is consistent with PDC 4 because safety-significant portions of the FHE are appropriately protected from external hazards and are designed to accommodate the effects of environmental conditions associated with both normal operations and postulated accidents.

7.3.2.1.5 Containment Design (PDC 16)

PDC 16 requires that a reactor functional containment be provided to control the release of radioactivity to the environment and to ensure that the functional containment safety-significant design conditions are not exceeded for the duration of accident conditions.

Radionuclide retention is implemented using a functional containment strategy, which consists of multiple barriers to control the release of radioactivity to the environment. The EVHM, FFV (located at the top of the CATT), and gas retaining parts of the CATT function as a passive part of this strategy by maintaining the integrity of the functional containment HAA barrier during fuel transfer operations. These components are designed such that their safety-significant design conditions are not exceeded for the duration of postulated accident conditions.

The staff reviewed evaluations of LBEs to determine the FHE components credited for achieving the evaluated dose consequences for postulated releases of radioactivity. The staff determined that the preliminary FHE design is consistent with PDC 16 because the evaluations indicate that FHE components contribute to isolation functions that control the release of radionuclides to the environment.

7.3.2.1.6 Fuel Storage and Handling and Radioactivity Control (PDC 61)

PDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of safety-significant components, (2) with suitable shielding for

radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the safety significance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.

Periodic Inspection and Testing

PSAR section 7.3.2.2 states that FHE components are designed to permit periodic inspection and testing to ensure safety-significant functional capability. The staff determined that the preliminary design of major FHE components support monitoring for leakage from the EVST and allows for periodic inspection of the safety-significant functions of other FHE components, consistent with PDC 61.

Heat Removal

PSAR section 7.3.2.2 states that FHE components provide adequate residual heat removal under normal and postulated accident conditions. Heat removal from the EVHM and BLTC shielded cask assemblies is provided by air flow through an annular volume around the confinement cask. Radiative and natural convection (i.e., passive) heat transfer provides the SR heat removal function. Heat removal can be increased by using fans to provide active cooling during the transfer of irradiated fuel or decreased by closing dampers to reduce airflow during movements of fresh fuel or other core assemblies. This approach maintains the sodium in a liquid state at appropriate temperatures.

The EVHM and BLTC passive cooling capabilities provide cooling under accident conditions, including loss of active cooling. As described in PSAR sections 7.3.2.1.1 and 7.3.2.1.2, the EVHM passive cooling system is designed to handle core assemblies seven days after shutdown with a decay heat of 15.2 kW, while the BLTC passive cooling system is designed to remove 1.2 kW of decay heat, which the staff determined represents approximately one year of decay (considering both in-vessel and EVST storage).

The design basis of the EVHM's passive heat removal capability, which accommodates fuel after seven days of decay, provides assurance that assemblies unloaded from the core will remain within passive cooling limits. This is supported by the time required to establish refueling conditions and install refueling equipment, which consumes a significant portion of the seven-day period. Administrative controls and fuel handling control systems are also intended to ensure fuel has adequately decayed before being loaded into the EVHM. Although the BLTC passive heat removal capability does not bound all potential fuel movements, it offers greater certainty that administrative controls will maintain the decay heat load of core assemblies within the limit. This is because the decay time of assemblies in the EVST can be effectively tracked, and core assemblies moved into the BLTC can be more easily identified.

The staff performed simplified calculations of radiative heat transfer using preliminary design information and confirmed that the passive heat removal rates are reasonable for the specified maximum pin cladding temperature of 1157°F (625°C) and the preliminary design configurations of the EVHM and BLTC. Therefore, the staff found that the EVHM and BLTC cooling systems have reliability that reflects the safety significance of residual and decay heat removal, consistent with PDC 61.

As described in PSAR section 7.3.2.1.3, the EVST can actively and passively remove decay heat from core assemblies stored within CAPs in EVST storage locations. PSAR section 7.3.2.3

states that the EVST passive heat removal is accomplished using inlet and outlet stacks and ducts that circulate air into and out of the EVST vault and is designed to remove the maximum decay heat load of 216 kW while maintaining the fuel assembly temperatures below the maximum pin cladding temperature of 1157°F (625°C). PSAR section 7.3.2.3 also states that dampers for the EVST passive cooling system can be adjusted to maintain temperature within operational limits during low heat load conditions and are designed to fail open. PSAR section 3.7.4.7 states that no fuel damage occurs for at least 72 hours following a loss of both active and passive cooling, although longer-term heat removal conditions require further assessment.

The heat load within the EVST is dominated by recent discharged fuel from the core (i.e., LDAs and LTAs) as well as by the operational management of fuel storage, specifically, the proportion of EVST storage locations occupied by spent fuel that has been transferred from the IVS after a decay period, relative to those holding fresh fuel. The staff determined that the EVST cooling systems and the EVST thermal heat capacity are capable of reliably protecting fuel cladding within the EVST, consistent with PDC 61. The staff will review operational management of fuel storage at the OL stage to confirm the thermal capacity of the EVST provides appropriate DID for events involving degraded passive EVST cooling.

PSAR section 7.3.2.1.7 describes a sodium tank within the PRC that receives an LTA or LDA from the EVHM. This tank provides cooling of the LTA or LDA by transferring heat through the liquid sodium to the tank wall, where free convection of the PRC atmosphere provides cooling. The PRC also contains an inert atmosphere that cools removed lead test pins to maintain their temperatures within limits. Therefore, the staff determined the PRC preliminary design has demonstrated residual heat removal with reliability reflecting the safety significance of this function for fuel assemblies within the PRC, consistent with PDC 61.

Preventing Reduction of Fuel Cooling

PSAR section 7.3.2.1 describes EVHM and BLTC design features that prevent a significant reduction in fuel storage cooling under accident conditions. These SSCs may see a reduction in heat removal due to a loss of power to the active cooling fans or inadvertent closure of the dampers. However, for both the EVHM and BLTC, passive cooling is sufficient for planned heat loads, and the dampers are designed to fail open to maximize passive heat removal under accident conditions. Therefore, the staff determined the EVHM and BLTC preliminary design provides appropriate design features to prevent a significant reduction in fuel cooling under accident conditions, consistent with PDC 61.

PSAR sections 7.3.2.1 and 7.3.2.3 describe EVST design features that ensure accident conditions do not result in a loss adequate cooling. The EVST design includes a GV that captures leaked liquid sodium while maintaining the sodium level above the fueled portion of the core assemblies. The attached active cooling systems are designed such that a failure (e.g., pipe rupture) would not prevent the EVST from performing its SR passive heat removal function. Dampers controlling passive cooling airflow fail open and penetrations in the EVST are above the minimum safe sodium level ensure that adequate cooling capability is maintained. In addition, PSAR section 7.2.4.2.1 states that the SPS-E prevents draining, sluicing, or siphoning the EVST below its safe level. Therefore, the staff determined the EVST preliminary design provides appropriate design features to prevent a significant reduction in fuel cooling under accident conditions, consistent with PDC 61.

The staff also evaluated fuel assembly cooling during transfer from the RV to the EVHM through the CATT due to its length. PSAR section 7.3.2.1.6 states that the CATT internal cavity temperature is maintained to prevent solidification of the liquid sodium coolant and overheating of the core assembly. In audit discussions, the staff assessed the potential for inadequate cooling in the event the hoist stalled during transfer with a fuel assembly within the CATT. In these discussions, USO provided information indicating that the frequency of such an event is sufficiently low to screen out of consideration as an LBE. A heat transfer analysis for the CATT is planned for the OL stage and is expected to show adequate time to implement operational actions to complete the lift or return the fuel assembly to the RV. The staff confirmed that the frequency of a hoist stall would be low enough to be negligible and that the potential operational recovery actions were reasonable. Therefore, the staff determined that the FHE preliminary design is adequate to prevent a significant reduction in fuel cooling under accident conditions in the CATT.

Containment and Shielding

The major FHE components described in PSAR sections 7.3.2.1, 7.3.2.2, and 7.3.2.3 provide containment and shielding during movement and storage of core assemblies. The EVHM and BLTC include a shielded cask assembly (containment vessel) that performs both shielding and containment functions. The EVST includes containment and shielding features in its design. The EVST main vessel and vessel head form a containment boundary around the liquid sodium coolant and the inert cover gas. Connections to the EVST cover gas space are equipped with isolation valves that close automatically on detection of a leak or loss of power. However, preliminary analysis does not rely on isolation valve closure to limit leakage during fuel handling events in the EVST because the connected systems are closed and a simultaneous auxiliary system breach during a fuel handling event is not considered credible. The connected systems consist of the EVST cover gas, SPS-E, and active cooling system.

PSAR section 7.3.2.3 describes the shielding provided by the EVST vessel head, additional shielding layers, and removable shield plugs in core assembly transfer ports. During core assembly transfers, the containment boundary of the EVHM and BLTC shielded cask assemblies is extended to the attached FFV and connected volumes, including the EVST, PRC, and CATT (along with connected FHI components). During transfers to the PIC, PSAR section 7.3.1.3 states that the BLTC and associated FFV, in conjunction with the PIC, provide a containment barrier function.

Conclusion

Based on the evaluation above, the staff determined that the preliminary design of the FHE is consistent with PDC 61 because FHE SSCs provide appropriate shielding, containment, confinement, and filtering capabilities.

7.3.2.1.7 Prevention of Criticality in Fuel Storage and Handling (PDC 62)

PDC 62 requires that criticality in the fuel storage and handling system be prevented through physical systems or processes, preferably by using geometrically safe configurations.

PSAR section 7.3.2.2 states that the EVST and NAPS maintain stored spent fuel subcritical by controlling the relative spacing within the racks (i.e., geometry control). In the EVST, geometric spacing of stored fuel is ensured by the SR structural support function of its components, including the rotating carousel that holds the fuel. As discussed in SE section 7.3.2.1.12.1, the

safety classification of NAPS design features for the prevention of criticality is subject to review under the process evaluated in SE section 5.1.

PSAR section 3.14.2.5 addresses new fuel receipt and describes it as follows: new core assemblies arrive in a shipping container designed to maintain a subcritical configuration. The shipping container is placed in a designated storage location, after which a core assembly is then placed in the assembly receipt inspection stand (ARIS) for inspection. Once inspection is complete, the new core assembly is placed in the NAPS. The PSAR states that criticality analysis for the shipping container, the shipping container storage location, and the ARIS will be available at the OL stage.

The staff determined that the preliminary FHE design is consistent with PDC 62 because it provides an acceptable means of criticality prevention based on the evaluation of the supporting criticality analyses for the EVST and NAPS, provided in SE section 3.13. The staff will evaluate the fuel receipt and inspection process at the OL stage.

7.3.2.1.8 Monitoring Fuel and Waste Storage (PDC 63)

PDC 63 requires that appropriate systems be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

PSAR section 7.3.2.3 states that the EVST design utilizes sensors to monitor internal bulk sodium temperatures to detect inadequate heat removal and identify when fuel cladding temperature limits are approached.

PSAR section 7.3.2 describes the ability to manage temperatures within the EVHM and BLTC using their associated cooling and heating systems. PSAR section 7.3.2.5 states that the fuel handling supervisory control system (FHC) provides process information and alarms for parameters such as EVHM and BLTC temperature, pressure, and argon flow. PSAR section 7.6.6.1.2 states that area and airborne radiation monitors are provided in the FHB and RXB to detect high radiation levels to support operator initiation of appropriate safety actions.

The staff determined that the preliminary FHE design is consistent with PDC 63 because of the described monitoring capabilities for FHE SSCs. The RMS is evaluated in SE section 7.6.6.

7.3.2.1.9 Sodium Heating Systems (PDC 72)

PDC 72 requires heating systems for safety-significant SSCs that contain sodium. The EVHM, EVST, BLTC, and PRC contain sodium-filled vessels or sodium-wetted fuel. PSAR section 7.3.2 describes methods of controlling passive cooling when decay heat within the EVHM, EVST, or BLTC is insufficient to maintain sodium temperatures within design limits. These methods include closing dampers to reduce cooling air flow. These major FHE components also have heating elements to maintain temperature when decay heat is inadequate.

The staff determined that the preliminary FHE design is consistent with PDC 72 because it provides an acceptably reliable means of maintaining sodium temperature in FHE components within design limits considering the availability of spent fuel decay heat, the ability to reduce heat losses through by closing dampers, and the inclusion of heating elements.

7.3.2.1.10 Sodium Leakage Detection and Reaction Prevention and Mitigation (PDC 73)

PDC 73 requires a means to detect sodium leakage and to limit and control sodium-air and sodium-concrete reactions. Systems from which sodium leakage constitutes a significant safety hazard shall include measures for protection, such as inserted enclosures or GVs.

As described in PSAR section 7.3.2, all sodium containing vessels in the FHE are equipped with seals and inert gas supplies to prevent oxygen ingress. The EVHM and BLTC are fitted with drip pans designed to collect any sodium leakage to prevent adverse reactions. These drip pans are conceptually similar to catch pans used in other SSCs, as discussed in SE section 8.3, but are intended to manage expected minor operational leaks and drips rather than significant accidental leakage.

PSAR sections 7.3.2.1.1 and 7.3.2.1.2 describe an operational capability of the EVHM and BLTC to use their respective grapple and hoist systems to interface with the drip pans and dispose of residual sodium through a movable closure valve located at the bottom of each cask. The EVST is surrounded by a GV, and the annular space is filled with inert gas and equipped with leakage detection.

The staff determined that the preliminary FHE design is consistent with PDC 73 because it provides an acceptable means to limit the extent of adverse sodium interactions and ensures that the EVST safety function would not be challenged by potential sodium leakage. The staff will evaluate operations to dispose of sodium collected in the EVHM and BLTC drip pans at the OL stage.

7.3.2.1.11 Sodium-Water Reaction Prevention and Mitigation (PDC 74)

PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water. PSAR section 7.3.2.2 states that FHE liquid sodium containing equipment is designed to be sealed, inerted, and segregated to prevent contact with water. The staff determined that the preliminary FHE design is consistent with PDC 74 because it provides an acceptable means of precluding sodium-water interactions and limiting the adverse effects of those chemical reactions at the BLTC transition to wet storage of fuel, which is addressed in SE section 7.3.1.

7.3.2.1.12 Safety Functions

7.3.2.1.12.1 Reactivity Control Functions

PSAR section 7.3.2.1 states that the EVST includes a rotating fuel storage carousel, which provides orderly storage of control assemblies within a CAP. The carousel maintains the stored spent fuel subcritical by controlling the relative spacing of fuel within the racks (i.e., geometry control). This reactivity control function has not been modeled in the PRA; instead, PSAR section 3.14 describes analyses demonstrating the prevention of criticality in fuel storage and handling. PSAR section 7.3.2.2 states that the EVST also provides a SR structural support function, which maintains the geometric arrangement of the core assemblies.

PSAR section 7.3.2.3 states that new core assemblies are preconditioned in the NAPS to support transfer to the EVST. The preliminary NAPS design accommodates up to eight fresh fuel assemblies in a geometric configuration that prevents criticality, as described in PSAR section 3.14.2.4. PSAR section 7.3.2.3 states that the NAPS is preliminarily classified as NST.

However, as discussed in SE section 5.1, USO provided supplemental information that defines a process for identifying a minimum set of design features or functions that prevent criticality and classifies those features or functions as NSRST. These resulting changes to the SSC safety classification will be reviewed at the OL stage.

As discussed in SE section 7.3.2.1.1, the EVST fuel storage carousel, as well as the supporting EVST structures, are designed to quality standards appropriate for SR structural support functions and are subject to programmatic special treatments. Based on the information and evaluation summarized above, the staff determined that the preliminary FHE design is consistent with its reactivity control functions. The classification of design features contributing to the criticality prevention function of the NAPS will be assessed at the OL stage.

7.3.2.1.12.2 Heat Removal Functions

PSAR section 7.2.3.2 identifies the following FHE heat removal functions:

- DL3-HR6 – Passive Heat Removal in EVHM (SR)
- DL3-HR7 – Passive Heat Removal in EVST (SR)
- DL3-HR8 – Passive Heat Removal in BLTC (SR)
- DL3-HR9 – Passive Heat Removal in PRC (SR)

The FHE provides SR heat removal functions for its major components that contain fuel assemblies for extended periods. Passive cooling for the EVHM and BLTC (DL3-HR6 and DL3-HR8) is achieved through radiant heat transfer from the fuel assembly (within a CAP in the EVHM) to the inner wall of the shielded cask assembly, followed by radiant and convective cooling of the assembly. The EVST passive cooling system (DL3-HR7) uses ducts to circulate air through the vault around the EVST GV via natural convection. In the PRC, passive cooling (DL3-HR9) is provided by radiation from the liquid sodium holding tank to the outer wall of the PRC. The staff determined that the preliminary FHE design is consistent with these functions because the reliability and capability of the preliminary design to perform these functions is acceptable based on the passive nature of the heat transfer mechanisms and the limited heat load associated with each vessel.

7.3.2.1.12.3 Radionuclide Retention Functions

PSAR section 7.3.2.2 identifies the following radionuclide retention functions associated with the FHE:

- DL3-RR3 – EVHM Cask Barrier (SR)
- DL3-RR3a – EVHM Transfer Barrier (SR)
- DL3-RR4 – EVST Barrier (SR)
- DL3-RR5 – BLTC Barrier (SR)
- DL3-RR5a – BLTC Transfer Barrier (SR)

- DL3-RR6 – PRC Barrier (SR)
- DL4-RR5 – EVST Guard Tank Leak Prevention Function (NSRST)

The SR EVHM and BLTC barrier functions (DL3-RR3 and DL3-RR5, respectively) refer to the radionuclide containment boundaries formed around a single fuel assembly once it is hoisted into the respective shielded cask assembly and isolated by the closure valve. In this isolated configuration, inadequate cooling is the dominant potential cause of cladding failure, which has been preliminarily determined to occur only in low frequency BDBE sequences.

The SR EVHM transfer barrier function (DL3-RR3a) refers to the barrier established when the EVHM is mated to the FFV above the RV, EVST, or PRC, and the FFV is open. This barrier includes the EVHM shielded cask assembly, its reserve argon gas supply, the FFV, and the respective vessel (i.e., RV, EVST, or PRC). When connected to the RV, the barrier also includes the CATT and the FTL; when connected to the EVST, it includes the FTA. This function contains radionuclides within the extended functional containment or confinement boundary for LBEs involving core assembly mechanical damage while the EVHM is attached to the RV or EVST.

The SR BLTC transfer barrier (DL3-RR5a) is established when the BLTC is mated to the FFV above the EVST or the PIC. For EVST transfers, this barrier consists of the BLTC shielded cask assembly, FFV, FTA, and EVST. The barrier function for the PIC interface is described in SE section 7.3.1.1.9.3. This function contains radionuclides within the confinement for LBEs involving core assembly mechanical damage while the BLTC is attached to the EVST.

DL3-RR4 refers to the EVST confinement barrier formed when the EVST is connected with the EVHM or BLTC during transfer of core assemblies to or from the EVST. This function confines radionuclide releases from potential fuel damage during core assembly handling activities when the EVST is connected to either the EVHM transfer barrier (DL3-RR3a) or the BLTC transfer barrier (DL3-RR5a). The EVST barrier function encompasses the confinement barriers up to the isolation valves leading to the EVST auxiliary systems; however, isolation valve closure is not credited for this SR function.

DL3-RR6 refers to the barrier established when the PRC is isolated as a hot cell for pin removal from an LTA or LDA, or when the PRC is connected to the EVHM for transfer of LTAs or LDAs to or from the PRC. This function confines radionuclide releases from potential fuel damage during fuel handling and pin removal activities, either within the PRC or within the extended boundary formed by the PRC with the EVHM transfer barrier (DL3-RR3a).

DL4-RR5 refers to the secondary confinement boundary formed around the EVST by its GV. This function is not associated with any LBEs but is provided for DID.

The staff reviewed the FHE functions credited for radionuclide retention and supporting documentation as part of the audit. During the review of LBEs involving the EVHM, the staff noted two BDBEs related to a loss of active cooling, RFH-LMCA-1 and RFH-LMCA-2, (described in PSAR sections 3.8.5.6 and 3.8.5.7, respectively) which have similar frequencies but differ in whether the EVHM barrier function retains the postulated release.

To confirm adequate independence of the barrier function from the passive heat removal function, as both are credited for DID, the staff audited the PRA analysis. Audit activities

revealed that the PRA analysis assumed that relief valves and dynamic seals in the shielded cask assembly would not retain radionuclides during overheating scenarios. PSAR section 7.3.2.3 was updated to clarify that an analysis will be available at the OL stage to demonstrate that radiant heat transfer ensures that the EVHM pressure boundary remains below its design temperature, even if both active and passive cooling are inadequate and cladding temperature limits are exceeded, thereby providing DID. Review of the preliminary design details in audit provides confidence that the approach will support OL demonstration of the barrier integrity with inadequate cooling because the relief valves and seals are not located in areas exposed to high radiant heat flux. At the OL stage, the staff will review the analysis demonstrating that the EVHM pressure boundary remains intact following LBEs involving inadequate cooling and potential cladding failure.

The isolated BLTC barrier function (DL3-RR5) also supports LBEs initiated by a loss of active cooling. Cladding integrity is maintained for this LBE. The staff determined that the design of the BLTC is consistent with providing this barrier function for DID.

The EVHM and BLTC transfer barrier functions (DL3-RR3a and DL3-RR5a) retain potential releases associated with mechanical damage to fuel cladding during FHE interface operations. The staff determined these functions were appropriately classified as SR and that the preliminary design provides an appropriate confinement barrier. The reliability and capability of the seals and other contributing components to these functions will be reassessed at the OL stage.

The SR EVST barrier function (DL3-RR4) also provides for radionuclide retention during fuel handling events within the EVST. The EVST design, including the main vessel, GV, and head structure provides confinement. However, since the design of EVST auxiliary systems is not fully developed, the staff will evaluate leak rate control and the confinement provided by connected auxiliary system at the OL stage.

The SR PRC barrier function (DL3-RR6) provides for radionuclide retention. As described in SE section 3.4.1, since the design is not fully developed, the staff will assess the reliability and capability of the PRC hot cell isolation, including seals and other components contributing to this function, at the OL stage.

Because the NSRST EVST GV leakage prevention function (DL4-RR5) is acceptably described, and its reliability is reasonably assured by the applied design standards, the staff determined that the preliminary design of the EVST GV is consistent with its function.

Overall, the staff determined that the preliminary design of the FHE is consistent with the identified radionuclide retention functions. However, the staff will assess the design of certain capabilities at the OL stage when final design details are available.

7.3.2.1.12.4 Additional Functions

PSAR section 7.3.2.2 identifies the provision of structural support to SR SSCs, action as a temporary load path, and external hazard protection as additional safety significant functions of FHE SSCs. The SR structural support function applies to the EVST, including the EVST main vessel, rotating carousel, and support structure and was addressed in SE section 7.3.2.1.12.1. The NSRST temporary load path function applies to the EVHM and BLTC transporters, which support their respective shielded cask assembly. It also applies to the EVHM and BLTC hoists, each of which supports a single core assembly for a short duration. This temporary support

function serves to prevent mechanical damage to the fuel cladding radionuclide retention barrier during fuel movement. The staff determined the preliminary FHE design is consistent with these structural support, temporary structural support, and protective functions because the FHE includes design features that prevent challenges to the FSFs of reactivity control and heat removal.

7.3.2.1.13 Programmatic Special Treatments

Programmatic special treatments preliminarily identified for application to the FHE are summarized in PSAR section 7.3.2.4 and include the QAP, D-RAP, RIM, EQ program, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the FHP contains SR and NSRST SSCs, the staff determined these programs are appropriate to apply to the FHE. Plant programs are evaluated further in SE chapter 8.

7.3.2.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.3.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.3.3 In-Vessel Fuel Handling System

PSAR section 7.3.3 describes the FHI, which includes the equipment necessary to manipulate the positions of core assemblies during Mode 4 refueling operations and to maintain the functional containment boundary when installed. The FHI interfaces with the FHE to transfer core assemblies into and out of the RV. The FHI is designed to:

- Shuffle core assemblies within the RV, including between core and IVS locations
- Move core assemblies to or from the IVS
- Receive core assemblies from the FHE for placement into the RV
- Discharge core assemblies from the RV to the FHE

The FHI is comprised of the following major components: in-vessel transfer machine (IVTM), in-vessel drive controller (IVDC), FTL, and fuel transfer lift shield plug (FTLSP). PSAR figure 7.3.3-1 shows the primary FHI equipment positioned within the RV and on the reactor head.

The IVTM is a multi-axis, pantograph-type fuel handling machine mounted to the RPA, which is part of the RES and located within an eccentrically positioned penetration in the reactor head. The IVDC controls the motion of the IVTM and RPA to relocate core assemblies between core locations, IVS locations, and the FTL.

The FTL is temporarily installed in the RV during core assembly transfer operations and is used to raise and lower core assemblies, contained within a CAP, between the core elevation and a position accessible to the EVHM grapple. When installed, the FTL forms part of the functional

containment boundary. When the FTL is not installed, the FTLSP provides the functional containment boundary.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the FHI:

- RG 1.100, Rev. 4;
- RG 1.244, Rev. 0;
- RG 1.87, Rev. 2;

PSAR sections 7.3.3 and 5.3 identify the following PDC as applicable to the FHI: 1, 2, 3, 4, 13, 16, 61, and 62.

7.3.3.1 Technical Evaluation

7.3.3.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected and tested to quality standards commensurate with the safety significance of the functions to be performed.

PSAR section 7.3.3.3 identified the codes and standards used for the design of FHI components. The SR portions of the IVTM, FTL, and FTLSP are designed to ASME BPVC III-5, which is consistent with the guidance in RG 1.87 for SR components in high temperature environments. The NSRST temporary load path function provided by the IVTM is designed using the guidance of ASME NML-1-2019, which is consistent with the guidance in RG 1.244 for the design of reliable material handling systems in areas with inadequate space for overhead cranes. Similarly, the NSRST temporary support provided by the FTL is designed to applicable portions of ASME NUM-1-2016, which is also consistent with RG 1.244.

Based on its review described above, the staff determined that the preliminary FHI design is consistent with PDC 1 because the codes and standards specified for the design and construction of the FHI SSCs are acceptable.

7.3.3.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

PSAR section 7.3.3.2 states that FHI SSCs are designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The SR RXB substructure, described in PSAR section 7.8.1.1, protects the FHI against high winds, external missiles, and flooding hazards. The capability of FHI components to perform their safety-significant functions will be established using the methods described in PSAR section 6.1.1 to evaluate natural phenomena and determine associated DBHLs.

PSAR table 7.3.3-1 states that the IVTM, FTL, and FTL shield plug are classified as SCS1 based on their containment functions. The IVTM and FTL also provide temporary structural support for core assemblies and thus are classified as NSRST and SCN3. These classifications

reflect the relative importance of the containment function in preventing releases during in-vessel fuel handling accidents as well as the assumed reliability of the temporary structural support function in preventing these postulated accidents, which are modeled as DBEs.

The staff determined that the preliminary FHI design is consistent with PDC 2 because, taken together, the RXB and FHI SSCs ensure that safety-significant FHI SSCs retain their capability to perform safety functions under the effects of natural phenomena. The RXB provides a barrier against most external hazards, and safety-significant FHI SSCs are designed to withstand postulated seismic events.

7.3.3.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.3.3.2 states that the FHI incorporates design features that minimize the potential for liquid sodium leakage to its surroundings, thereby limiting the potential for fires. The FHI is constructed of noncombustible and non-flammable materials to the extent practical. The staff determined that the preliminary FHI design appropriately minimizes the probability and effects of fires and explosions by minimizing the potential for sodium leakage during refueling operations, consistent with PDC 3.

7.3.3.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions.

PSAR section 7.3.3.2 states that FHI SSCs are designed to accommodate the effects of, and to be compatible with, environmental conditions associated with spent fuel decay heat and radiation, as well as exposure to liquid sodium and its aerosols, vapors, and oxidation products. Materials used in FHI SSCs are selected to withstand the operating environments to which they are exposed.

As noted in PSAR section 7.3.3, the SR RXB substructure provides protection against external hazards, including missiles. To prevent adverse interactions during installation and removal of the IVTM, the IVTM nozzle flange is designed to mate with the RES structure and prevent the IVTM from dropping into the core area following a handling system failure.

The staff determined that the preliminary design of the FHI is consistent with PDC 4 because, as described above, the safety-significant portions of the FHI system are appropriately protected from external hazards and are designed to accommodate the effects of environmental conditions associated with normal operations and postulated accidents.

7.3.3.1.5 Instrumentation Supporting Safety-Significant Functions (PDC 13)

PDC 13 requires that instrumentation is provided to monitor variables and systems over their anticipated ranges during normal operation and accident conditions, including those that can affect functional containment.

PSAR section 7.3.3.2 states that the FHI positions and supports the instrumentation necessary to calibrate and control in-vessel motion of the IVTM and to identify of core assemblies. FHI-related instrumentation also monitors containment barrier seal gas. The FHC provides

supervisory, control, and data acquisition functions to coordinate, monitor, and control machine movements during initial fuel loading, subsequent refueling, placement of non-fuel core assemblies, and the transfer of spent fuel and components from the reactor.

The NST FHC supports the identification of core assemblies, execution of fuel movements, and enforcement of interlocks (i.e., seating of core assemblies in proper locations). However, as discussed in SE section 5.1, USO provided supplemental information (ML25253A386) defining a process for identifying a minimum set of design features or functions that prevent postulated initiating events and classifying those features or functions as NSRST. In audit discussions, certain FHC functions, such as preventing core assembly drops from fuel handling machines and ensuring proper placement of fuel assemblies in the core, were preliminarily identified as preventive measures that contribute to DID.

The staff determined that the preliminary FHI design is consistent with PDC 13 because the information provided demonstrates appropriate instrumentation to support normal operations and maintain functional containment under accident conditions. Changes to the FHC safety classification for preventive functions will be reviewed at the OL stage.

7.3.3.1.6 Containment Design (PDC 16)

PDC 16 requires that a reactor functional containment be provided to control the release of radioactivity to the environment and to ensure that the safety-significant design conditions for the functional containment are not exceeded for the duration of accident conditions.

Radionuclide retention is implemented using a functional containment strategy, which consists of multiple barriers to control the release of radioactivity to the environment. The IVTM, FTL, and FTL shield plug function as a passive part of this strategy by maintaining the integrity of the reactor barrier during refueling operations. The staff determined the preliminary FHI design is consistent with PDC 16 because FHI SSCs are designed to ensure that their safety-significant design conditions are not exceeded for the duration of postulated accident conditions. Additionally, FHI component seals can be individually monitored to ensure appropriate control of containment barrier leakage.

The staff reviewed LBEs associated with function DL3-RR1 in PSAR table 5.2-3 and their descriptions in PSAR chapter 3 to identify the FHI components credited for achieving the evaluated dose consequences for postulated releases of radioactivity. The staff determined the preliminary FHI design is consistent with PDC 16 because the evaluations indicate that FHI components contribute to the RES isolation function (DL3-RR1) that controls the release of radionuclides to the environment.

7.3.3.1.7 Fuel Storage and Handling and Radioactivity Control (PDC 61)

PDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of safety-significant components, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the safety significance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.

PSAR section 7.3.3.2 describes how, in conjunction with the RES, FHI components provide a containment barrier function and radiation shielding during refueling operations. PSAR section 1.1.4.3.19 discusses the NI major maintenance equipment used to maintain the inert environment within the RES and to prevent excess leakage into the RXB during installation and removal of the in-vessel transfer machine and FTL. The FHI system handles core assemblies beneath the surface of the liquid sodium within the RV, which provides cooling for the core assemblies. The staff determined that the preliminary FHI design is consistent with PDC 61 because it provides appropriate containment, cooling, and shielding capabilities.

7.3.3.1.8 Prevention of Criticality in Fuel Storage and Handling (PDC 62)

PDC 62 requires that criticality in the fuel storage and handling system be prevented through physical systems or processes, preferably by using geometrically safe configurations. PSAR section 7.3.3.2 states that IVS locations outside the reactor core maintain stored spent fuel subcritical by controlling the relative spacing within the racks (i.e., geometry control). The staff evaluated the criticality prevention measures for the IVS in SE section 3.13. The IVTM handles a single core assembly at a time.

The staff determined that the preliminary FHI design provides an acceptable means of criticality prevention based on the evaluation of the supporting criticality analysis in SE section 3.13, and is therefore consistent with PDC 62.

7.3.3.1.9 Safety Functions

PSAR section 7.3.3.2 states that the FHI has one SR function, DL3-RR1f, "FHI Seal," related to radionuclide retention. DL3-RR1f refers to the radionuclide containment boundary formed by the interface of the IVTM and FTL with the rotatable plug assembly and reactor head, respectively, and the interface of the FTLSP with the FTL bore. The IVTM, FTL, and FTLSP thereby form part of the RES functional containment boundary in the refueling operational state when the IVTM and FTL are installed. PSAR section 7.3.3.2 also identifies additional safety-significant functions for the FHI SSCs, including providing structural support to SR SSCs and serving as a temporary load path. The NSRST temporary load path function applies to the IVTM and FTL, which support the core assemblies during relocation in support of refueling operations.

The staff determined the preliminary FHI design is consistent with the radionuclide retention function DL3-RR1f because the FHI components contribute to the RES isolation function that controls the release of radionuclides to the environment by providing structural support and a sealing interface between the reactor head and FHI components. Additionally, the staff found that the FHI design provides temporary structural support to the core assemblies during handling within the reactor vessel, which helps to maintain the frequency of fuel handling events appropriately low.

7.3.3.1.10 Programmatic Special Treatments

Programmatic special treatments preliminarily identified for application to the FHI are summarized in PSAR section 7.3.3.4 and include the QAP, D-RAP, RIM program, EQ program, CVAP, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the FHI contains SR and NSRST SSCs, the staff determined these programs are appropriate to apply to the FHI. Plant programs are evaluated further in SE chapter 8.

7.3.3.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.3.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.4 Radwaste Systems

PSAR section 7.4 describes the NSRST RWG. The liquid radwaste processing system (RWL) is described in PSAR section 9.1 and is classified as NST. The solid radwaste processing system (RWS) is described in PSAR section 9.3 and is classified as NST.

7.4.1 Gaseous Radwaste Processing System

PSAR section 7.4.1 describes the RWG, which processes the gaseous waste streams from both the primary and intermediate SCGs. The RWG provides holdup for the decay of short-lived radionuclides using activated charcoal adsorption beds. The RWG is located below grade in the FHB.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the RWG:

- 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents – nuclear power reactors;”
- RG 1.140, Rev. 3; and
- RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Rev. 2 (ML013100305).

PSAR sections 7.4.1 and 5.3 identify the following PDC as applicable to the RWG: 1, 2, 3, 4, 60, 61, 63, and 64. The staff also identified PDC 16 as applicable to the RWG.

7.4.1.1 Technical Evaluation

7.4.1.1.1 Equipment Design Requirements (PDC 1, 2, 4, and 61)

PDC 1 requires safety-significant structures, systems, and components to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PDC 2 requires safety-significant SSCs to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions

PSAR section 7.4 states that the RWG is classified as NSRST and that the KU1 radioactive waste treatment systems are designed to meet the applicable guidance of RG 1.143. The

NSRST classification of the RWG is based on the need for DID adequacy, as identified in PSAR table 5.2-4. PSAR sections 7.4.1.1 and 7.4.1.2 describe the design guidance used for the RWG. PSAR section 7.4.1.2 also states that the KU1 facility is in full conformance with RG 1.140, and partial conformance with RG 1.143. Sections C.5 and C.6 of RG 1.143 are replaced by the methodologies discussed in PSAR sections 5.1 and 6.1.3. PSAR section 7.4.1.3.3 states that the RWG equipment and components are designed, constructed, and tested in accordance with the codes and standards referenced in RG 1.143.

PSAR section 7.4.1.2 further states that the NSRST portions of the RWG are designed to SCN1, consistent with PSAR section 6.4.1.1, which states that SCN1 is initially assigned to components that perform an NSRST function following the SSE. PSAR section 6.1.3 discusses the DBHL requirements for NSRST SSCs, referencing American Society of Civil Engineers (ASCE) 7-16. PSAR table 6.4-1B provides the seismic criteria for SCN1 classification. PSAR table 7.4.1-1 provides the list of codes and standards applicable to the RWG design.

PSAR sections 3.7.3.6 and 3.7.3.7 describe the DBE analysis performed for a release from the RWG. PSAR section 3.9.4.5 provides the postulated DBA for an RWG leak and states that the mechanistic source term is modeled as a release of holdup tank contents to downstream RWG piping and the plant vent stack over a 10-minute period.

The staff reviewed PSAR section 7.4.1 and its references to RG 1.143 and RG 1.140. RG 1.143 provides acceptable codes and standards for the design of SSCs in radwaste facilities, while RG 1.140 describes an acceptable method for meeting NRC's regulations for the design, inspection, and testing of normal atmosphere cleanup systems for controlling airborne radioactive releases to the environment during normal operations and AOOs. RG 1.140 also discusses acceptable standards for the design and testing of the RWG, including ASME AG-1-2009.

The staff determined that the codes and standards provided in PSAR table 7.4.1-1 are consistent with those listed in RG 1.140 and RG 1.143. The staff determined that RG 1.140 is applicable to the RWG design because it supports the assessment of design requirements for systems containing radioactivity and ensures appropriate containment, confinement, and filtering, consistent with PDC 61. The staff also determined that RG 1.143 is appropriate because it is consistent with design guidance previously used for similar systems in LWRs. Reference to the methodologies in PSAR sections 5.1 and 6.1.3 are also appropriate given the safety classification approach described in NEI 18-04 and the RWG's NSRST classification.

The staff reviewed RWG-related events to determine if its NSRST classification is appropriate. Based on PSAR sections 3.7.3.6, 3.7.3.7, and 3.9.4.5, the associated DBEs and DBA reflect a holdup tank release with minimal or no barriers to mitigate release to the environment. As discussed in SE section 3, the staff determined that the NSRST classification of the RWG is appropriate because LBEs involving the RWG consider minimal or no retention of radionuclides and releases from these events result in a 2-hour exclusion area boundary (EAB) total effective dose equivalent (TEDE) dose far below the accident dose criteria in 10 CFR 50.34(a)(1)(ii)(D). Specifically, the RWG release DBA results in a 2-hour EAB TEDE dose of 9.59E-2 rem and 30-day low population zone (LPZ) boundary TEDE dose of 9.59E-2 rem, both significantly below regulatory limits.

The staff determined that the preliminary RWG design is consistent with PDC 1, 2, 4, and 61 because the PSAR states full conformance with RG 1.140 and partial conformance with

RG 1.143. The elements of RG 1.143 that are not addressed (sections C.5 and C.6) are covered by the discussions in PSAR sections 5.1 and 6.1.3 on NSRST safety classifications.

7.4.1.1.2 Fire Protection (PDC 3)

PDC 3 requires that safety-significant SSCs be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.4.1.2 states that the RWG is designed to minimize the probability and effect of fires through the use of non-combustible and non-fire sustaining materials to the extent practical. PSAR section 7.5.2 discusses fire detection and firefighting features. PSAR section 7.5.2.3 states the fire protection program and the fire hazards analysis will be provided at the OL stage. In addition, the PSAR states that a fire protection program which addresses SSCs that could affect safety will be provided in the OL application. The staff determined that the preliminary RWG design is consistent with PDC 3 due to the use of low combustible materials and the planned fire protection program.

7.4.1.1.3 Functional Containment (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents.

PSAR section 3.9.4.5 provides the postulated DBA event for an RWG leak. The mechanistic source term is modeled as a release of holdup tank contents to downstream RWG piping and the plant vent stack over a 10-minute period. The resulting radiological consequences are $9.59\text{E-}2$ rem for both the 2-hour EAB and the 30-day LPZ.

PSAR section 7.4.1.2 describes the functional requirements and design criteria related to the RWG's NSRST function, DL2-RR4, "Retain Radionuclide – RWG Radionuclide Boundary." This function requires that RWG components (including seals) form a radionuclide confinement boundary that prevents the uncontrolled release of radionuclides.

The staff reviewed PSAR sections 3.9.4.5 and 7.4.1.2 and determined that the RWG confinement boundary credited in the DBA analysis is consistent with the NSRST function described. Radiation monitoring is not credited to perform an isolation function for this system because the associated LBEs either bypass or assume direct release from the holdup tank through the RWG piping to the plant stack.

The staff determined that the preliminary RWG design is consistent with PDC 16 and DL2-RR4 because the postulated DBA for an RWG leak does not result in radiological releases exceeding the limits specified in 10 CFR 34(a)(1)(ii)(D).

7.4.1.1.4 Gaseous Radwaste Processing System Description (10 CFR 50.34a, PDC 60)

As specified in 10 CFR 50.34a, a CP application shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations.

PDC 60 requires that the design includes means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including AOOs. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where

unfavorable site environmental conditions may impose unusual operational limitations on effluent releases.

PSAR sections 7.4.1.1 and section 7.4.1.3.1 describe the RWG. Specifically, the PSAR states that the RWG filters particulates, provides holdup and adsorption to allow radioactive decay of both short and long-lived isotopes, and discharges to the NHV plant exhaust stack, from which the effluent is released to the environment as a monitored release. PSAR section 7.4.1.3.1 also describes the components of the RWG, as shown in PSAR figure 7.4.1-1.

The RWG is composed of the following equipment:

- an inlet radiation monitoring,
- vacuum tank,
- two particulate filters (upstream of each compressor),
- two compressors,
- aftercooler,
- holdup tank,
- high-efficiency particulate air (HEPA) filter before the charcoal delay beds,
- four charcoal delay beds configured in parallel,
- radiation monitoring to verify charcoal delay bed performance, and
- second HEPA filter after the delay beds

PSAR section 7.4.1.3.1 states that the RWG is designed to delay flow for 24 hours to allow for the decay of Ar-41. The charcoal delay beds are designed to provide holdup and decay for radioactive isotopes of argon, krypton, and xenon, and are housed within a shielded room. PSAR table 9.1-6 provides the holdup times being provided for these isotopes.

The staff reviewed PSAR section 7.4.1, including the RWG's functions to filter particulates and provide holdup and adsorption of both short and long-lived isotopes. The system components provided are consistent with other RWG systems seen in LWR designs for the removal of particulates, such as Rb-88 and Cs-137, and the holdup and adsorption of gases such as Ar-41, Kr-85, Xe-131m and Xe-133.

The staff determined that the RWG description is sufficient to meet the preliminary design information required in 10 CFR 50.34a and demonstrate consistency with PDC 60 because the PSAR provides adequate design information demonstrating the RWG's ability to control gaseous effluent and describes features for radiation monitoring at the RWG's inlet and at the outlet of the charcoal delay beds to verify adsorption performance, ensuring continued control of radioactive material releases in gaseous effluents during plant operations.

7.4.1.1.5 Radiation Shielding Considerations for the RWG (PDC 61)

PDC 61 requires that fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions and to ensure suitable shielding is provided for radiation protection.

PSAR section 7.4.1.3.1 states that the charcoal delay beds are housed within a shielded room. PSAR section 10.3 notes that additional details regarding design considerations to maintain occupational exposures as low as reasonably achievable (ALARA) will be provided at the OL stage. PSAR sections 7.4.1 and 10.3 indicate that dose analyses showing that occupational exposures are ALARA will be provided at the OL stage. To support the shielding design reviews for PDC 61, the staff reviewed information made available during the audit to support the staff's understanding of available shielding. The audit determined that, while the radiation zone information is preliminary, USO has established radiation zoning criteria to ensure occupational exposures remain ALARA for the OL application.

The staff determined that the preliminary RWG design is consistent with PDC 61 because the charcoal delay beds are housed within a shielded room and USO has committed to providing further shielding information at the OL stage aligning with ALARA for occupational exposures. The staff will review the final RWG design at the OL stage to ensure adequate radiation zoning is established.

7.4.1.1.6 Monitoring Radioactive Releases (PDC 63 and 64)

PDC 63 requires that the design be capable of detecting conditions that may result in excessive radiation levels and initiating appropriate safety actions. PDC 64 requires that the RWG be designed to monitor radiation levels and radioactivity in effluents, as well as detect radioactive leaks and spills during routine operations, including AOOs.

PSAR section 7.4.1.3.1 states that radiation monitoring is provided at the inlet of the vacuum tank and at the outlet of the delay beds to verify their performance. Effluent monitoring is also provided at the NHV plant stack. Additionally, PSAR section 7.6.6.1 states that effluents from the RWG and NHV are routed to the plant stack, where radiation monitoring is performed. This stack includes a tritium sample skid to ensure tritium releases are monitored. PSAR section 7.6.6.3.3 further states that the stack will have two redundant wide range gas monitors for the monitoring plant gaseous effluents.

The staff reviewed the information contained in PSAR sections 7.4.1 and 7.6 to assess the monitoring capabilities available for tracking gaseous effluent releases. The staff determined that the preliminary RWG design is consistent with PDC 63 and 64 because it includes radiation monitoring at the RWG inlet and outlet, as well as effluent release points to the environment.

7.4.1.1.7 Safety Functions

7.4.1.1.7.1 Radionuclide Retention

PSAR section 7.4.1.2 states that the RWG has one NSRST function, DL2-RR4. This function consists of the RWG components (including seals) that form the radionuclide confinement boundary which prevents the uncontrolled releases of radionuclides. PSAR table 5.2-4 states that this function is required for DID adequacy. The staff review of DL2-RR4 is contained in SE section 7.4.1.1.3 as part of the RWG functional containment evaluation.

7.4.1.1.8 Programmatic Special Treatments

Programmatic special treatments applied to the RWG are summarized in PSAR section 7.4.1.4 and include the QAP, D-RAP, EQ program, PITAP, and RIM program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs and the staff determined that they are thus appropriate to apply to RWG. Because the RWG contains NSRST SSCs, the staff determined these programs are appropriate to apply to the RWG. Plant programs are evaluated further in SE chapter 8.

7.4.1.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.4.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.5 Ancillary Systems

PSAR section 7.5 describes the following ancillary systems:

- NHV
- NI fire protection system (NFP)

7.5.1 Nuclear Island Heating, Ventilation, and Air Conditioning Systems

PSAR section 7.5.1 describes the NHV, which provides cooling to safety-significant equipment under normal operating conditions and controls environmental conditions within portions of the following five NI buildings: RXB, RAB, FHB, fuel auxiliary building (FAB), and NCB, with specific functions assigned to the NHV in each building.

The RXB, RAB, FHB, and FAB contain radiologically controlled areas (RCAs). The NHV is configured to control airflow in these buildings such that the buildings are maintained at a negative pressure, ensuring air moves from areas of lower potential contamination to areas of higher potential contamination. Air from the RCAs is collected in a combined release path, filtered, monitored for radioactivity, and exhausted through the main plant stack.

The NHV supports the functional containment of radionuclides by providing for the isolation of interfacing piping and ducting with the reactor HAA, secondary enclosures around SCG exhaust treatment components, and secondary enclosures around SPS components. Isolation dampers and valves connected to these areas are designed to fail closed. The NHV also supports radionuclide retention by providing HEPA filtration of the FHB exhaust to reduce the consequences of postulated radionuclide releases within the FHB. USO classified these components as NSRST.

The NHV also controls environmental conditions in the NCB and provides protection for control room operators and equipment. Sensors in the NHV intake alert operators to airborne hazards, enabling isolation of the main control room (MCR) envelope and the below grade portion of the NCB that contains the remote shutdown complex (RSC). Under accident conditions, the NHV provides conditioned, HEPA-filtered air to the MCR and contains manually operated compressed breathable air bottles in the RSC to support PAM. USO classified the NHV SSCs that provide control of environmental conditions in the MCR and RSC and support PAM as NSRST.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the NHV:

- 10 CFR 20.1406, “Minimization of contamination;”²⁵
- RG 1.13, Rev. 2;
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” Rev. 0, (ML031530505);
- RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” Rev. 2, (ML21253A071);
- RG 1.140, Rev. 3;
- RG 1.189, Rev. 5;
- RG 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors,” Rev. 1 (ML031490611);
- RG 1.97, Rev. 5; and
- RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” Rev. 3 (ML003739549).

PSAR sections 7.5.1 and 5.3 identify the following PDC as applicable to the NHV: 1, 2, 3, 4, 16, 19, 44, and 64. The staff also identified PDC 13, 45, and 46 as applicable to the NHV.

7.5.1.1 Technical Evaluation

7.5.1.1.1 Quality Standards and Records (PDC 1)

²⁵ As provided in 10 CFR 20.1002, “Scope,” the regulations in 10 CFR Part 20, “Standards for Protection Against Radiation,” apply to “persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material or to operate a production or utilization facility....” USO has applied for a CP, which does not provide a license to operate the facility. In the CP application, USO also has not applied for licenses to receive, possess, use, transfer, or dispose of any byproduct, source, or special nuclear material at the facility. Therefore, the staff did not evaluate whether the requirements in 10 CFR Part 20 would be met in support of the CP. Instead, the staff assessed whether USO had identified the relevant requirements for an operating facility and provided descriptions of how the preliminary facility design will, to the extent practicable, minimize contamination of the facility and the environment, as required for operating facilities by 10 CFR 20.1406. This is consistent with 10 CFR 50.40(a), which provides that in determining whether a CP may be issued, the Commission will be guided by consideration of reasonable assurance that USO will comply with the NRC’s regulations, including the regulations in 10 CFR Part 20, and that the health and safety of the public will not be endangered.

PDC 1 requires safety-significant structures, systems, and components to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.5.1.3 states that NSRST portions of the NHV are designed to ASME AG-1-2009 (including the 2010 addenda 1a and 2011 addenda 1b) or, for piping, ASME B31.1-2022, to ensure isolation and retention or controlled release path functions can be performed. The NRC endorsed ASME AG-1-2009 and its addenda in RG 1.140 and PSAR section 7.5.1.2 states that the design of the NHV is in full conformance with the guidance in RG 1.140. Although PSAR section 8.1.1 states that KU1 uses NEI 18-04 as the basis for plant design SSC classifications rather than RG 1.26, the staff determined that the use of ASME B31.1-2022 for NHV piping is acceptable because the code is identified in RG 1.26 as appropriate for portions of systems that contain or may contain radioactive material that do not meet the criteria of RG 1.26 quality groups A, B, or C, which is consistent with the NHV design.

The staff determined that the preliminary NHV design is consistent with PDC 1 because the codes and standards specified for the design and construction of the NHV isolation, retention, and controlled release path components are acceptable based on its preliminary design and safety functions.

7.5.1.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

PSAR section 7.5.1.2 states that safety-significant NHV equipment is located within structures that provide protection from natural phenomena. The NHV isolation dampers and valves, which isolate potential radionuclide release paths, are located near the SCG and SPS enclosures in the SR RXB and NSRST RAB substructures as well as the HAA. The RSC, including the air bottles used for its pressurization, is located in the SR NCB substructure. PSAR section 6.4.2.3.1 describes how these areas are protected from the effects of natural phenomena, including missiles. Additionally, PSAR sections 7.8.2.2.2. and 7.8.4.2.2 state that the FHB and NCB superstructures, respectively, are designed to protect NSRST SSCs from the effects of external hazards defined in PSAR section 6.4.2, without the loss of structural integrity.h

PSAR section 7.5.1.2 also states that the NSRST portions of the NHV are designed to seismic classification SCN1. The analysis methods used to demonstrate conformance with this classification are described in PSAR section 6.4.1.

Because the safety-significant NHV SSCs are located within protective structures and are classified SCN1, the staff determined that the preliminary NHV design is adequately protected from, or designed to withstand, the effects of natural phenomena, consistent with PDC 2.

7.5.1.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. The NHV design includes provisions to minimize the effects of fires. PSAR section 7.5.1.2 states that the NHV is designed to incorporate noncombustible materials to the extent practical and responds to signals received from the fire protection system. PSAR section 7.5.2.3.5 states that the NHV responds to the detection of smoke within system ducts by shutting down the associated air handling unit and closing local smoke dampers to prevent the spread of smoke.

The staff determined that the preliminary NHV design is consistent with PDC 3 because its design basis includes features that minimize the effects of potential fires by using noncombustible materials and providing for the isolation of the effects of fires.

7.5.1.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.5.1.2 states that the NHV components located in areas where airborne contaminants or sodium may be present are either selected for compatibility or have protective coatings applied to mitigate adverse interactions, ensuring that their safety-significant functions are not compromised.

Because the safety-significant portions of the NHV are appropriately protected from, or designed to accommodate, the effects of environmental conditions associated with normal operations and postulated accidents, the staff determined that the preliminary NHV design is consistent with PDC 4.

7.5.1.1.5 Instrumentation Supporting Safety-Significant Functions (PDC 13)

PDC 13 requires that instrumentation is provided to monitor variables and systems over their anticipated ranges during normal operation and accident conditions, including those that can affect functional containment.

PSAR section 7.5.1.5 identifies the instrumentation that supports the automatic closure of certain NHV isolation valves and dampers and alerts operators to initiate manual actions. Detection of high radiation, toxic chemicals, or sodium vapor in the fresh air intake of the MCR produces an indication to MCR operation personnel, prompting them to evaluate the condition and manually isolate the control room heating, ventilation, and air conditioning (HVAC) system. These manual actions include aligning the NCB ventilation to the isolation (recirculation) mode for the MCR and RSC as well as closing isolation valves and dampers.

Automatic isolation of the NHV at the barrier around the SPS-P and SCG cells occurs upon detection of a leak in the respective enclosure. Additionally, automatic isolation of the vapor trap cell cooling supply from the NHV occurs on detection of a primary coolant pressure relief valve opening. High area radiation detectors located in the SPS cells, SCG cells, and HAA provide further information to support manual operator initiation of barrier isolation.

Because the preliminary NHV design supports manual isolation of the MCR and RSC, allows for automatic isolation of the SPS-P and SCG cells, and provides high area radiation signals to operators, the staff determined that it contains appropriate instrumentation to support functional containment DID under accident conditions, consistent with PDC 13.

7.5.1.1.6 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment be provided to control the release of radioactivity to the environment and ensure that the functional containment safety-significant design conditions are not exceeded for as long as postulated accident conditions require.

Radionuclide retention under accident conditions is implemented using a functional containment strategy, which consists of multiple barriers that control the release of radioactivity to the environment. The NHV contributes to this function by isolating potential sources of radionuclides and providing HEPA filtration of radionuclide releases.

The staff reviewed LBE evaluations in PSAR chapter 3 to identify the NHV barriers credited in achieving the evaluated dose consequences for postulated radioactive releases. The relevant LBEs involved failures affecting the SPS, SCG, and RWG, as described in PSAR sections 3.7.3 and 3.8.4, as well as fuel handling accidents described in PSAR section 3.7.4. The evaluations indicate that the NHV contributes to radionuclide isolation and filtration by controlling radioactive releases to the environment.

Based on the staff's review of NHV radionuclide retention barriers discussed above, the staff determined that the preliminary NHV design is consistent with PDC 16. The staff notes that the actual performance of NHV SSCs was generally not modeled in these analyses, as discussed in SE section 7.5.1.1.13; the staff will review this aspect when detailed design information is available at the OL stage.

7.5.1.1.7 Control Room (PDC 19)

PDC 19 requires, in part, that adequate radiation protection be provided to permit access to and occupancy of the control room under accident conditions. It also requires that adequate habitability measures be provided to permit access to and occupancy of the control room during normal operations and under accident conditions. Additionally, PDC 19 requires that equipment be provided at appropriate locations outside the control room, with the capability to support prompt shutdown and transition to safe shutdown conditions through use of suitable procedures.

PSAR section 5.3.2.10 describes the following NHV capabilities:

- During normal conditions, the NHV provides HVAC to the MCR. Aerosol detectors, smoke detectors, and radiation detectors are provided in the MCR outside air duct.
- If instrumentation alerts operators to high levels of radiation or aerosols (including sodium aerosols), operators manually isolate the MCR HVAC system and place it in recirculation mode.
- In the event of an MCR evacuation, operators scram the reactor, perform any required manual actions and evacuate to the RSC, where safe shutdown can also be achieved using equipment located near the RSC.

PSAR section 7.5.1.2 states that the NHV design provides suitable environmental conditions for normal MCR operation and ensures control room habitability is maintained under accident conditions involving radiological, sodium aerosol, and chemical hazards. PSAR section 7.5.1.3 identifies isolation of the MCR envelope and the below grade portion of the NCB as an NSRST function. The MCR HVAC recirculation mode uses NSRST HEPA filters to minimize operator dose. However, PSAR section 5.3.2.10 states that MCR HVAC isolation (recirculation mode) is not necessary to satisfy PDC 19 dose objectives for personnel in the control room during DBAs.

Regarding MCR radiological habitability, PSAR section 5.3.2.10 states that radiological dose analyses for MCR occupants during DBAs are evaluated using conservative control room

atmospheric dispersion factors and DBE source terms. These analyses credit the existence of buildings as well as HEPA filtration by the FHB HVAC system prior to release to the environment. PSAR section 5.3.2.10 also states that the use of DBE source terms and NSRST SSCs for dose mitigation is acceptable because no operator actions from the MCR are necessary for the successful performance of SR functions.

During the audit, the staff reviewed the MCR radiological dose analyses supporting the PSAR and confirmed that DBE source terms were used to model dose to occupants in the control room. The staff also confirmed that the analyses were performed using methods consistent with the control room evaluation model described in the NRC-approved TR, NAT-9391-A, "Radiological Release Consequences Methodology Topical Report," Rev. 0 (ML24208A181). Although the approved methodology includes clarifying revisions made after the performance of the MCR dose analysis described in the PSAR, the staff found no differences that would make the PSAR analyses inconsistent with the approved approach.

The staff's audit confirmed that the MCR dose analyses used conservative MCR atmospheric dispersion factors, developed in accordance with RG 1.194. These factors were not based on site-specific meteorological data but were conservatively developed to envelop a range of reactor sites, as described in PSAR section 3.3.1.4. The staff confirmed in audit that the values used in the analysis differed from the site characteristic MCR atmospheric dispersion values reported in PSAR table 2.1-2 but noted that they are higher than the site characteristic values for KU1 across all time periods, making the enveloping MCR atmospheric dispersion factors bounding for the site. Because dose is directly proportional to the atmospheric dispersion factor, the PSAR MCR dose estimates are higher than those that would be calculated using the site-specific values, therefore making the preliminary MCR consequence analyses bounding for the site. The staff's audit of the supporting calculations confirmed that the preliminary analyses show that the MCR dose is within the PDC 19 control room radiological habitability criterion. The staff anticipates that the MCR habitability analyses may be revised to support the OL application, with a description of the analysis of the final MCR design and dose results in the final safety analysis report.

The staff determined that the preliminary NHV design is consistent with PDC 19 because the control building portion (1) provides acceptable radiation protection and habitability measures under both normal and accident conditions for both the MCR and RSC and (2) supports safe reactor shutdown from the RSC using NHV SSCs separate from the MCR ventilation system to provide radiation protection and habitability control measures. I&C design capabilities supporting these functions are addressed in SE section 7.6.6.

7.5.1.1.8 Structural and Equipment Cooling (PDC 44)

PDC 44 requires that the NHV transfers heat from safety-significant SSCs to an ultimate heat sink, as necessary, to accommodate the combined heat load of these SSCs under normal operating and accident conditions.

PSAR section 5.3.4.9 states that the NHV cooling function removes heat from safety-significant SSCs during normal operating conditions. During accident conditions, these SSCs rely on passive cooling. The PSAR further states that active cooling by the NHV is not required to support the function of safety-significant SSCs during accident conditions. In audit discussions, the applicant clarified that the only SR SSCs expected to require passive cooling through heat transfer to passive heat sinks are those providing structural support functions in the RXB and FHB substructures. USO stated that analysis will be performed at the OL stage to determine

whether additional insulation or monitored local temperature limits are necessary to ensure adequate passive cooling. Instrumentation located in the NCB substructure that performs SR functions (e.g., reactor trip and pump trip functions) is designed to be fail-safe on loss of AC power and completes its function before room heat-up from a loss of NHV active cooling becomes a concern.

Because the NHV provides active cooling during normal operations to safety-significant SSCs, the staff determined that the preliminary NHV design describes an acceptable means of transferring heat from safety-significant SSCs to an ultimate heat sink during normal operations and, therefore, is consistent with PDC 44. The staff will assess local analyses at the OL stage to confirm that functions in the RXB and FHB substructure establish necessary conditions for adequate passive heat removal during accidents.

7.5.1.1.9 Inspection of Structural and Equipment Cooling Systems (PDC 45)

PDC 45 requires that structural and equipment cooling systems allow for the periodic inspection of important components to ensure their integrity and capability. PSAR section 5.3.4.10 states that the NHV is designed to permit appropriate periodic inspection of its NST heat removal functions that support the normal operation of safety-significant SSCs. The staff determined that the preliminary NHV design is consistent with PDC 45 because of this described inspection capability.

7.5.1.1.10 Testing of Structural and Equipment Cooling Systems (PDC 46)

PDC 46 requires that structural and equipment cooling systems be designed to permit appropriate periodic functional testing to ensure (1) the structural integrity of their components, (2) the operability and performance of the system components, and (3) the operability of the systems as a whole. PSAR section 5.3.4.11 states that the NHV is designed to permit appropriate periodic inspection of its NST heat removal functions that support normal operation of safety-significant SSCs.

The staff determined that the preliminary NHV design is consistent with PDC 46 because safety significant SSCs are passively cooled and do not rely on active structural and equipment cooling systems during DBAs and the design features described under PDC 46 in PSAR section 5.3.4.11 are adequate to ensure structural integrity and operability of the NST NHV cooling function.

7.5.1.1.11 Monitoring Radioactivity Releases (PDC 64)

PDC 64 requires, in part, that effluent discharge paths be designed to monitor radioactivity that may be released from normal operations, including AOOs, and postulated accidents.

PSAR section 7.5.1.2 states that the NHV controls the distribution of negative pressure within buildings containing radiological controlled areas, ensuring air flows from areas of lesser potential contamination to areas of higher potential contamination. Exhaust from these areas within the RXB, FHB, RAB, and FAB passes through a combined stack for effluent monitoring by the RMS, as described in PSAR section 7.6.6. The NHV also provides a filtered release path, considered in evaluation of several LBEs, that allows monitoring of releases during postulated accident conditions.

Because the preliminary NHV design allows for monitoring radioactivity from effluent discharge paths during both normal operations and accident conditions, the staff determined that it is consistent with PDC 64.

7.5.1.1.12 Consideration of 10 CFR Part 20 Requirements for Operation

As specified in 10 CFR 20.1406, applicants for OLs must describe how facility design and procedures will minimize contamination of the facility and the environment, as well as control the concentrations of radioactive material in the air. PSAR section 7.5.1.2 states that the NHV controls the distribution of negative pressure within buildings containing radiological controlled areas, ensuring air flows from areas of lesser potential contamination to areas of higher potential contamination. This design provides for control of contamination by minimizing the likelihood of contamination spreading from more contaminated to less contaminated areas. This capability is consistent with regulatory position C.2.d of RG 8.8. The staff notes that the approach described in the PSAR appears consistent with 10 CFR 20.1406 but is not making a determination and will evaluate this approach at the OL stage.

7.5.1.1.13 Safety Functions

PSAR section 7.5.1.2 identifies the following radionuclide retention functions associated with the NHV:

- DL4-RR1 – Ex-RES Functional Containment Barrier (NSRST)
- DL4-RR1a – HAA Barrier (NSRST)
- DL4-RR1c – HAA HVAC Operations Following Postulated Release (NSRST)
- DL4-RR3 – SPS Cells Barrier (NSRST)
- DL4-RR3c – SPS Cells Barrier Isolation on Leak Detection (NSRST)
- DL4-RR4 – SCG Cells Barrier (NSRST)
- DL4-RR4b – SCG Cells Barrier Isolation on Leak Detection (NSRST)
- DL4-RR4c – Vapor Trap Cell Isolation on Overpressure (NSRST)
- DL4-RR7 – Fuel Handling Building Barrier (NSRST)

DL4-RR1 refers to the functional containment boundary formed around the RES by the HAA barrier, reactor GV, NHV, and other systems penetrating the HAA. DL4-RR1a represents the HAA portion of that boundary, including NHV penetrations, while DL4-RR1c refers to isolation of the HAA boundary upon indication of leakage within the HAA.

PSAR section 7.5.1.3 describes that NSRST NHV valves and dampers are powered by the NST NI AC electrical power low voltage system (NLV) and are designed to fail closed upon loss of power to ensure the radionuclide retention function is accomplished. The reliability of the NHV function to isolate the HAA is enhanced by a double isolation arrangement, with one isolation device located inside and one outside the HAA.

DL4-RR3 refers to the SPS cell barriers, with DL4-RR3c allowing for isolation of the NHV lines penetrating a cell upon leak detection. Similarly, DL4-RR4 refers to the SCG cell barriers, with DL4-RR4b allowing for isolation of the NHV lines penetrating a SCG cell upon leak detection, and DL4-RR4c allowing for automatic isolation of the SCG vapor trap cell upon indication of a primary coolant overpressure relief valve opening. These functions reduce the dose consequences from postulated SPS and SCG events.

The NHV is also associated with DL4-RR7 regarding NHV collection and filtration of air from the FHB prior to release through the monitored main plant stack. This function ensures DID adequacy for ESWRs in the PIC.

The staff determined that the preliminary NHV design is consistent with the identified radionuclide retention functions by providing isolation and filtration capabilities commensurate with the safety significance of the function. The staff notes that performance of the NHV functions in the PSAR LBE analyses are based on assumptions about leakage rates across the HAA boundary, NHV operation and isolation, and filtration performance. While the design is consistent with the ability to perform the functions, the staff will perform a more detailed evaluation at the OL stage.

PSAR section 7.5.1.2 also identifies an NSRST PAM function, DL5-PAM1, identified for DID adequacy but not associated with a particular LBE. The NHV supports this function by providing conditioned and filtered air to the MCR or RSC. It also supplies compressed breathable air bottles that can be manually operated to maintain a habitable environment in the RSC if the MCR must be evacuated. PSAR section 7.5.1.2 states partial conformance to RG 1.97, which endorses IEEE 497-2016 for PAM. The staff determined that this function provided by the NHV has a design basis and classification consistent with the safety significance of monitoring plant parameters to ensure maintenance of safe shutdown conditions. The staff will review selected PAM variables and the extent of conformance with RG 1.97 at the OL stage.

7.5.1.1.14 Programmatic Special Treatments

Programmatic special treatments preliminarily identified for application to the NHV are summarized in PSAR section 7.5.1.4 and include the QAP, D-RAP, RIM program, EQ program, testing program, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the NHV contains NSRST SSCs, the staff determined that these programs are appropriate to apply to the NHV. Plant programs are evaluated further in SE chapter 8.

7.5.1.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.5.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.5.2 Nuclear Island Fire Protection System

PSAR section 7.5.2 describes the SSCs that make up the NFP. The following subsystems are part of the NFP:

- Nuclear island fire water distribution system (NFD)

- Nuclear island fire suppression system (NFS)
- Nuclear island fire detection and alarm system (NFA)
- Sodium leak detection, collection, and containment system (NNA)

The NFP provides fire protection for the NI by detecting, suppressing, and mitigating fires and fire-related hazards. The NFD distributes fire water for fire hydrants and water-based fire suppression within NI buildings, facilities, structures, and yard area and that the water supply to the NFD is from the energy island fire protection system (EFP). The NFS suppresses and controls fire with water, clean agents, and chemical means in both automatic systems and manual fire-fighting methods. The NFA detects smoke and fire, monitors and controls suppression systems, and notifies personnel of fire emergency situations. The applicant states that the NFS, NFD, NFS, and NFA are NST. The NNA detects, collects, and contains sodium leaks to mitigate the risk of sodium fires. It includes leak detectors, leak mitigative features, drain lines, and catch pans. The NNA is located in all sodium-handling areas. It provides signals to the AMC and UMC to support automatic responses such as valve isolation and HVAC damper closure. The NNA is NSRST. The NFP supports three NSRST functions, which are evaluated below.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the NFP:

- 10 CFR 50.48 “Fire protection,”;
- RG 1.189, Rev. 5; and
- RG 1.191, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” Rev. 1 (ML20287A199).

PSAR sections 7.5.2 and 5.3 identify the following PDC as applicable to the NFP: 1, 2, 3, 4, 73, and 74.

7.5.2.1 Technical Evaluation

7.5.2.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant structures, systems, and components to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR table 1.4-5 lists many codes and standards related to fire protection and building construction that will be used including those from the American Society of Testing and Materials (ASTM), the IEEE, the National Fire Protection Association (NFPA), and Underwriters Laboratories (UL). The staff determined that the preliminary NFP design is consistent with PDC 1 because the codes and standards specified for fire protection and building construction are acceptable as they are nationally recognized or endorsed in applicable NRC guidance.

The staff notes that NNA SSCs are currently classified as NSRST based on events currently considered in the PRA, which does not include fire hazards. The staff will review the safety

classification of NNA SSCs at the final design stage once a final design, fire analysis, and fire PRA are available to inform NNA SSC safety classification.

7.5.2.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires safety-significant SSCs to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.5.2.2 states that NFP components are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 6.1.3.1 states that sodium-containing SSCs are treated as a special seismic interaction source based on the potential for seismic-induced fire and are required to demonstrate the ability to retain sodium inventory following an SSE. Based on the commitment to design NFP SSCs to withstand natural phenomena and the application of seismic treatment to sodium-containing SSCs to prevent seismic-induced sodium fires, the staff determined that the preliminary design information of the NFP is consistent with PDC 2. Given that fire hazards, as well as the locations and functions of NFP SSCs, are highly dependent on the final design configuration and will be informed by the fire PRA to be provided with the OL application, the staff determined that further information can reasonably be left for later consideration.

7.5.2.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. It also requires fire detection and suppression systems to be provided with sufficient capacity and capability to protect safety-significant SSCs while also not impairing the safety functions of these SSCs in the event of rupture or inadvertent activation.

PSAR table 1.4-1 indicates full conformance with RG 1.189. The PSAR also states that the guidance provided in RG 1.191 is not applicable to the design and construction phase but will be addressed at the OL stage. PSAR table 3.1-3 states that a fire PRA will be provided with the OL application. The staff notes that fire hazards are highly dependent on the final design configuration, and that following the guidance in RG 1.189 and RG 1.191 provides a reasonable approach to address most conventional fire hazards.

Based on conformance with RG 1.189, the applicant's plans to address RG 1.191 at the OL stage, and the commitment to provide a fire PRA with the OL application, the staff determined that the preliminary design information for the NFP is consistent with PDC 3 with respect to conventional fire hazards. Given that fire hazards are highly dependent on the final design configuration and the planned submittal of fire PRA, the staff determined that further information can reasonably be left for later consideration. The staff evaluation of PDC 3 with respect to sodium fire hazards is provided in SE section 8.3.n

7.5.2.1.4 Environmental and Dynamic Effects (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.5.2.2 states that NFP SSCs are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. These systems and components are appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit. The PSAR additionally states that chemical consequences of accidents, such as

sodium leakage, are considered for the design of NNA components. Given the mild environments expected for NFP SSCs in normal operation and the design of NNA SSCs to address sodium leakage, the staff determined that the preliminary NFP design is consistent with PDC 4. The staff evaluation of the KU1 design with respect to sodium fire hazards is provided in SE section 8.3.

7.5.2.1.5 Sodium Leakage Detection and Reaction Prevention and Mitigation (PDC 73)

PDC 73 requires a means to detect sodium leakage and to limit and control sodium-air and sodium-concrete reactions. PSAR section 7.5.2 discusses sodium leakage detection, collection, and fire mitigation and PSAR section 8.2 describes the sodium fire protection design. Based on the staff evaluation of PSAR sections 7.5.2 and 8.2 in SE section 8.3, the staff determined that the preliminary design for sodium leakage detection and reaction prevention and mitigation, which includes the NFP, is consistent with PDC 73.

7.5.2.1.6 Sodium-Water Reaction Prevention (PDC 74)

PDC 74 requires SSCs containing sodium to be designed and located to avoid contact between sodium and water. PSAR section 8.2.1 states that there are no water-containing systems within the RXB and RAB to avoid contact between sodium and water and to limit the adverse effects of chemical reactions between sodium and water or the capability of SSCs to perform safety-significant functions. Fire water suppression systems are not provided in areas with sodium-containing systems in the FHB. Based on separation of water-containing NFP SSCs from areas containing sodium, the staff determined the preliminary NFP design is consistent with PDC 74.

7.5.2.1.7 Safety Functions

The applicant states that PSFs are determined through the plant-specific PRA described in PSAR section 3.1. Safety-significant PSFs are defined in PSAR section 5.2. The NFP supports several safety-significant functions, which are designated as NSRST to indicate the safety classification of NFP systems and components relied upon to perform the function.

7.5.2.1.8 NFP Leak Detection Functions

PSAR section 7.5.2.2 states that the NFP supports three NSRST functions that isolate SSCs in the event of a sodium leak. These are:

- DL4-RR3b SPS Pump Trip on Leak Detection (NSRST)
- DL4-RR3a, SPS Supply Valve Isolation on Leak Detection (NSRST)
- DL4-RR3c, SPS Cell Barrier Isolation on Leak Detection (NSRST)

PSAR section 7.5.2.2 states that DL4-RR3b requires that the SPS-P pumps trip on detection of a sodium leak within the SPS-P. DL4-RR3a requires the SPS-P supply isolation valve to automatically close upon leak detection. Both functions are automatically initiated by the NSRST AMC, which receives leak detection signals from the NSRST NNA. The PSAR further states DL4-RR3c requires automatic isolation of SPS-P cell ventilation upon leak detection. This function is initiated by the NSRST UMC, which receives leak detection signals from the NSRST

NNA. The NNA's leakage detection capability is evaluated in SE section 8.3. These functions, along with their roles in the SPS design, and the designs of the AMC and UMC systems, are evaluated in SE sections 7.2.4 and 7.6.2, respectively. Based on the evaluation of NNA in SE section 8.3, SPS in SE section 7.2.4, and AMC and UMC in SE section 7.6.2, the staff determined that the preliminary design is consistent with these three NSRST functions to isolate in the event of a sodium leak.

7.5.2.1.9 Programmatic Special Treatments

Programmatic special treatments applied to the NFP are summarized in PSAR section 7.5.2.4 and include the QAP, D-RAP, EQ program, IST, PITAP, and testing program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the NFP contains NSRST SSCs, the staff determined that these programs are thus appropriate to apply to NFP. Plant programs are evaluated further in SE chapter 8.

7.5.2.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.5.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.6 Integrated Control System

7.6.1 Instrumentation and Control Systems

PSAR section 7.6.1 describes the I&C systems, which provide automatic actuation in response to LBEs, contribute to satisfying PDC, and provide means for manual actuation of components in support of DID. I&C systems are designed to perform assigned SR, NSRST, and NST functions to assist in satisfying the PDC.

The NRC approved TR, NAT-4950-A, "I&C Architecture and Design Basis Topical Report," Rev. 2 (ML25232A241), is incorporated by reference into the PSAR and describes the overall architecture and associated design basis for the I&C system. The staff's SE for NAT-4950-A imposed the following limitations and conditions (L&Cs):

1. Applicants or licensees referencing this TR will need to address the areas which are not evaluated by the TR with respect to I&C architecture and design basis. As an example, section 4.3 of the TR states that an applicant or licensee referencing the TR "will provide the AST architecture including interface with RTBs." An applicant or licensee referencing the TR must submit documentation and justify that the activities have been completed to a state that is appropriate for the intended licensing application..
2. Because this TR is based on a preliminary design and partial implementation of the LMP, an applicant or licensee referencing this TR must utilize the NEI 18-04 methodology as the I&C system design matures and provide justifications for areas that deviate from the TR.
3. Applicants or licensees referencing this TR outside the context of an LMP-based approach must describe how the TR remains applicable outside of an LMP-based context and, as appropriate, supplement the TR as needed.

4. Because this TR relies on the NRC approved RadICS TR 2016-RPC003-TR-001, NP-A “RadICS Topical Report,” Rev. 2 (ML20202A030), an applicant or licensee referencing this TR must fully address that applicable generic open items and plant-specific action items in the RadICS TR and provide justifications for items that are not applicable.

The staff’s evaluation of the L&C implementation is summarized in the bullets below:

- The PSAR’s use of NAT-4950-A meets TR L&C 1 for a CP application based on the information provided on the I&C systems in PSAR section 7.6 and the staff’s evaluation discussed in this section of the SE.
- The PSAR’s use of NAT-4950-A meets TR L&C 2 because the preliminary Sodium design described in the PSAR is consistent with the design information in the TR and utilizes the NEI 18-04 LMP methodology.
- L&C 3 is not applicable because the preliminary Sodium design described in the PSAR utilizes the LMP methodology.
- USO intends to address the generic open items and plant-specific action items described in the RadICS TR at the OL stage. The staff determined this is acceptable because these generic open items and plant-specific action items can only be addressed using detailed design information not expected to be available at the CP stage. The staff will review how the applicant addresses L&C 4 at the OL stage.

The relationship between the I&C systems and the defense line (DLs) is described in NAT-4950-A, while PSAR table 7.6.1-1 identifies the I&C subsystems, their safety classification, and associated DLs.

The I&C system architecture incorporates the following functional I&C design principles, consistent with the Design Review Guide (DRG), “Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews” (ML21011A140):

- Independence
- Communication and logical independence
- Redundancy
- Diversity
- System integrity
- Reliable I&C
- Human factors engineering

The staff also considered the guidance provided in DNRL-ISG-2022-01, “Safety Review of Light-Water Power Reactor Construction Permit Applications” Rev. 0 (ML22189A099) in scoping its review of the I&C design at the CP stage. Appendix A of DNRL-ISG-2022-01 states that the following items should be evaluated for a CP application review:

- an overall I&C architecture that demonstrates adherence to the fundamental I&C design principles;
- plant safety functions allocated to each of the safety-related I&C systems;
- proposed communications between safety-related and non-safety-related I&C systems;
- regulations that the applicant intends to comply with for the I&C design;
- regulations that the applicant intends to take exemption from or deems not applicable to its design; and
- TRs incorporated by reference in the application.

Due to the preliminary nature of the KU1 I&C design the CP application review scope was limited to the DRG and DNRL-ISG-2022-01 items above. Detailed review of the I&C design will be performed at the OL stage when final design details are available.

The fundamental I&C design principles for the preliminary system architecture are described in NAT-4950-A. Since the I&C systems are classified as SR, NSRST, and NST, the I&C architecture implements the following communication strategy to support the fundamental I&C design principles:

- There are no direct digital data communications from either the NSRST or NST systems to the SR systems.
- One-way communication from SR I&C systems to the NSRST or NST systems is via isolated hardwire or through data-diode and gateways.

The SR I&C systems do not have inter-divisional communications with the exception of the RPS voting.

PSAR section 5.3 identifies PDC 1 as applicable to the I&C systems.

7.6.1.1 Technical Evaluation

7.6.1.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.6.1 states the I&C systems are designed in accordance with IEEE 603-2018 with further details provided in PSAR sections 7.6.2 through 7.6.6 and section 7.6.8. PSAR section 7.6.1.1.3 notes that descriptions of compliance with regulatory requirements and conformance with regulatory guidance and industry codes are provided in NAT-4950-A. The staff notes that 10 CFR 50.55a(h) incorporates by reference IEEE 603-1991, not the newer edition, IEEE 603-2018.

NAT-4950-A states that the Sodium design can adopt IEEE 603-2018 and still demonstrate compliance with IEEE 603-1991. The staff determined that this comparison demonstrating

compliance with IEEE 603-1991 can reasonably be left for later consideration at the OL stage because a final design is needed to demonstrate full compliance and the standards are largely aligned.

7.6.1.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.6.2 Nuclear Island Control System

PSAR section 7.6.2 describes the NIC, a digital system with both NSRST and NST functions. The NIC is a group of I&C subsystems and is primarily implemented on a NI distributed control system (DCS). While most of the NIC is implemented on a DCS, it also includes controls executed outside of the DCS.

PSAR figure 7.6.2-1 is a block diagram of the NIC and its sub-systems, showing their connections and representing a generic interface that shows data is shared to and from the NIC.

The following sub-systems are part of the NIC:

- AMC – monitors and controls NI auxiliary systems
- CTC – monitors and controls various heat transport and other coolant temperature systems
- Fuel handling supervisory control system (FHC) - monitors and controls the sub-systems and sub-components of the fuel handling systems
- Plant monitoring and control system (PMC) – includes plant information processing and display devices, storage devices, and operator manual command devices
- Rod monitoring and control system (RMC) – monitors and controls the sub-systems and sub-components of the control rod drive.
- UMC – monitors and controls the various auxiliary electrical and ancillary systems.

Safety classification and associated NSRST functions are specific to each NIC sub-system and NIC component. The NIC as a whole is not classified.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the NIC:

- RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid, and Gaseous Effluents and Solid Waste,” Rev. 3 (ML21139A224);
- RG 1.75, “Criteria for Independence of Electrical Safety Systems,” Rev. 3 (ML043630448);

- RG 1.89, Rev. 2;
- RG 1.100, Rev. 4;
- RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," Rev. 2 (ML19175A044);
- RG 1.204, "Guidelines for Lightning Protection for Production and Utilization Facilities," Rev. 1 (ML23241A957); and
- DRG.

PSAR sections 7.6.2 and 5.3 identify the following PDC as applicable to the NIC: 1, 2, 10, 13, 15, 24, 25, 26, 28, 29, 60, 61, 63, 64, and 72.

7.6.2.1 Technical Evaluation

7.6.2.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. The NIC performs both NSRST and NST functions. Quality, design, and special treatments are discussed in PSAR section 7.6.2.3.

The NIC design, as described in PSAR section 7.6.2.3, includes elements of quality to minimize failures in its NSRST functions and utilizes design principles supported by industry codes and standards. These principles of qualification, reliability, robustness, security, and DID features are discussed in PSAR sections 7.6.2.3.1 through 7.6.2.3.5.

The NIC meets the quality standards established in the QAPD. PSAR section 7.6.2.3 identifies the seismic classification as SCN3. No part of the NIC is required to operate during or after a seismic event and is designed to ensure it will not fail in a way that would prevent another SSC from performing its safety-significant function. The NIC includes provisions for redundancy to increase reliability for NSRST functions. Reliability and capability targets for SSCs will be determined at the OL stage and are described in PSAR section 6.2.

The RPS sends data to the NIC via a one-way data diode to ensure separation and any hardwire interfaces between the RPS and NIC include isolation devices. Physical and cyber security programs are described in PSAR section 11.6.

For DID, where diversity is required, redundant sensors are provided with one sensor supplying a signal to the SR I&C system and another to the NSRST or NST I&C system. In cases where a single SR sensor is shared between SR and NSRST or NST I&C systems, the sensor is routed through a splitter to provide isolation. For instances where manual control is an NSRST function, the NSRST manual actuation is performed using a hardwired switch.

The NIC also performs data acquisition and display of type D and E PAM variables as defined in IEEE 497-2016, which is endorsed by RG 1.97. PAM for the I&C systems is evaluated in SE section 7.6.1.11.7.

Based on its review described above, the staff determined that the preliminary NIC design is consistent with PDC 1 because the codes and standards specified are acceptable and support the design principles of qualification, reliability, robustness, security, and DID, in alignment with guidance on the fundamental I&C design principles from the DRG.

7.6.2.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 requires that safety-significant SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

PSAR section 7.6.2.2.2 states that the NSRST NIC SSCs are located either in the SR RXB or NCB substructures, or the NSRST NCB superstructure, all of which are designed to withstand the effects of applicable external hazards without loss of structural integrity.

The staff determined that the preliminary NIC design is consistent with PDC 2 because its NSRST SSCs are located in substructures or superstructures that are designed to withstand the effects of applicable external hazards. Final safety classification of the NIC will determine the requirements necessary to ensure that the effects of applicable external hazards on the NIC do not impact the capability of SSCs to perform their safety functions.

7.6.2.1.3 Instrumentation and control (PDC 13)

PDC 13 requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions, to ensure adequate safety. Additionally, appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

PSAR section 7.6.2.2.2 states that the NIC provides the capability to reliably monitor and control the plant systems during normal power operations, startup, and shutdown, including steady state conditions and anticipated transients. The NIC is implemented on a DCS, which provides a means for monitoring and execution of both manual and automatic control of plant systems.

The staff determined that the preliminary NIC design is consistent with PDC 13 because the NIC is designed to control and monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

7.6.2.1.4 Separation of protection and control systems (PDC 24)

PDC 24 requires that the protection system be separated from control systems such that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system.

PSAR section 7.6.2.2.2 states that the RPS is separated from NIC control systems to the extent necessary to ensure that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the RPS. Additionally, there is no direct communication from the NIC to the RPS. One-way communication between the RPS to NSRST and NST control systems is via a data diode or hardware connection.

The staff determined that the preliminary NIC design is consistent with PDC 24 because the described failure scenarios still allow the RPS to satisfy all reliability, redundancy, and independence requirements as well as any communication from the RPS to the NIC is one-way or includes SR isolation hardware.

7.6.2.1.5 Protection system requirements for reactivity control malfunctions (PDC 25)

PDC 25 requires that the protection system be designed to ensure that specified acceptable system radionuclide release design limits are not exceeded during any AOOs, accounting for a single malfunction of the reactivity control systems.

PSAR section 7.6.2.2.2 states that the NIC provides reactivity control during normal operations and AOOs through power runbacks performed by the RMC and CTC.

The staff determined that the preliminary NIC design is consistent with PDC 25 because it ensures radionuclide design limits are not exceeded during AOOs by controlling reactivity during normal operations and AOOs independent of the RPS and by ensuring failure of the NIC does not impact the safety and reactivity control functions of the RPS.

7.6.2.1.6 Reactivity Limits (PDC 28)

PDC 28 requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the safety-significant elements of the primary coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other RV internals to significantly impair the capability to cool the core.

PSAR section 7.6.2.2.2 states that the RMC subsystem of the NIC, along with the CRD, control the maximum rod withdrawal speed to limit the reactivity worth insertion rate.

The staff determined that the preliminary RMC subsystem of the NIC design is consistent with PDC 28, because it includes controls that apply appropriate limits on the potential amount and rate of reactivity increase by controlling maximum rod withdrawal speed.

7.6.2.1.7 Radiation Monitoring (PDC 60, 63, and 64)

PDC 60 requires that systems be designed to control the release of radioactive materials in gaseous and liquid effluents and that waste systems have sufficient holdup capacity for retaining these effluents. PDC 63 requires that appropriate systems be provided to detect conditions in radioactive waste systems that may result in loss of residual heat removal capability or excessive radiation levels, and to initiate appropriate safety actions. PDC 64 specifies the monitoring requirements for the effluent discharge paths for radioactivity that may be released from normal operations.

PSAR section 7.6.2.2.2 states that the NSRST AMC and NST PMC are designed and operated with appropriate controls and setpoint margins to support the control of radioactive material releases to the environment during all normal, AOO, and postulated accident conditions. The RMS and PMC monitor for radiological releases through the exhaust air in the vent stack.

PSAR section 7.6.6.2 states that the RMS provides radiation detection for systems, buildings, and release pathways throughout the plant at strategic interface points. The RMS is capable of supporting NSRST functions during all AOOs as well as during normal, post-accident, and maintenance operations. PSAR section 7.6.6.2 further states that the RMS conforms to RG 1.21 and RG 1.97. However, PSAR table 1.4-1 notes that RG 1.21 is not applicable to the design and construction phase and will be addressed at the OL stage. In addition, the design partially conforms to RG 1.97, with the extent of the conformance to be addressed at the OL stage.

The staff determined that RG 1.21 is appropriate to describe methods for measuring, evaluating, and reporting radioactive material in liquid and gaseous effluents and solid waste. At the OL stage, RG 1.21 is also appropriate for assessing and reporting public dose to demonstrate compliance with applicable 10 CFR Part 20, "Standards for Protection Against Radiation," and 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," dose and concentration limits, as well as the dose objectives in 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." The reference to RG 1.21 provides confidence to the staff that the KU1 site will implement a monitoring program that will follow approved NRC methods. As stated in PSAR section 7.6.2.2, the RMS supports the monitoring of components containing radioactivity to ensure releases remain below the above-mentioned regulatory limits.

The staff determined that the preliminary NIC design is consistent with PDC 60, 63, and 64 because the preliminary information on radiation monitoring needed to support the control of radioactive materials includes high level monitoring locations and the applicant stated it will conform to RG 1.21 at the OL stage.

7.6.2.1.8 Sodium Heating Systems (PDC 72)

PDC 72 requires that heating systems be provided for safety-significant systems and components that contain, or could be required to contain, sodium. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature are maintained within design limits assuming a single failure.

PSAR section 7.6.2.2.2 states that the NIC subsystems (RMC, CTC, and PMC) are designed and operated with appropriate controls and setpoint margins to ensure that the temperature distribution and rate of change of temperature in SSCs containing sodium are maintained within design limits. Additionally, the NIC controllers are redundant to mitigate single system hardware failures.

The staff determined that the preliminary NIC design is consistent with PDC 72 because it provides the capability to control and maintain temperature distribution and rate of change of temperature for SSCs containing sodium within design limits and includes the capability to mitigate a single failure.

7.6.2.1.9 Other supported PDC

PSAR section 7.6.2.2.2 states the NIC functions support meeting the following PDC:

- Reactor design (PDC 10) as discussed in SE section 7.6.3
- Primary coolant system design (PDC 15) as discussed in SE section 7.6.3

- Reactivity control systems (PDC 26) as discussed in SE section 7.6.3
- Protection against AOOs (PDC 29) as discussed in SE section 7.6.3
- Fuel storage and handling and radioactivity control (PDC 61) as discussed in SE section 7.3.3

The staff's primary evaluation of these PDC is performed in other SE sections, as the NIC only provides a support function.

7.6.2.1.10 Safety Functions

7.6.2.1.10.1 AMC Sub-System Function

PSAR section 7.6.2.2.1 states the AMC is a digital NIC sub-system with three DL4 NSRST functions:

- DL4-RR3a – SPS supply valve isolation on leak detection (NSRST)
- DL4-RR3b – SPS pump trip on leak detection (NSRST)
- DL4-RR4a – Automatically close SCG isolation valves on leak detection (NSRST)

DL4-RR3a specifies the AMC receives sodium leak detection signals from the NFP and provides a signal to close the SPS-P pump supply side isolation valve on the detection of a sodium leak. DL4-RR3b specifies that the AMC receives sodium leak detection signals from the NFP and provides a signal to trip the SPS-P pump on the detection of a sodium leak. The final DL4 function is DL4-RR4a, which specifies the AMC receives radioactivity indication signals from the RMS, compares them to a setpoint, and provides a signal to close the SCG RV exhaust isolation valves that are downstream of the vapor cell. PSAR table 5.2-4 identifies these three DL4 functions that are required for DID adequacy with no associated LBEs.

The staff determined that the preliminary AMC design is consistent with DL4-RR3a and DL4-RR3b because they rely on equipment within the AMC sub-system, using signals from the NFP, that are diverse from equipment in the SPS for initiating the SR pump trip, which is the SR means of stopping an SPS-P leak. Additionally, the staff determined that the preliminary AMC design is consistent with DL4-RR4a because it relies on equipment within the AMC sub-system, using signals from the RMS, and operates independently from the SR SCG isolation (DL3-RR10).

7.6.2.1.10.2 CTC Sub-System Function

PSAR section 7.6.2.2.1 states the CTC is a digital NIC sub-system with one DL2 and two DL4 functions:

- DL2-HR2 – ISP trip on low IHT level (NSRST)
- DL4-HR2 – PSP trip automatic backup (NSRST)

- DL4-HR3 – ISP pump trip automatic backup (NSRST)

DL2-HR2 specifies the CTC receives IHT expansion tank level signals from the IHT and provides a signal to trip the ISPs if a low level setpoint is reached. DL4-HR2 and DL4-HR3 specify that the CTC receives primary sodium hot pool temperature signals from the RIS and provides a signal to trip the PSPs (DL4-HR2) and the ISPs (DL4-HR3) if the high-high primary sodium temperature is exceeded. PSAR table 5.2-4 identifies these DL2 and DL4 functions associated with the CTC are required for DID adequacy with no associated LBEs.

The staff determined that the preliminary CTC design is consistent with DL4-HR2 and DL4-HR3 because it relies on equipment within the CTC sub-system, using signals from the RIS, that are diverse from the SR equipment in the IHT for tripping the PSP and ISP, respectively. Additionally, the staff determined that the preliminary CTC design is consistent with DL2-HR2 because it relies on equipment within the CTC sub-system, using IHT expansion tank level signals, that is diverse from the primary equipment in the IHT for tripping to ISP.

7.6.2.1.10.3 UMC Sub-System Function

PSAR section 7.6.2.2.1 states the UMC is a digital NIC sub-system with four DL4 functions:

- DL4-RR1c – HAA HVAC operations following postulated release (NSRST)
- DL4-RR3c – SPS cells barrier isolation leak detection (NSRST)
- DL4-RR4b – SCG cells barrier isolation leak detection (NSRST)
- DL4-RR4c – Vapor trap cell isolation on overpressure (NSRST)

DL4-RR1c specifies the UMC receives radioactivity indication signals from the RMS and sends signals to the NHV to close the HAA isolation valves if a setpoint is exceeded. DL4-RR3c specifies the UMC receives sodium leak detection signals from the NFP and sends signals to close dampers in the NHV to isolate ventilation to the SPS-P cells. DL4-RR4b specifies the UMC receives radioactivity indication signals from the RMS and sends signals to close valves to isolate the SCG vapor trap cells if a setpoint is exceeded. DL4-RR4c specifies the UMC receives position indication signals for the RV relief valves and sends a signal to close valves to isolate the SCG vapor trap cells if a relief valve is open. PSAR table 5.2-4 identifies these four DL4 functions as required for DID adequacy, with only DL4-RR1c associated with an LBE.

The staff determined that the preliminary UMC design is consistent with DL4-RR1c because it relies on equipment within the UMC sub-system, using signals from the RMS, that are diverse from the equipment in the NHV initiating HAA barrier isolation. DL4-RR1c is associated with two DBEs, RRS-CGR-BL and RRS-CGR-1, both of which are mapped to DBA RRS-CGR-CN, SCG leak downstream the SCG cell. This DBA has a 2-hour EAB TEDE dose of 7.75 rem and 30-day LPZ boundary TEDE dose of 7.77 rem, both of which are projected to be well below the accident dose criteria described in 10 CFR 50.34(a)(1)(ii)(D).

Additionally, the staff determined that the preliminary UMC design is consistent with DL4-RR3c because it relies on equipment within the UMC sub-system, using signals from the NFP, that is diverse from equipment in the NHV to isolate ventilation to the SPS-P cells. The staff determined that the preliminary UMC design is consistent with DL4-RR4b because it relies on

equipment within the UMC sub-system, using signals from the RMS, that are diverse from equipment in the NHV. Finally, the staff determined that the preliminary UMC design is consistent with DL4-RR4c because it relies on equipment within the UMC sub-system, using signals that are diverse from the primary coolant overpressure relief valve position indication.

7.6.2.1.11 Programmatic Special Treatments

Programmatic special treatments applied to the NIC are summarized in PSAR section 7.6.2.3 and include the QAP, D-RAP, EQ program, and PITAP. For the NIC DCS and all NIC subsystems, the Human Factors Engineering Program Plan (PSAR section 11.2) is applicable. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the NIC contains NSRST SSCs, the staff determined that these programs are thus appropriate to apply to NIC. Plant programs are evaluated further in SE chapter 8, with the exception of the human factors engineering program plan, which is evaluated in SE chapter 11.

7.6.2.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.6.3 Reactor Protection System

PSAR section 7.6.3 describes the RPS. The RPS accepts input signals from plant instrumentation, applies required logic, and automatically generates output signals to initiate safety functions, including reactor scram and engineered safety feature (ESF) actuations. The portions of the RPS that support these safety functions are arranged into four redundant divisions (A, B, C, and D). Each process variable used by the RPS to perform a safety function is measured by four redundant instruments or groups of instruments, which are also divided into four divisions.

The RPS also provides the capability to manually initiate each reactor scram and ESF actuation function from the MCR, although these manual functions are not credited in the safety analysis. These manual reactor scram and ESF control circuits are diverse from the RPS automatic logic.

Reactor Protection System Architecture, and Equipment Locations

PSAR section 7.6.3.1.1 states that the RPS architecture is described in NAT-4950-A. The SE within NAT-4950-A states that the overall I&C architecture diagram depicts how the architecture is designed to align with the five layers of defense discussed in table 5-2 of NEI 18-04. The preliminary design of the PSAR I&C systems is shown in NAT-4950-A, figure 5-1. This TR also provides additional information on the I&C systems, classification, trip function, and the Sodium I&C defensive model for the KU1 I&C design.

The RadICS platform, as described in the RadICS TR, is used for the RPS. The generic open items and plant-specific action items described in sections 6.0 and 7.0 of the SE for the RadICS TR will be provided at the OL stage.

PSAR section 7.6.3.2, states the RPS equipment is located in four division-RPS vaults to provide independence for the four SR platforms and separation from the NST systems.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the RPS:

- IEEE 603-1991 as incorporated by reference in 10 CFR 50.55a, "Codes and standards";
- RG 1.22, Rev. 0;
- RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," Rev. 1 (ML003740127);
- RG 1.53, "Application of the Single-Failure Criterion to Safety Systems," Rev. 2 (ML033220006);
- RG 1.75, Rev. 3;
- RG 1.97, Rev. 5;
- RG 1.100, Rev. 4;
- RG 1.105, "Setpoints for Safety Related Instrumentation," Rev. 4 (ML20330A329);
- RG 1.118, "Periodic Testing of Electric Power and Protection Systems," Rev. 3 (ML003739468);
- RG 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," Rev. 4 (ML23054A463);
- RG 1.153, Rev. 1;
- RG 1.180, Rev. 2; and
- DRG.

The applicable TRs for the evaluation of the I&C architecture and the RPS are as follows:

- NAT-4950-A
- RadICS TR

PSAR sections 7.6.3 and 5.3 identify the following PDC as applicable to the RPS: 1, 2, 3, 4, 10, 13, 15, 20, 21, 22, 23, 24, 25, 26, 28, and 29.

7.6.3.1 Technical Evaluation

7.6.3.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. The RadICS TR describes the codes, standards, and RGs applicable to the RPS, along with accompanying application-specific requirements that will be evaluated at the OL stage.

The following preliminary industry codes and standards are applied to the overall design of safety-significant, application-specific RPS equipment that is outside the scope of the RadICS platform:

- IEEE 603-2018;
- IEEE 7-4.3.2-2016;
- IEEE 379-2014; and
- IEEE 384-2018.

These industry codes and standards support the design principles of qualification, reliability, robustness, security, diversity, and DID, as discussed in PSAR sections 7.6.3.3.1 through 7.6.3.3.4.

Human factors considerations for the RPS are designed in accordance with IEEE 420-2013 and are reviewed per the human factors engineering program plan discussed in PSAR section 11.2.

Design Principle - Qualified I&C

The following preliminary industry codes, standards, and RGs are applied for environmental, seismic, and electromagnetic capability qualification of safety-significant, application-specific RPS equipment outside the scope of the RadICS platform:

- IEEE/IEC 60780-323-2016;
- IEEE/IEC 60980-344-2020;
- IEEE 420-2013; and
- RG 1.180, Rev. 2.

A list of RPS equipment and associated safety and seismic classifications is provided in PSAR table 7.6.3-1. The evaluation of the seismic classification process is provided in SE section 6.4.1.

Design Principle - Reliable I&C

Reliability and capability targets for SSCs will be determined at the OL stage and are described in PSAR section 6.2. IEEE 338-2022 and IEEE 603-2018 are applied as special treatments to safety-significant RPS equipment to ensure these targets are met.

Repair, identification, capability for test and calibration, and maintenance bypass are addressed by various plant-specific action items from section 7.0 of the SE for the RadICS TR. Additionally, various PDC are related to these features and capabilities.

Predictable and repeatable behavior, such that the RPS will perform in a deterministic manner, is described in the RadICS TR.

Design Principle - Robust I&C

Robustness of the RPS design refers to the degree to which it can perform its SR functions correctly in the presence of invalid inputs or stressful environmental conditions.

Single failure is addressed by applying IEEE 379-2014 to the RPS design and through plant-specific action items from section 7.0 of the SE for the RadICS TR.

Completion of protective action, system integrity, independence, interaction between sense and command features, operating bypass, and redundancy are addressed through either various plant-specific action items from section 7.0 of the SE for the RadICS TR or applicable PDC.

Design Principle – Diversity and Defense-in-depth

Diversity in support of DID to address common-cause failure is described in NAT-4950-A.

Design Principle – Security

A secure development and operational environment for the RPS is established in accordance with RG 1.152 and is further addressed through plant specific action items from section 7.0 of the SE for the RadICS TR. Security for I&C is addressed through the physical security and cyber security programs in PSAR section 11.6.

Summary

Based on its review described above, the staff determined that the preliminary RPS design is consistent with PDC 1 because the codes and standards specified are acceptable and support the design principles of qualification, reliability, robustness, security, and DID, in alignment with guidance on the fundamental I&C design principles from the DRG. In addition, plant-specific action items from the RadICS TR support these design principles.

7.6.3.1.2 Protection against Natural Phenomena (PDC 2)

PDC 2 requires that safety-significant SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.6.3.2.2 states that, in the event of adverse conditions caused by natural phenomena, all safety-significant RPS SSCs maintain the capability to perform their required functions. This includes consideration of earthquakes, external flooding, high winds, and extreme climate. SR portions of the RPS are located in the SR NCB substructure.

The staff determined that the preliminary RPS design is consistent with PDC 2 because its SR equipment is located in NCB substructure, which is designed to withstand the effects of applicable external hazards without loss of capability to perform necessary safety functions.

7.6.3.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.6.3.2.2 states that each SR RPS division is located in its own division-specific RPS vault in the NCB substructure, separated by 3-hour fire rated walls and doors. The staff determined that the preliminary RPS design is consistent with

PDC 3 because the RPS is designed and located to minimize the probability of fire simultaneously affecting multiple RPS divisions.d

7.6.3.1.4 Environmental and dynamic effects design bases (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.6.3.2.2 states that the SR NCB substructure protects the SR portions of the RPS from external hazards caused by transportation incidents, incidents at offsite facilities, and external missiles. It also provides a mild environment for SR portions of the RPS.

The staff determined that the preliminary RPS design is consistent with PDC 4 because the RPS is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating conditions, AOOs, and postulated external hazards.

7.6.3.1.5 Reactor Design (PDC 10)

PDC 10 requires the reactor core and associated systems are designed with appropriate margin to assure SARRDLs are not exceeded during normal operation and AOOs. PSAR section 7.6.3.2.2 states that the RPS DL3 reactor scram and ESF functions are designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded during AOOs that are not mitigated by DL2 functions.

The staff confirmed that this is demonstrated based on the AOO analyses provided in PSAR section 3.6, where no AOOs result in fuel limits being exceeded. For these LBEs, the fuel limits are used as a surrogate for the SARRDLs because maintaining cladding integrity ensures there will not be an increase in circulating radionuclide inventory.

The staff determined that the preliminary RPS design is consistent with PDC 10 because the RPS is designed to ensure appropriate margin so that SARRDLs are not exceeded during any condition of normal operation and AOOs, as demonstrated in the analyses provided in PSAR chapter 3.

7.6.3.1.6 Instrumentation and control (PDC 13)

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. PSAR section 7.6.3.2.2 states that the RPS accepts input signals from plant instrumentation in order to monitor plant variables and automatically generate reactor scram and ESF actuation signals when required to mitigate AOOs and DBEs within prescribed limits. The RPS provides displays to operators in the MCR and RSC during normal and postulated post-accident conditions and is designed to monitor variables of their anticipated ranges for normal operation, AOOs, and accident conditions.

The staff determined that the preliminary RPS design is consistent with PDC 13 because the RPS is designed to control and monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

7.6.3.1.7 Primary coolant system design (PDC 15)

PDC 15 requires the design conditions of the primary coolant boundary to not be exceeded during conditions of normal operation, including AOOs. PSAR section 7.6.3.2.2 states that the

RPS DL3 reactor scram and ESF functions are designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded as a result of AOOs that are not mitigated by DL2 functions.

The staff determined that the preliminary RPS design is consistent with PDC 15 because the RPS is designed with sufficient margin to ensure that the design conditions of the primary coolant boundary are not exceeded during normal operations and AOOs.

7.6.3.1.8 Protection system (PDC 20, 21, 22, 23, 24, and 25)

PDC 20 requires that the protection system be designed to automatically initiate appropriate systems during AOOs to prevent exceeding radionuclide release limits and to activate safety-significant systems in response to accident conditions. PDC 21 requires that the protection system be designed with high reliability and testability, ensuring that no single failure or component removal compromises its safety function, and that it supports independent, in-service testing during reactor operation. PDC 22 requires that the protection system be designed to withstand natural events and operational conditions without losing its function, using techniques like functional or component diversity where practical, or otherwise demonstrating acceptable performance. PDC 23 requires that the protection system be designed to default to a safe or otherwise acceptable state in the event of disconnection, power loss, or adverse environmental conditions. PDC 24 requires that the protection system remain fully functional and independent despite any single failure or shared component with the control system. PDC 25 requires that the protection system be designed to ensure that specified acceptable system radionuclide release design limits are not exceeded during any AOOs, accounting for a single malfunction of the reactivity control systems.

PSAR section 7.6.3.2.2 states that, consistent with PDC 20, plant instrumentation continuously senses plant conditions and provides input signals to the RPS. The RPS continuously monitors those input signals, applies required logic, and automatically initiates DL3 reactor scram and ESF functions in certain conditions.

PSAR section 7.6.3.2.2 states that, consistent with PDC 21, the SR portions of the RPS are arranged into four redundant and independent divisions. It actuates when at least two out of the four divisions independently determine such action is required. The divisional arrangement allows in-service testability and promotes high reliability.

PSAR section 7.6.3.2.2 states that, consistent with PDC 22, the SR portions of the RPS are arranged into four redundant and independent divisions that are physically separated to the degree necessary to ensure the accomplishment of safety functions. These divisions are isolated from the effects of DBEs which they are designed to mitigate and the equipment is located to minimize potential damage from hazards. The equipment is also qualified to substantiate design basis performance requirements.

PSAR section 7.6.3.2.2 states that, consistent with PDC 23, on loss of incoming power to the RPS, the undervoltage release mechanism opens the reactor trip breakers, configured in such a way that the RPS reactor scram circuit is designed as “de-energize to trip.” All RPS ESF functions execute similarly, such that the RPS ESF circuits are also designed as “de-energize to trip.” The SR RPS is located in the mild environment of the SR NCB substructure and is environmentally and seismically qualified to remain operable during anticipated adverse conditions. The RPS modules also perform continuous self-diagnostics and range-checking,

which upon detection of a critical fault or other specific conditions, cause the system to place the appropriate outputs to the safe (tripped) state.

PSAR section 7.6.3.2.2 states that, consistent with PDC 24, the RPS is designed with sufficient separation from control systems such that all reliability, redundancy, and independence requirements of the RPS remain intact, even in the event of a single control system failure, or failure or removal from service of any component common to the RPS or any control system.

PSAR section 7.6.3.2.2 states that, consistent with PDC 25, the RPS DL3 reactor scram and ESF functions are designed to ensure that analytical limits established in the safety analysis are not exceeded during AOOs, even with a single failure of the reactivity control system. Related to PDC 24 and 25, the RPS is designed in accordance with IEEE 603-2018, clause 5.1, 5.6, and 6.3, which are applied in a manner consistent with the codes and standards identified in section 7.6.3.2 of the RadICS TR.

The staff determined that the preliminary RPS design is consistent with PDC 20, 21, 22, 23, 24, and 25 because the RPS is designed:

- to (1) automatically initiate safety functions to ensure radionuclide release design limits are not exceeded during AOOs and (2) to sense accident conditions and initiate SSCs (PDC 20);
- for high reliability and in-service testability such that (1) no single failure results in the loss of protection function and (2) removal from service does not result in loss of minimum redundancy unless acceptable reliability can be otherwise demonstrated (PDC 21);
- to ensure that the effects of natural phenomena, normal operation, AOOs, and accident conditions on redundant channels do not result in loss of protection function (PDC 22);
- to fail into a safe state or otherwise acceptable state under conditions such as disconnection of the system, power loss, or adverse environments (PDC 23);
- to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact the RPS satisfying all reliability, redundancy, and independence requirements (PDC 24); and
- to ensure that SARRDLs are not exceeded during normal operation or AOOs, accounting for a single malfunction of the reactivity control systems (PDC 25).

7.6.3.1.9 Reactivity control systems and limits (PDC 26 and PDC 28)

PDC 26 requires a minimum of two reactivity control systems or means to provide the capability to insert negative reactivity and maintain the reactor shutdown under various conditions. PDC 28 requires limits on the amount and rate of reactivity increase to ensure the primary coolant boundary and core are not damaged.

PSAR section 7.6.3.2.2 states that consistent with PDC 26, the RPS DL3 reactor scram function is designed to promptly insert sufficient negative reactivity, with appropriate margin, to ensure analytical limits established by the safety analysis are not exceeded during AOOs not mitigated by DL2 functions. The RPS DL3 reactor scram function is also designed to insert sufficient negative reactivity, with appropriate margin, to achieve and maintain a safe shutdown condition following a postulated accident. Additionally, the RPS is designed such that when any division generates a DL3 reactor scram demand, that division also generates an independent signal to the CRD to demand a DL4 CRD driveline scram follow. Consistent with PDC 28, the RPS DL3 reactor scram function is designed with appropriate margin to ensure that analytical limits in the safety analysis are not exceeded.

The staff determined that the preliminary RPS design is consistent with PDC 26 because the RPS provides a means of inserting negative reactivity, with appropriate margin, to ensure that radionuclide design limits are not exceed and that safe shutdown is achieved and maintained during normal operation, AOOs, and following a postulated accident. The staff also determined that the preliminary RPS design is consistent with PDC 28 because the RPS DL3 reactor scram function is designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents do not (1) result in damage to the safety-significant primary coolant boundary or (2) sufficiently disturb the core or other associated structures in a way that significantly impairs the capability to cool the core.

7.6.3.1.10 Protection against AOOs (PDC 29)

PDC 29 requires the reactivity control system to be designed with an extremely high probability of accomplishing its safety function during an AOO. PSAR section 7.6.3.2.2 states that the RPS DL3 reactor scram and ESF functions are designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded during AOOs not mitigated by DL2 functions.

The staff determined that the preliminary RPS design is consistent with PDC 29 because the PSAR states that the RPS will be designed with an extremely high probability of accomplishing scram and ESF functions and have appropriate margin to ensure that analytical limits are not exceeded and as such is reasonable to leave for later consideration.

7.6.3.1.11 Safety Functions

PSAR section 7.6.3.2.1 lists safety-significant PSFs, which are designated as SR (DL3 functions) or NSRST (DL4 and DL5), that the RPS supports.

7.6.3.1.11.1 Reactor Scram Functions

Table 7.6.3-1 below lists the Function ID and description that specifies the RPS variable monitored to initiate the SR function DL3-RC1, "Scram – Gravity Driven Absorber Insertion by Latch Release."

Table 7.6.3-1: PSF DL3-RC1 RPS Function Detail

Function ID	Function Description Variable for Reactor Scram Initiation	Supporting System
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DL3-RC1a	High-high neutron flux	XIS
DL3-RC1b	High-high hot pool temperature	RIS
DL3-RC1c	High-high primary sodium level	RIS
DL3-RC1d	High-high power-to-flow ratio	RIS, XIS
DL3-RC1e	High-high cold pool temperature	RIS
DL3-RC1f	High-high positive neutron flux rate	XIS
DL3-RC1g	High-high negative neutron flux rate	XIS
DL3-RC1h	Low-low primary sodium level	RIS
DL3-RC1i	Low power, high neutron flux	XIS
DL3-RC1j	Loss of primary sodium flow	RIS

The RPS supports DL3-RC1 by continuously monitoring the specified variable (i.e., neutron flux for DL3-RC1a) and if the specified condition is present (i.e., monitored variable exceeds setpoint), the RPS sends a signal to initiate the gravity-driven scram. Additionally, the RPS contributes to DL3-RC2, “Reactor scram on loss of power,” by ensuring that the RPS outputs fail to a safe state in the event of a loss of incoming power.

The staff determined that the LBEs supported by DL3-RC1 and DL3-RC2 span an appropriate range of initiating conditions and event sequences to demonstrate the ability of the gravity scram function to protect the reactor. This includes loss of flow transients, loss of heat sink transients, reactivity transients (including those induced by spurious control rod withdrawal and seismic core compaction), and intermediate-to-primary leak scenarios. Transients that rely on the gravity scram do not exceed the 10 CFR 50.34 dose criteria or the F-C target curve. The staff therefore determined that the preliminary RPS design is consistent with DL3-RC1 and RC2 because LBEs relying on the gravity scram functions maintain dose consequences below acceptable limits.

7.6.3.1.11.2 Manual Reactor Scram Function

The RPS supports NSRST function DL4-RC1, “Reactor scram manual,” by providing the capability to manually initiate a reactor scram from the MCR. This manual function is not credited in the safety analysis but is included to provide diversity from the RPS automatic logic. PSAR table 5.2-4 identifies that DL4-RC1 is required for DID adequacy with no associated LBEs. The staff therefore determined that the preliminary RPS design is consistent with DL4-RC1, as it provides adequate DID through manual initiation of automatic RPS functions.

7.6.3.1.11.3 CRD Driveline Scram Follow

The RPS supports the NSRST function DL4-RC3, “CRD driveline scram follow,” by issuing a failsafe control signal to mechanically drive control rods in when an automatic or manual scram occurs. The RPS output signals that initiate a CRD driveline scram are independent of the RTB demand output signals from the RPS. PSAR table 5.2-4 identifies that DL4-RC3 is a risk-significant NSRST function associated with six LBEs. PSAR section 7.2.5.2 describes how the CRD supports this function by providing motor-driven CRA insertion on a driveline scram

follow demand insertion, as well as indication of full CRA insertion. Performance evaluation of DL4-RC3 is provided through the LBEs described in PSAR chapter 3.

Because of the reliability of the gravity scram function, no AOOs or DBEs rely on the motor-driven scram function. Additionally, because DL4-RC3 is NSRST, it is not used to mitigate any DBAs. However, several BDBEs rely on the motor-driven scram function (e.g., PSAR section 3.8.1.2). Though one of the events, DHP-LOOP-3, the loss of offsite power with scram motor drive-in, results in exceeding the fuel performance acceptance criteria, the dose consequences for this event remain well below the 10 CFR 50.34 dose criteria and the F-C target curve.

The staff determined that the preliminary RPS design is consistent with DL4-RC3 because five of the six LBEs maintain fuel integrity with no radiological release and remaining LBE (DHP-LOOP-3) results in radiological release with dose equivalents below required limits.

7.6.3.1.11.4 Automatic Seismic Trip

The RPS supports the NSRST function DL4-RC4, "Automatic seismic trip," by opening the RTBs upon receipt of a seismic trip demand signal from the anticipatory automatic seismic trip system (AST). PSAR table 5.2-4 identifies that DL4-RC4 is required for DID adequacy with no associated LBEs. The staff determined that the preliminary RPS design is consistent with DL4-RC4, as it provides adequate DID by opening the RTBs upon receipt of a seismic trip demand signal that is diverse from the isolated signal from the AST to the shunt trip feature of the RTBs.

7.6.3.1.11.5 Pump Automatic Trip Functions

The RPS supports the following automatic pump trip functions:

- DL3-HR2 – PSP Trip on High-High Sodium Temperature (SR)
- DL3-HR3 – ISP Trip on High-High Primary Sodium Temperature (SR)
- DL3-HR11 – SPS Pump Trip on Low Primary Sodium Level (SR)
- DL3-HR12 – ISP Trip on High-High Primary Sodium Level (SR)

The RPS supports DL3-HR2 and DL3-HR3 by continuously monitoring primary sodium cold pool temperature inputs from the RIS and neutron flux inputs from the XIS. These inputs are compared to setpoints. If a scram signal exists in the RPS, primary sodium cold pool temperature exceeds a setpoint, and neutron flux is below a setpoint, the PSPs (DL3-HR2) and ISPs (DL3-HR3) are tripped.

For all LBEs relying on DL3-HR2 and DL3-HR3, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff therefore determined that the preliminary RPS design is consistent with DL3-HR2 and DL3-HR3 because all LBEs relying on the RPS maintain fuel integrity and do not result in radiological release.

The RPS supports DL3-HR12 by continuously monitoring primary sodium hot pool level inputs from the RIS and neutron flux inputs from the XIS. These inputs are compared to setpoints. If a

scram signal exists in the RPS, primary sodium hot pool level exceeds a setpoint, and neutron flux is below a setpoint, the ISPs are tripped.

DL3-HR12 is associated with one DBE (IPI-IHEL-BL), and two BDBEs (IPI-IHEL-1 and IPI-IHEL-2). For all LBEs relying on DL3-HR12, PSAR chapter 3 states that fuel integrity is maintained, resulting in no radiological release. The staff therefore determined that the preliminary RPS design is consistent with DL3-HR12 because all LBEs relying on the RPS maintain fuel integrity and do not result in radiological release.

The RPS supports DL3-HR11 by continuously monitoring primary sodium hot pool level inputs from the RIS and comparing them to a setpoint. When the setpoint is exceeded, the SPS-P pump is tripped. DL3-HR11 is associated with two DBEs (RRS-SPLA-BL and RRS-SPLX-BL) which are mapped to two DBAs (RRS-SPLA-CN and RRS-SPLX-CN, respectively). Both DBAs have a 2-hour EAB TEDE dose of $8.13\text{E-}02$ rem and a 30-day LPZ boundary TEDE dose of $2.78\text{E-}01$ rem, which are projected to be far below the accident dose criteria in 10 CFR 50.34(a)(1)(ii)(D). The staff therefore determined that the preliminary RPS design is consistent with DL3-HR11 because these LBEs result in radiological releases with dose equivalents below required limits.

7.6.3.1.11.6 Manual Pump Trip Functions

The RPS supports the following manual pump trip functions:

- DL4-HR6 – Manual PSP Trip (NSRST)
- DL4-HR7 – Manual ISP Trip (NSRST)
- DL4-RR8 – Manual SPS Pump Trip on Low Primary Sodium Level (NSRST)

The RPS supports these functions by providing the capability to manually trip the PSPs, ISPs, and SPS-P. PSAR section 7.6.3.2.1 states that these NSRST manual functions are diverse from the automatic safety functions. PSAR table 5.2-4 identifies that these functions are required for DID adequacy with no associated LBEs. The staff determined that the preliminary RPS design is consistent with these functions as they provide adequate DID through manual initiation of automatic RPS functions.

7.6.3.1.11.7 PAM Function

The RPS contributes to DL5-PAM1 by supporting the display of Type B, C, and F PAM variables. These variables are defined by IEEE 497-2016, which is endorsed by RG 1.97. PSAR table 5.2-4 identifies that DL5-PAM1 is required for DID adequacy with no associated LBEs and that the design of PAM capabilities is informed by IEEE 497-2016. PSAR section 7.6.3.2 states that the RPS partially conforms with the guidance in RG 1.97, with the extent of conformance to be provided at the OL stage. The staff determined that the preliminary RPS design is consistent with DL5-PAM1 as it provides adequate DID by enabling the display of Type B, C and F PAM variables diverse from those on the NIC. The staff will review selected PAM variables and the extent of conformance with RG 1.97 at the OL stage.

7.6.3.1.12 Programmatic Special Treatments

Programmatic special treatments applied to the RPS are summarized in PSAR section 7.6.3.2 and include the QAP, D-RAP, EQ program, PITAP, and human factors engineering program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the RPS contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to RPS. Setpoints for RPS are determined using a methodology endorsed by RG 1.105, with a setpoint control program described in section 2.0 of PSAR table 11.5-1, and will be evaluated at the OL stage. Plant programs are evaluated further in SE chapter 8, with the exception of the Human Factors Engineering Program Plan, which is evaluated in SE chapter 11.o

7.6.3.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.6.4 Nuclear Instrumentation System

PSAR section 7.6.4 describes the XIS. The analog XIS provides instrumentation to detect neutron flux leakage from the core during fuel movement, reactor startup, power operations, shutdown, and accident conditions. The XIS uses a high temperature fissions chamber to measure flux as an electrical signal that is converted into source, wide, and power range signals. The XIS amplifies and conditions the instrument signals and provides inputs to the RPS and the NIC.

The XIS is comprised of four redundant divisions, each with a division-specific flux detector and signal conditioning. Each XIS division is separate and independent from other XIS divisions and provides flux signals to its corresponding RPS division in support of SR protective functions. Each XIS division interfaces with the corresponding RPS division only, thus maintaining the division separation. PSAR figure 7.6.4-1 provides an overview of the XIS.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the XIS:

- IEEE 603-1991 as incorporated by reference in 10 CFR 50.55a;
- RG 1.47, Rev. 1;
- RG 1.53, Rev. 2;
- RG 1.75, Rev. 3;
- RG 1.97, Rev. 5;
- RG 1.100, Rev. 4;
- RG 1.118, Rev. 3;
- RG 1.152, Rev. 4;
- RG 1.153, Rev. 1;

- RG 1.180, Rev. 2; and
- DRG.

PSAR sections 7.6.4 and 5.3 identify the following PDC as applicable to the XIS: 1, 2, 3, 4, 13, 15, 20, 21, 22, 23, 24, 25, and 29.

7.6.4.1 Technical Evaluation

7.6.4.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.6.4.3 states that the XIS is designed to perform both SR and NSRST functions. Special treatments are applied to all XIS SR SSCs, and as required by clause 5.12 of IEEE 603-2018, these special treatments are also applied to all NSRST and NST SSCs.

The following codes and standards are applied as special treatments to the XIS:

- IEEE 338-2022;
- IEEE 379-2014;
- IEEE 384-2018;
- IEEE 420-2013;
- IEEE 603-2018; and
- IEEE 7-4.3.2-2016.

These industry codes and standards support the XIS design principles of qualification, reliability, robustness, security, and DID, as discussed in PSAR sections 7.6.4.3.1 through 7.6.4.3.5.

The XIS meets the quality standards established in the QAPD. SR XIS equipment is located in both mild and harsh environments of the SR NCB and RXB substructures, respectively. The SR XIS is environmentally and seismically qualified to remain operable under anticipated adverse conditions in these locations. The XIS is designed for high functional reliability and in-service testability.

Reliability and capability targets for SSCs will be determined at the OL stage and are described in PSAR section 6.2. XIS reliability goals include minimizing out-of-service time for repair and reducing surveillance testing. These goals are designed to meet clauses 5.7 and 5.10 of IEEE 603-2018 and section 6.11 of IEEE 497-2016. The robustness of the XIS reflects the use of design methods as well as adherence to engineering best practices and are designed to meet clauses 5.1, 5.2, 5.5, 5.6, 6.3, and 6.7 of IEEE 603-2018.

Physical and cyber security programs are described in PSAR section 11.6. For DID, the XIS is designed as an analog system that does not contain software and therefore has no credible

common-cause failure, consistent with clause 5.16 of IEEE 603-2018. If embedded digital devices are procured, guidance from SECY-22-0076 will be used.

Based on its review described above, the staff determined that the preliminary XIS design is consistent with PDC 1 because the codes and standards specified are acceptable and support the design principles of qualification, reliability, robustness, security, and DID, in alignment with guidance on the fundamental I&C design principles from the DRG.

7.6.4.1.2 Protection against Natural Phenomena (PDC 2)

PDC 2 requires that safety-significant SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.6.4.2.2 states that, under adverse conditions due to natural phenomena, all XIS SSCs maintain the capability to perform their safety-significant functions. This includes consideration of earthquakes, external flooding, high winds, and extreme climate. The XIS is located within the SR RXB and NCB substructures and is classified as SCS1.

The staff determined that the preliminary XIS design is consistent with PDC 2 because it is classified as SCS1 and located in the SR RXB and NCB substructures that are designed to withstand the effects of applicable external hazards without loss of capability to perform their safety functions.

7.6.4.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.6.4.2.2 states that the XIS equipment is located within the SR RXB and NCB substructures. In the NCB, each XIS division is located in its own division-specific 3-hour fire rated room to minimize fire impacting multiple divisions. In the RXB, XIS is grouped and spaced to minimize likelihood of fire affecting multiple divisions. The staff determined that the preliminary XIS design is consistent with PDC 3 because the XIS is designed and located to minimize the probability and effect of fires and explosions.

7.6.4.1.4 Environmental and dynamic effects design bases (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.6.4.2.2 states that the SR RXB and NCB substructures protect the XIS from external hazards caused by transportation incidents, incidents at offsite facilities, and external missiles. The SR NCB substructure also provides a mild environment for the SR portions of the XIS located there, while the safety-significant portions of the XIS located in the RXB substructure are qualified for harsh environments.,

The staff determined that the preliminary XIS design is consistent with PDC 4 because the XIS is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating conditions, AOOs, and postulated accidents.

7.6.4.1.5 Instrumentation and control (PDC 13)

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. PSAR section 7.6.4.2.2 states that the XIS accepts input signals from instrumentation in order to monitor plant variables and interfaces with the RPS. All instrumentation and XIS signal conditioning associated with reactor scram,

ESF, and PAM functions are designed to monitor associated variables within their anticipated ranges for normal operation, AOOs, and accident conditions.

The staff determined that the preliminary XIS design is consistent with PDC 13 because the XIS is designed to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

7.6.4.1.6 Protection system (PDC 21, 22, 23, and 24)

PDC 21 requires that the protection system be designed with high reliability and testability, ensuring that no single failure or component removal compromises its safety function, and that it supports independent, in-service testing during reactor operation. PDC 22 requires that the protection system be designed to withstand natural events and operational conditions without losing its function, using techniques like functional or component diversity where practical, or otherwise demonstrating acceptable performance. PDC 23 requires that the protection system be designed to default to a safe or otherwise acceptable state in the event of disconnection, power loss, or adverse environmental conditions. PDC 24 requires that the protection system remain fully functional and independent despite any single failure or shared component with the control system.

PSAR section 7.6.4.2.2 states that consistent with PDC 21, the SR portions of the XIS are arranged into four redundant and independent divisions. Each XIS division provides process variable input to the corresponding RPS division using four redundancy sensing instruments. The divisional arrangement of the XIS and RPS along with the two out of four voting design within RPS enables in-service testability of the XIS. The design allows any division to be taken out of service for testing without loss of safety function and is designed to withstand a failure of any division without loss of safety function or loss of minimum required redundancy. The XIS is designed to withstand a single failure of any division even when one division is taken out of service for testing. The capability to periodically test divisions independently while in-service to determine failures and losses of redundancy further promotes high reliability.

Consistent with PDC 22, the SR portions of the XIS are arranged into four redundant and independent divisions. The divisions are independent and physically separated from each other to the degree necessary to ensure accomplishment of their safety functions.

Consistent with PDC 23, loss of incoming power to XIS results in loss of neutron flux signals to the RPS. In response, the RPS places the associated circuit outputs into the safety state. The SR XIS is located in both mild and harsh environments within the SR NCB and RXB substructures, respectively, and is environmentally and seismically qualified to remain operable under anticipated adverse conditions at these locations. The XIS modules also perform continuous self-diagnostics and, upon detection of a critical fault, automatically place all outputs into the safe state.

Consistent with PDC 24, the XIS is designed with sufficient separation from control systems such that all reliability, redundancy, and independence requirements of the XIS remain intact, even when there is a single control system failure, or failure or removal from service of any component common to the XIS or any control system.

The staff determined that the preliminary RPS design is consistent with PDC 21, 22, 23, and 24 because the XIS is designed:

- for high reliability and in-service testability such that (1) no single failure results in the loss of protection function and (2) removal from service does not result in loss of minimum redundancy unless acceptable reliability can be otherwise demonstrated (PDC 21);
- to ensure the effects of natural phenomena, normal operation, AOOs, and accident conditions on redundant channels do not result in loss of protection function (PDC 22);
- to fail into a safe state or into an otherwise acceptable state under conditions such as disconnection of the system, loss of power, or postulated adverse environments (PDC 23); and
- to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact the RPS satisfying all reliability, redundancy, and independence requirements (PDC 24).

7.6.4.1.7 Other supported PDC

PSAR section 7.6.4.2.2 state the XIS functions support meeting the following PDC:

- Primary coolant system design (PDC 15) as discussed in SE section 7.6.3;
- Protection system functions (PDC 20) as discussed in SE sections 7.6.3 and 7.6.5;
- Protection System Requirements for Reactivity Control Malfunctions (PDC 25) as discussed in SE sections 7.6.2 and 7.6.3; and
- Protection against AOOs (PDC 29) as discussed in SE section 7.6.3.

The staff's primary evaluation of these PDC is performed in other SE sections, as the XIS only provides a support function.

7.6.4.1.8 Safety Functions

7.6.4.1.8.1 Automatic Safety Functions

PSAR section 7.6.4.2.1 lists the following SR PSFs that the XIS supports:

- DL3-RC1a – Reactor Scram on High-High Neutron Flux (SR)
- DL3-RC1d – Reactor Scram on High-High Power-to-Flow Ratio (SR)
- DL3-RC1f – Reactor Scram on Positive Neutron Flux Rate (SR)
- DL3-RC1g – Reactor Scram on Negative Neutron Flux Rate (SR)
- DL3-RC1i – Reactor Scram on Low Power, High Primary Sodium Temperature (SR)

- DL3-HR2 – PSP Trip on High-High Primary Sodium Temperature (SR)
- DL3-HR3 – ISP Trip on High-High Primary Sodium Temperature (SR)
- DL3-HR12 – ISP Trip on High-High Primary Sodium Level (SR)

For all DL3 functions, the XIS supports the RPS by monitoring neutron flux and sending signals to the RPS to allow initiation when specific logic and conditions are met. The staff determined that the preliminary XIS design is consistent with these functions because the XIS supports the RPS and NIC in performing these specified functions by monitoring neutron flux.

7.6.4.1.8.2 PAM Function

For the NSRST function DL5-PAM1, “Post accident monitoring,” the XIS includes sensors that support both the RPS and NIC to provide the display of Type B and D PAM variables. These variables are defined by IEEE 497-2016, which is endorsed by RG 1.97. The staff’s primary evaluation of this function for the I&C systems is performed in SE section 7.6.3, as the XIS only provides a support function.

7.6.4.1.9 Programmatic Special Treatments

Programmatic special treatments applied to the XIS are summarized in PSAR section 7.6.4.3 and include the QAP, D-RAP, EQ program, PITAP, and human factors engineering program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the XIS contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to XIS. Plant programs are evaluated further in SE chapter 8, with the exception of the Human Factors Engineering Program Plan, which is evaluated in chapter 11.

7.6.4.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.4 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.6.5 Reactor Instrumentation System

PSAR section 7.6.5 describes the RIS. The RIS contains instrumentation that measures selected parameters in or near the RV and transmits signals to the RPS and NIC for the use in protective, control, accident monitoring, and surveillance functions. The RIS is a collection of individual instrument channels, each of which may support one or more functions classified as SR, NSRST, or NST.

The RIS has four redundant and independent divisions for providing SR inputs to the RPS. Signal splitters are used to duplicate these SR inputs and provide them to the NIC as NSRST and NST signals. A block diagram of the RIS is shown in PSAR figure 7.6.5-1. A summary of variables monitored by the RIS, the number of instruments, and the safety classification is shown in PSAR table 7.6.5-1.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the RIS:

- IEEE 603-1991 as incorporated by reference in 10 CFR 50.55a
- RG 1.22, Rev. 0;
- RG 1.53, Rev. 2;
- RG 1.75, Rev. 3;
- RG 1.97, Rev. 5;
- RG 1.100, Rev. 4;
- RG 1.118, Rev. 3;
- RG 1.153, Rev. 1;
- RG 1.180, Rev. 2; and
- DRG.

PSAR sections 7.6.5 and 5.3 identify the following PDC as applicable to the RIS: 1, 2, 3, 4, 13, 15, 20, 21, 22, 24, and 29.

7.6.5.1 Technical Evaluation

7.6.5.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.6.5.2 states that the RIS is designed to perform SR, NSRST, and NST functions and to support applicable PDC. The following codes and standards are applied as special treatments to the RIS:

- IEEE 603-2018;
- IEEE 379-2014; and
- IEEE 384-2018.

These industry codes and standards support the RIS design principles of qualification, reliability, robustness, security, and DID.

Design Principle – Qualified I&C

The following preliminary industry codes, standards and RGs are applied for environmental, seismic, and electromagnetic capability qualification to the RIS:

- IEEE/IEC 60780-323-2016;
- IEEE/IEC 60980-344-2020;
- IEEE 420-2013; and
- RG 1.180, Rev. 2

Design Principle - Reliable I&C

Reliability and capability targets for SSCs will be determined at the OL stage, as described in PSAR section 6.2. IEEE 338-2022, IEEE 384-2018, IEEE 420-2013, and IEEE 497-2016 are applied as special treatments to aspects of the RIS to support the reliability of safety-significant equipment in the RPS and to ensure reliability and capability targets are met. Reliability of the RIS is also supported by repair, capability for test calibration, and maintenance bypass features.

SR channels of the RIS are designed so that their behavior is predictable and repeatable and will be provided in the OL stage.

Design Principle - Robust I&C

Robustness of the RIS design refers to the degree to which it can perform its SR functions correctly in the presence of invalid inputs or stressful environmental conditions.

Single failure is addressed by applying IEEE 379-2014 for the design of the RIS. Independence is addressed by applying IEEE 384-2018 for the design of the RIS.

Completion of protective action, system integrity, information displays, automatic control, interaction between sense and command features, operating bypass, and redundancy features support the robust design of the RIS and related I&C systems.

Design Principle – Diversity and Defense-in-depth

Diversity in support of DID to address common-cause failure is described in NAT-4950-A.

Design Principle – Security

Security for I&C is addressed through physical security and cyber security programs in PSAR section 11.6.

Summary

Based on its review described above, the staff determined that the preliminary RIS design is consistent with PDC 1 because the codes and standards specified are acceptable and support the design principles of qualification, reliability, robustness, security, and DID, in alignment with guidance on the fundamental I&C design principles from the DRG.

7.6.5.1.2 Protection against Natural Phenomena (PDC 2)

PDC 2 requires that safety-significant SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.6.5.2.2 states that, under adverse conditions caused by natural phenomena, all safety-significant RIS SSCs maintain the capability to perform their safety functions. This includes consideration of earthquakes, external flooding, high winds, and extreme climate. The RIS is located in the SR RXB and NCB substructures. The seismic classification of RIS components is SCS1. The evaluation of the seismic classification process is provided in SE section 6.4.1.

The staff determined that the preliminary RIS design is consistent with PDC 2 because the equipment is located in RXB and NCB substructures, which are designed to withstand the effects of applicable external hazards without loss of capability to perform necessary safety functions.

7.6.5.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.6.5.2.2 states that the RIS equipment is located within the SR RXB and NCB substructures. In the NCB, each RIS division is located in its own division-specific 3-hour fire rated room to minimize fire impacting multiple divisions. In the RXB, RIS is grouped and spaced to minimize the likelihood of fire affecting multiple divisions. The staff determined that the preliminary RIS design is consistent with PDC 3 because the RIS is designed and located to minimize the probability and effect of fires and explosions.

7.6.5.1.4 Environmental and dynamic effects design bases (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.6.5.2.2 states that the SR RXB and NCB substructures protects the RIS from external hazards caused by transportation incidents, incidents at offsite facilities, and external missiles. The SR NCB substructure also provides a mild environment for SR portions of the RIS located there, while the SR portions of the RIS located in the RXB substructure are qualified for harsh environments.

The staff determined that the preliminary RIS design is consistent with PDC 4 because it is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating conditions, AOOs, and postulated accidents.

7.6.5.1.5 Instrumentation and control (PDC 13)

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. PSAR section 7.6.5.2.2 states that the RIS monitors variables in and around the RV and transmits corresponding signals to the RPS and NIC. These signals are used in protective, control, accident monitoring, and surveillance functions to maintain variables and systems within prescribed operating ranges during normal operation, AOOs, and postulated accidents.

The staff determined that the preliminary RIS design is consistent with PDC 13 because it is designed to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

7.6.5.1.6 Protection system (PDC 20, 21, 22, and 24)

PDC 20 requires that the protection system be designed to automatically initiate appropriate systems during AOOs to prevent exceeding radionuclide release limits and to activate safety-significant systems in response to accident conditions. PDC 21 requires that the protection system be designed with high reliability and testability, ensuring that no single failure or component removal compromises its safety function, and that it supports independent in-service testing during reactor operation. PDC 22 requires that the protection system be designed to withstand natural events and operational conditions without losing its function, using techniques like functional or component diversity where practical, or otherwise demonstrating acceptable performance. PDC 24 requires that the protection system remain fully functional and independent despite any single failure or shared component with the control system.

PSAR section 7.6.5.2.2 states that, consistent with PDC 20, the RIS continuously monitors parameters necessary to detect accident conditions and provides signals to the RPS for the performance of protection functions.

PSAR section 7.6.5.2.2 states that, consistent with PDC 21, the RIS is designed with four redundant and independent divisions to ensure high functional reliability and in-service testability. The design ensures a single failure does not result in a loss of protection functions and that removal from service of any component or division does not result in the loss of required minimum redundancy.

PSAR section 7.6.5.2.2 states that, consistent with PDC 22, the RIS is designed to ensure that the effects of natural phenomena as well as the effects of normal operating, maintenance, testing, and postulated accident conditions on redundant divisions do not result in the loss of a safety function. The SR portions of the RIS are arranged into four redundant and independent divisions. The divisions are independent and physically separated from other divisions to the degree necessary to ensure accomplishment of their safety functions.

PSAR section 7.6.5.2.2 states that, consistent with PDC 24, the RIS inputs to the NIC are electronically isolated and independent of the inputs to the RPS, which ensure that control system failures or removal of control system components do not impact RPS functionality.

The staff determined that the preliminary NIC design is consistent with PDC 20, 21, 22, and 24 because the RIS is designed:

- to (1) automatically initiate to ensure radionuclide release design limits are not exceeded during AOOs and (2) sense accident conditions and initiate SSCs (PDC 20);
- for high reliability and in-service testability such that (1) no single failure results in the loss of protection function and (2) removal from service does not result in loss of minimum redundancy unless acceptable reliability can be otherwise demonstrated (PDC 21);
- to ensure the effects of natural phenomena, normal operation, AOOs, and accident conditions on redundant channels do not result in loss of protection function (PDC 22); and

- to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact the RPS satisfying all reliability, redundancy, and independence requirements (PDC 24).

7.6.5.1.7 Other supported PDC

PSAR section 7.6.5.2.2 states the RIS functions support meeting the following PDC:

- Primary coolant system design (PDC 15) as discussed in SE section 7.6.3
- Protection against AOOs (PDC 29) as discussed in SE section 7.6.3

The staff's primary evaluation of these PDC is performed in other SE sections, as the RIS only provides a support function.

7.6.5.1.8 Safety Functions

7.6.5.1.8.1 Reactor Scram and Pump Trip Functions

PSAR section 7.6.5.2.1 lists the following PSFs that the RIS supports:

- DL3-RC1b – Reactor Scram on High-High Hot Pool Temperature (SR)
- DL3-RC1c – Reactor Scram on High-High Primary Sodium Level (SR)
- DL3-RC1d – Reactor Scram on High-High Power-to-Flow Ratio (SR)
- DL3-RC1e – Reactor Scram on High-High Cold Pool Temperature (SR)
- DL3-RC1h – Reactor Scram on Low-Low Primary Sodium Level (SR)
- DL3-RC1j – Reactor Scram on Loss of Primary Sodium Flow (SR)
- DL3-HR2 – PSP Trip on High-High Sodium Temperature (SR)
- DL3-HR3 – ISP Trip on High-High Primary Sodium Temperature (SR)
- DL3-HR11 – SPS Pump Trip on Low Primary Sodium Level (SR)
- DL3-HR12 – ISP Trip on High-High Primary Sodium Level (SR)
- DL4-HR2 – PSP Trip Automatic Backup (NSRST)
- DL4-HR3 – ISP Trip Automatic Backup (NSRST)
- DL4-RR8 – Manual SPS Pump Trip on Low Primary Sodium Level (NSRST)

For all DL3 functions and DL4-RR8, the RIS supports the RPS by including sensors that monitor the specified variables and sends signals to the RPS to enable initiation when the specific logic and conditions are met. For DL4-HR2 and DL4-HR3, the RIS supports the NIC by including sensors that monitor the specified variables and sends signals to the NIC to initiate when specific logic and conditions are met.

The staff determined that the preliminary RIS design is consistent with these functions because the RIS supports both the RPS and NIC in performing these functions by including appropriate sensors to monitor the specified variables.

7.6.5.1.8.2 PAM Function

For the NSRST function DL5-PAM1, "Post accident monitoring," the RIS includes sensors that support RPS for display of Type B, C, and F PAM variables. These variables are defined by IEEE 497-2016, which is endorsed by RG 1.97. The staff's primary evaluation of this function for the I&C systems is performed in SE section 7.6.3, as the RIS only provides a support function.

7.6.5.1.9 Programmatic Special Treatments

Programmatic special treatments applied to the RIS are summarized in PSAR section 7.6.5.2 and include the QAP, D-RAP, EQ program, PITAP, and human factors engineering program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the RIS contains SR and NSRST SSCs, the staff determined that these programs are thus appropriate to apply to RIS. Plant programs are evaluated further in SE chapter 8, with the exception of the Human Factors Engineering Program Plan, which is evaluated in chapter 11.

7.6.5.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.5 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a CP.

7.6.6 Radiation Monitoring System

PSAR section 7.6.6 describes the RMS, a digital system that performs NSRST and NST functions by providing radiation detection for systems, buildings, and release pathways at strategic interface points throughout the plant.

Each RMS monitor sends activity, monitor health, control, and status information to the digital radiation monitor system (DRMS). DRMS is a data acquisition computer system that provides a direct digital connection to the AMC, which serves as the interface between the operator in the MCR and the DRMS.

The RMS consists of three main monitoring groups: area radiation monitoring (ARM), effluent radiation monitor (ERM), and process radiation monitoring (PRM). PSAR figure 7.6.6-1 provides a high-level overview of the RMS interfaces.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following regulations and guidance in its evaluation of the RMS:

- 10 CFR Part 20;
- IEEE 603-1991 as incorporated by reference in 10 CFR 50.55a;
- RG 1.21, Rev. 3;
- RG 1.97, Rev. 5;
- RG 1.100, Rev. 4; and
- RG 1.180, Rev. 2.

PSAR sections 7.6.6 and 5.3 identify the following PDC as applicable to the RMS: 1, 2, 13, 60, 63, and 64.

7.6.6.1 Technical Evaluation

7.6.6.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. The RMS is designed to perform NSRST functions. PSAR section 7.6.6.3 identifies the following system performance goals for the RMS:

- Selected instrument ranges envelop the full spectrum of minimum and maximum conditions during normal operations, AOOs, postulated accidents, and post-accident conditions.
- Selected setpoints provide indication when monitored radiation levels exceed normal limits.
- No automatic control actions are provided by RMS.
- Response times meet plant requirements.

The following codes and standards are applied as special treatments to the RMS:

- IEEE 338-2022;
- IEEE 603-2018;
- IEEE 379-2014; and
- IEEE 497-2016.

These industry codes and standards support the RMS design principles of qualification, reliability, robustness, security, and DID, as discussed in PSAR sections 7.6.6.3.1 through 7.6.6.3.5.

Design Principle – Qualified I&C

The RMS meets quality standards established by the QAPD. RMS components are designed to the environmental requirements for continuous operation within the normal and abnormal minimum and maximum ambient conditions specific to their installed locations. Instrumentation and equipment used for NSRST functions meet the seismic and environmental requirements of IEEE 497-2016.

The seismic classification of RMS components is SCN1. The evaluation of the seismic classification process is provided in SE section 6.4.1.

Design Principle - Reliable I&C

Reliability and capability targets for SSCs will be determined at the OL stage and are described in PSAR section 6.2. Reliability of the RMS is also supported by redundant monitoring points, reliable power by the NI uninterruptible AC power supply system (NUP), self-test features, and quality design in accordance with the QAPD. Radiation monitors credited with an NSRST function are included in the surveillance testing and periodic calibration programs.

Design Principle - Robust I&C

The robustness of the RMS design reflects the use of design methods and adherence to engineering best practices to ensure functions are achieved for all operational states and accident conditions. The design provides that any single component failure does not prevent the RMS from performing its assigned NSRST function. Redundant monitors are provided as part of the design when supporting NSRST functions. Additionally, the RMS is powered by the NUP, which includes a 72-hour battery backup capability.

Design Principle – Diversity and Defense-in-depth

Post accident monitoring provided by the RMS is required for sufficient DID to provide information following a postulated release.

Design Principle – Security

Security for I&C is addressed through physical security and cyber security programs in PSAR section 11.6.

Summary

Based on its review described above, the staff determined that the preliminary RMS design is consistent with PDC 1 because the codes and standards specified are acceptable and support the design principles of qualification, reliability, robustness, security, and DID, in alignment with guidance on the fundamental I&C design principles from the DRG.

7.6.6.1.2 Protection against Natural Phenomena (PDC 2)

PDC 2 requires that safety-significant SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.6.6.2.2 states that the NSRST RMS components are located within structures designed to withstand the effects of applicable external hazards without the loss of structural integrity. This ensures that

the NSRST RMS functions can be performed in the event of an earthquake, external flooding, high winds, or extreme climate conditions.

The staff determined that the preliminary RMS design is consistent with PDC 2 because its equipment is designed and located to withstand the effects of applicable external hazards without loss of capability to perform its NSRST functions.

7.6.6.1.3 Instrumentation and control (PDC 13)

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. PSAR section 7.6.6.2.2 states that the RMS monitors plant area, effluent, and process radiation variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. Controls are provided by RMS and NIC to maintain these variables and systems within prescribed operating ranges.

The staff determined that the preliminary RMS design is consistent with PDC 13 because the RMS is designed to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

7.6.6.1.4 Radiation Monitoring (PDC 60, 63, and 64)

PDC 60 requires that systems be designed to control the release of radioactive materials in gaseous and liquid effluents. In addition, PDC 60 requires that waste systems have sufficient holdup capacity for retention of gaseous and liquid effluents containing radioactive materials. PDC 63 requires that appropriate systems be provided for radioactive waste systems to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions. PDC 64 specifies the monitoring requirements for the effluent discharge paths for radioactivity that may be released from normal operations.

PSAR section 7.6.6.2.2 states that the RWG, RWL, and RWS SSCs suitably control the release of radioactive materials in gaseous and liquid effluents, handle radioactive solid wastes produced during normal reactor operation, including AOOs, and provide sufficient holdup capacity for retention of gaseous and liquid effluent containing radioactive materials. The RMS supports these functions by monitoring radiation levels and indicating whether waste processing has sufficient radiological margin to allow for waste release (gaseous or solid) or whether further processing or re-use (liquid) is required.

In addition, PSAR section 7.6.6.2.2 states that the RMS monitors all NI building atmospheres with the potential for radioactive release, spaces containing components for primary system sodium and cover gas cleanup and processing, effluent discharge paths, and plant environments via the NHV. The RMS monitors for radioactivity that may be released during normal operations, AOOs, and postulated accidents.

PSAR section 5.3.5.4 states that the RMS monitors normal operational radiation levels and potential off-normal conditions, including area radiation monitoring, process radiation monitoring, and effluent radiation monitoring.

PSAR section 7.6.6.1.2 provides the general monitor locations for the RMS and states that each monitor detection system is located close to its detector location to minimize signal losses and provide local indication and control.

PSAR section 7.6.6.2 states that the RMS provides radiation detection for systems, buildings, and release pathways at strategic interface points throughout the plant. The RMS is capable of supporting its NSRST functions during normal operation, AOOs, post-accident, and maintenance plant conditions. PSAR section 7.6.6.2 also states that the RMS conforms to RG 1.21 and RG 1.97. PSAR table 1.4-1 states that RG 1.21 is not applicable to the design and construction phase of the project and will be addressed at the OL stage. The plant is stated to partially conform to RG 1.97, with the extent of the conformance to be addressed at the OL stage.

PSAR section 7.6.6.3 states that the system performance of each RMS monitor has ranges that envelope the complete minimum and maximum conditions over normal operations, AOOs, postulated accidents, and post-accident conditions, as applicable; and have setpoints that indicate when monitored radiation levels exceed nominal limits. This section also states that the RMS has no automatic actions and the setpoints only provide indication in the MCR and trigger local or remote alarms.

The staff determined that referencing RG 1.21 is appropriate for describing methods to measure, evaluate, and report radioactive material in liquid and gaseous effluents and solid waste. At the OL stage, RG 1.21 is also appropriate for assessing and reporting public dose to demonstrate compliance with 10 CFR Part 20 and 40 CFR Part 190 dose and concentration limits, as well as meeting the dose objectives of 10 CFR Part 50, Appendix I. Reference to RG 1.21 indicates that USO intends to implement a monitoring program that will follow approved NRC methods. As stated in PSAR section 7.6.2.2, the RMS supports the monitoring of components containing radioactivity to ensure releases remain below the above-mentioned regulatory limits. The staff determined that the information contained during the CP application ensures that monitoring is appropriately considered to track radioactive materials, with additional details to be provided during the OL stage.

The staff determined that the preliminary RMS design is consistent with PDC 60, 63, and 64 because the preliminary information on radiation monitoring needed to support the control of radioactive materials includes high level monitoring locations and the applicant stated it will conform to RG 1.21 at the OL stage.

7.6.6.1.5 Safety Functions

PSAR section 7.6.6.2.1 lists the following functions that the RMS supports:

- DL4-RR1c – HAA HVAC Operations following Postulated Release (NSRST)
- DL4-RR4a – Automatically Close SCG Isolation Valves on Leak Detection (NSRST)
- DL4-RR4b – SCG Cells Barrier Isolation on Leak Detection (NSRST)
- DL5-PAM1 – Post-Accident Monitoring (NSRST)

For all DL4 functions, the RMS supports the NIC by detecting radioactivity and sending signals to an NIC subsystem (UMC or AMC) to allow initiation when specific logic and conditions are met. For DL5-PAM1 function, the RMS supports the RPS by detecting radioactivity and sending signals to the RPS for display of Type C and F PAM variables. The primary system (RPS or NIC) is responsible for accomplishing this function, not the RMS. The staff's primary evaluation

of this function for the I&C systems is performed in SE section 7.6.3, as the RMS only provides a support function.

The staff determined that the preliminary RMS design is consistent with these functions because the RMS supports the RPS and NIC in performing these specified functions by detecting radioactivity.

7.6.6.1.6 Programmatic Special Treatments

Programmatic special treatments applied to the RMS are summarized in PSAR section 7.6.6.3 and include the QAP, D-RAP, EQ program, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the RMS contains NSRST SSCs, the staff determined that these programs are thus appropriate to apply to RMS. Plant programs are evaluated further in SE chapter 8.

7.6.6.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.6 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.6.7 Control Room and Indications

PSAR section 7.6.7 describes the MCR and its indications, which are located within the NSRST NI control building superstructure. A conceptual layout of the MCR is shown in PSAR figure 7.6.7-1. The MCR and indications are provided to satisfy PDC 19 and the requirements of the emergency plan. The MCR contains equipment necessary to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Additionally, PSAR section 7.6.7.1 states that no SR operator actions are required from the MCR.

In the event that the MCR cannot be inhabited, operators are evacuated to the RSC, also located in the NSRST NI control building superstructure. The MCR and RSC are located in close proximity to support prompt staffing of the RSC in the event of an evacuation of the MCR.

In addition to the regulations and guidance provided at the beginning of this chapter, 10 CFR 50.34(f)(2)(iii) is applicable to the evaluation of the MCR and indications.

PSAR sections 7.6.7 and 5.3 identify the following PDC as applicable to the MCR: 1 and 19.

7.6.7.1 Technical Evaluation

7.6.7.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PSAR section 7.6.7.1 states that the MCR contains equipment that allows operators to initiate or take manual controls of functions associated with the RPS and NIC via human system interfaces. Applicable codes, standards, and special treatments apply to the RPS and NIC, as described in PSAR sections 7.6.3.3 and 7.6.2.3, respectively. The Human Factors Engineering

Program Plan, as described in PSAR section 11.2, is identified as a preliminary programmatic special treatment applicable to the MCR and indications.

The staff determined that the preliminary MCR design is consistent with PDC 1 based on the staff's review of PSAR section 11.2 as well as the staff's evaluation of the codes and standards applied to the RPS and NIC (SE sections 7.6.3.1.1 and 7.6.2.1.1).

7.6.7.1.2 Control Room (PDC 19)

PDC 19 requires that a control room be provided to ensure safe operation of the nuclear unit during normal and accident conditions, with adequate habitability and radiation protection for personnel. It also requires equipment outside the control room capable of promptly shutting down the reactor and maintaining it in a safe condition.

PSAR section 7.6.7.1 states that the MCR contains the equipment necessary to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. NI HVAC systems control the environmental conditions during normal operations and take measures to ensure MCR habitability under accident conditions. PSAR section 7.6.7.2 states that the RSC contains equipment that provides the indications and controls necessary for prompt shutdown and for maintaining the unit in a safe condition during shutdown, with capability for subsequent safe shutdown.

The staff determined that the preliminary MCR and indications design is consistent with PDC 19 because the MCR is designed to serve as a location from which the nuclear power unit can be operated safely under normal conditions and maintained in a safe condition under accident conditions. Adequate habitability measures are provided to permit access and occupancy during normal operations and under accident conditions. Equipment at locations outside the MCR is provided to: (1) enable prompt shutdown of the reactor with ability to control and maintain the unit, and (2) with a potential capability for subsequent safe shutdown.

7.6.7.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.7 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.6.8 Anticipatory Automatic Seismic Trip System

PSAR section 7.6.8 describes the AST. The AST measures ground motion, applies required signal processing and logic, and automatically generates output actuation signals to the RTBs through the interface device provided by the RPS. The AST is used during seismic BDBEs greater than the SSE to scram the reactor before the peak seismic wave reaches the facility.

The AST signals and interfaces are shown in PSAR figure 7.6.8-1. The AST consists of four channels placed in four different quadrants equidistantly spaced around the RXB. The RTB arrangement constitutes 2-out-of-4 logic for the reactor scram function.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the AST:

- RG 1.53, Rev. 2;

- RG 1.100, Rev. 4;
- RG 1.118, Rev. 3; and
- RG 1.180, Rev. 2.

PSAR sections 7.6.8 and 5.3 identify the following PDC as applicable to the AST: 1, 2, 3, 4, 13, and 24.

7.6.8.1 Technical Evaluation

7.6.8.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. The AST is designed to perform an NSRST function. Special treatments are applied to the AST to meet quality, design, reliability, and performance requirements, described in PSAR section 7.6.8.2.

The following codes and standards are applied as special treatments to the AST:

- IEEE 379-2014; and
- IEEE 384-2018.

These industry codes and standards support the AST design principles of qualification, reliability, robustness, security, diversity and DID, as discussed in PSAR sections 7.6.8.3.1 through 7.6.8.3.4.

Design Principle – Qualified I&C

The AST is qualified per the EQ program. The AST seismic classification is SCN1. The evaluation of the seismic classification process is provided in SE section 6.4.1.

Design Principle - Reliable I&C

Reliability and capability targets for SSCs will be determined at the OL stage and are described in PSAR section 6.2. Reliability of the AST is supported by identification features consistent with IEEE 384-2018 to distinctly identify redundant portions of the system. Reliability is also supported by the D-RAP. Capability for testing and calibration of the AST is consistent with IEEE 338-2022.

Design Principle - Robust I&C

Robustness of the AST design reflects the degree to which it can perform its NSRST function in the presence of invalid inputs or stressful environmental conditions. The AST design is consistent with IEEE 379-2014 to address single failure criteria. IEEE 384-2018 is applied as a special treatment to the AST to meet independence requirements.

Design Principle – Security

Security for I&C is addressed through physical security and cyber security programs in PSAR section 11.6.

Summary

Based on its review described above, the staff determined that the preliminary AST design is consistent with PDC 1 because the codes and standards specified are acceptable and support the design principles in alignment with guidance on the fundamental I&C design principles from the DRG.

7.6.8.1.2 Design bases for protection against natural phenomena (PDC 2)

PDC 2 requires that safety-significant SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.6.8.2.2 states that the AST is seismically qualified for postulated earthquakes and SSCs are protected from external flooding, high winds, and extreme climate. The staff determined that the preliminary AST design is consistent with PDC 2 because the equipment is located in the RXB and is designed to withstand the effects of applicable external hazards without loss of capability to perform their safety function. a

7.6.8.1.3 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PDC 3 specifies that safety-significant SSCs shall be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.6.8.2.2 states that the AST is protected from fire hazards based upon the location of its SSCs. The staff determined that the preliminary AST design is consistent with PDC 3 because the AST is designed and located to minimize the probability and effect of fires and explosions.

7.6.8.1.4 Environmental and dynamic effects design bases (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.6.8.2.2 states that the AST processing cabinets are located in and protected from external hazards by the NCB. The NCB is designed in accordance with codes and standards and specific load combinations to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, AOOs, and postulated accidents without the loss of structural integrity.

The staff determined that the preliminary AST design is consistent with PDC 4 because the AST is located in the NCB, which is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating conditions, AOOs, and postulated accidents.

7.6.8.1.5 Instrumentation and control (PDC 13)

PDC 13 requires instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. PSAR section 7.6.8.2.2 states that the AST monitors and measures ground motion, applies required processing and logic, and

automatically generates signals to open the RTBs. The AST monitors the anticipated range of seismic activities to ensure a reactor scram is initiated with adequate margin during a seismic event. The staff determined that the preliminary AST design is consistent with PDC 13 because the AST is designed to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

7.6.8.1.6 Protection system (PDC 24)

PDC 24 requires that the protection system remain fully functional and independent despite any single failure or shared component with the control system. Consistent with PDC 24, AST interconnects to the RPS are limited to the interface with the RTBs. The RPS is separated and isolated from the AST to the event that any single failure of an AST component or channel, or failure or removal from service of a single protection system component or channel common to the AST or RPS, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the RPS.

The staff determined that the preliminary AST design is consistent with PDC 24 because the AST is designed to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact the RPS satisfying all reliability, redundancy, and independence requirements.

7.6.8.1.7 Safety Functions

The AST supports the NSRST function DL4-RC4, "Automatic seismic trip," by continuously monitoring signal inputs from triaxial accelerometers and sending an isolated signal to the shunt trip feature in the RTBs when the signal inputs exceed a bi-stable setpoint. PSAR table 5.2-4 identifies that DL4-RC4 is required for DID adequacy with no associated LBEs. The staff determined that the preliminary AST design is consistent with DL4-RC4 because it provides adequate DID as the AST sends an isolated signal to the shunt trip feature in the RTB, diverse from the RPS, that opens the RTBs upon receipt of a seismic trip demand signal.

7.6.8.1.8 Programmatic Special Treatments

Programmatic special treatments applied to the AST are summarized in PSAR section 7.6.8.3 and include the QAP, D-RAP, EQ program, PITAP, and human factors engineering program. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the AST contains NSRST SSCs, the staff determined that these programs are thus appropriate to apply to AST. Plant programs are evaluated further in SE chapter 8, with the exception of the Human Factors Engineering Program Plan, which is evaluated in SE chapter 11.

7.6.8.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.6.8 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.7 Electrical Systems

PSAR section 7.7 describes the KU1 electrical systems, which include the following:

- NI auxiliary electrical system (NES)
- NI ancillary electrical system (NEA)
- Energy Island (EI) auxiliary electrical system

These systems are classified as NST, except for portions of the NES which are classified as NSRST and are described in PSAR section 7.7.1. PSAR section 7.7.1 states the NES consists of the NUP and the NI DC power supply system (NDC) to provide AC and DC power to NI systems and components during all modes of operation. Certain portions of the NUP and NDC are classified as NSRST and these portions are identified in PSAR figures 7.7.1-1 and 7.7.1-2. The NSRST portions of the NUP and NDC support the safety-significant functions listed in PSAR section 7.7.1.2. The NST portions of the NUP and NDC are identified in PSAR figures 7.7.1-1 and 7.7.1-2 and provide power to the NST control systems.

PSAR section 1.1.4.3.11 provides an overview of the NEA including a discussion on the NI lighting system (normal, egress, essential, and emergency operating lighting) and grounding, earthing, and lightning protection. PSAR section 1.1.4.3.14 provides an overview of the EI auxiliary electrical system including a discussion on the EI AC electrical power medium voltage system, EI AC electrical power low voltage system, EI DC power supply system, EI AC uninterruptible power supply system and EI heat tracing system. The staff evaluation of these SSCs is provided in SE section 1.3.3.1.2, where the staff determined that their preliminary NST classification and the information provided in the PSAR was consistent with the applicable guidance.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the NES:

- RG 1.75, Rev. 3.

PSAR sections 7.7.1 and 5.3 identify the following PDC as applicable to the NES: 1, 2, 3, 4, 17, 18.

7.7.1 Technical Evaluation

7.7.1.1 Quality Standards and Records (PDC 1)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed.

Evaluation of NUP

PSAR section 7.7.1.3.1 describes how the NUP provides uninterrupted AC electrical power to the NI systems and components listed in PSAR section 7.7.1.1.

As described in PSAR section 7.7.1.3.1, the NUP will provide power to the RPS through independent and redundant divisions (A, B, C, and D) consistent with IEEE 603-1991 and IEEE 379-2000.

PSAR section 7.7.1.3.1 states that during a loss of all AC power event, inverters remain powered by the batteries for 72 hours (Divisions A and B) and 2 hours (Divisions C and D) to keep the NSRST loads energized. In addition, there are provisions for connection of a portable diesel generator to provide power to the PAM loads beyond the (Divisions A and B) 72 hour battery duty.

PSAR table 1.4-1 and PSAR section 7.7.1.3.1 provide information on the independence of redundant NSRST divisions maintained by physical separation and electrical isolation between divisions, and between NST and NSRST SSCs, consistent with IEEE 384-1992 and partial conformance with RG 1.75. In addition, PSAR section 7.7.1.3.1 states that the design will ensure a postulated single failure affects only a single division, which is consistent with IEEE 379-2000.

PSAR table 1.4-5 lists IEEE 741-2007 as used in the design of the facility. The staff confirmed through audit that the design of the NUP is planned to conform to IEEE 741-2007. The staff determined that the NUP has sufficient protection and coordination in accordance with IEEE 741-2007. The extent of conformance to IEEE 741-2007 will be reviewed by the staff at the OL stage.

Based on its review described above, the staff determined that the preliminary NUP design is consistent with PDC 1 because the codes and standards specified for the design and construction of NUP SSCs are acceptable.

Evaluation of NDC

PSAR section 7.7.1.3.2 describes how the NDC provides uninterrupted DC electrical power to NI systems and components listed in PSAR section 7.7.1.1.

As described in PSAR section 7.7.1.3.2, the NDC will provide power to the RPS through independent and redundant divisions (A, B, C, and D), consistent with IEEE 603-1991 and IEEE 379-2000.

PSAR section 7.7.1.3.2 states that the batteries are sealed valve-regulated lead-acid batteries (VRLA) selected and designed per IEEE 1187-2013 and IEEE 1189-2007. This PSAR section also discusses that maintenance and testing will be performed on the batteries. During the audit, the staff confirmed that the VRLA maintenance and testing of batteries is planned to be performed in accordance with IEEE 1188-2005 (R2010), including the 2014 amendment.

PSAR section 7.7.1.3.2 states that the batteries are sized per IEEE 485-2020. The batteries for Divisions A and B were sized to supply 72 hours of power. The batteries for Divisions C and D were sized to supply 2 hours of power.

The staff reviewed the battery monitoring system's design's conformance with IEEE 1491-2012 as described in PSAR section 7.7.1.3.2, and determined that the battery monitoring system design adequately monitors the key battery parameters such as cell temperature, cell resistance, charging current, float voltage, and battery room temperature in order to prompt corrective action prior to failure.

PSAR table 1.4-5 lists IEEE 946-2020 and IEEE 1375-1998 as used in the design of the facility. The staff confirmed during its audit that the design of the NDC (including instrumentation, indication, and alarms) will follow the guidelines of IEEE 946-2020. The staff determined that the

NUP has sufficient protection and coordination in accordance with IEEE 946-2020 and IEEE 1375-1998.

PSAR table 1.4-1 and section 7.7.1.3.2 provide information on the independence of redundant NSRST divisions maintained by physical separation and electrical isolation between divisions, and between NST and NSRST SSCs, consistent with IEEE 384-1992 and partial conformance with RG 1.75. In addition, PSAR section 7.7.1.3.1 states that the design will ensure a postulated single failure affects only a single division, which is consistent with IEEE 379-2000.

Based on its review described above, the staff determined that the preliminary NDC design is consistent with PDC 1 because the codes and standards specified for the design and construction of NDC SSCs are acceptable.

7.7.1.2 Protection Against Natural Phenomena (PDC 2)

PDC 2 specifies that safety-significant SSCs shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PSAR section 7.7.1.2 states that the safety-significant portions of the NDC and NUP are located within and protected by the reinforced concrete NCB substructure, which is designed to withstand the effects of applicable natural phenomena without the loss of capability to perform safety-significant functions. The design of the NCB provides protection of SSCs that perform safety-significant functions from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. PSAR section 7.7.1.2 states that the NSRST components of the NDC and NUP are classified as SCN1 for seismic design, analysis, and qualification. PSAR section 7.7.1.2 states that the NSRST components of the NDC and NUP are classified as SCN1 for seismic design, analysis, and qualification. KU1's seismic design methods are described in PSAR section 6.4 and evaluated by the staff in SE section 6.4.1.1.

The staff determined that the preliminary NUP and NDC design is consistent with PDC 2 because the NSRST components of the NUP and NDC are located in the NCB substructure that is designed to withstand the effects of applicable natural phenomena without the loss of structural integrity and the NSRST components are classified as SCN1.

7.7.1.3 Fire Protection (PDC 3)

PDC 3 specifies that safety-significant SSCs shall be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.7.1.2 states that NUP and NDC NSRST components are designed and located to minimize the probability and effect of fires through the selection of non-combustible and non-fire sustaining materials to the extent practical. PSAR section 7.5.2.3 states that a fire protection program addressing those SSCs that could affect safety, or the protection of licensed radioactive materials will be included in the OL application. The staff determined that the preliminary NUP and NDC design is consistent with PDC 3 due to the use of low combustible materials and the planned fire protection program.

7.7.1.4 Environmental and Dynamic Effects Design Bases (PDC 4)

PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions. PSAR section 7.7.1.2 states that the NUP and NDC NSRST components are designed to be compatible with the environmental conditions associated with normal

operations, maintenance, testing, anticipated operational occurrences, and postulated accidents. These systems and components are located in the NCB substructure and are protected from dynamic effects of fluid, pipe whipping, discharge fluids, external hazards, including missiles.

The staff determined that the preliminary design of the NUP and NDC NSRST components is consistent with PDC 4 because the electrical systems and components are located in the NCB substructure and are protected from external hazards, including missiles and discharging fluids. The NCB substructure is designed to accommodate environmental conditions associated with normal operations and postulated accidents.

7.7.1.5 Electrical Power Systems (PDC 17)

PDC 17 requires that electric power systems shall be provided when required to permit the functioning of SSCs. In addition, if electric power is not needed for AOOs or postulated accidents, the design shall demonstrate that power for safety-significant functions is provided.

PSAR section 7.7.1.2 states that electric power is not required in order to meet the SARRDLs for AOOs or postulated accidents. The SR scram and ESF functions discussed in PSAR section 7.6 are accomplished without the need for electric power in response to DBAs.

PSAR section 7.7.1.2 states that power is provided for safety-significant functions by the NSRST NUP and NDC design. SSCs that require electrical power to perform NSRST functions are supplied through the NSRST electrical systems. Furthermore, PSAR sections 7.7.1.3.1 and 7.7.1.3.2 describe how the NUP and NDC are comprised of four independent divisions (A, B, C, and D), one for each RPS division. PSAR section 7.7.1.2 states, in part, that the NSRST electrical systems are designed with the capability to test periodically. The staff notes that the preliminary design description of the NUP and NDC NSRST SSCs has redundancy, independence, and testing capability.

The staff determined that the preliminary design of the NSRST NUP and NDC is consistent with PDC 17 because no electrical power is required to perform any SR function for safe shutdown of the plant or to keep the plant in a safe shutdown condition and the NUP and NDC will provide power for safety-significant functions.

7.7.1.6 Inspection and Testing of Electric Power Systems (PDC 18)

PDC 18 requires that safety-significant electric power systems shall be designed to permit appropriate periodic inspection and testing of important areas and features. PSAR section 7.7.1.2 states that NSRST electrical systems are designed with (1) the capability to periodically test the operability and functional performance of the components of the systems and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among systems. The staff determined that the preliminary design of the NSRST NUP and NDC is consistent with PDC 18 because the systems have testing capability.

7.7.1.7 Safety Functions

The NUP and NDC provide power to support the following NSRST safety functions, as shown in PSAR table 5.2-4:

- DL4-RC3 – Control Rod Drive Driveline Scram
- DL4-RC4 – Automatic Seismic Trip
- DL2-HR2 – Intermediate Sodium Pump Trip on Low Intermediate Heat Transport System Level
- DL4-HR2 – Primary Sodium Pump Trip Automatic Backup
- DL4-HR3 – Intermediate Sodium Pump Trip Automatic Backup
- DL4-HR6 – Manual Primary Sodium Pump Trip
- DL4-HR7 – Manual Intermediate Sodium Pump Trip
- DL4-RR3b – Sodium Processing System Pumps Trip on Leak Detection
- DL4-RR4c – Vapor Trap Cell Isolation on Overpressure
- DL4-RR7 – Fuel Handling Building Barrier
- DL4-RR8 – Manual Sodium Processing System Pump Trip on Leak Indication
- DL5-PAM1 – Post-Accident Monitoring

The staff determined that the preliminary designs of the NUP and NDC are consistent with the associated safety functions, based on the selected codes and standards discussed under PDC 1 and their alignment with the requirements of PDC 17 and 18.

7.7.1.8 Programmatic Special Treatments

Programmatic special treatments applied to the NUP and NDC are summarized in PSAR section 7.7.1.4 and include the QAP, D-RAP, EQ program, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the NUP and NDC contain NSRST SSCs, the staff determined that these programs are thus appropriate to apply to the NUP and NDC. Plant programs are evaluated further in SE chapter 8.

7.7.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.7 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.8 Buildings and Structures

PSAR section 7.8 describes the safety-significant buildings and structures located on the NI. The following buildings and structures evaluated in this section are:

- Reactor Building (RXB)
- Fuel Handling Building (FHB)
- Reactor Auxiliary Building (RAB)
- Nuclear Island Control Building (NCB)

7.8.1 Reactor Building

PSAR section 7.8.1 describes the RXB. This includes information on the design bases, structural design and performance, and special treatments. The RXB building consists of an embedded concrete substructure and above grade steel-framed superstructure. The structural configuration of the RXB is shown in cross-sectional views in PSAR figures 1.1-5 and 1.1-6. The PSAR states that the RXB superstructure is structurally isolated from the substructure. The RXB is located between the RAB and FHB as shown in the layout plan in PSAR figure 1.2-1. The substructures of the RAB and FHB are located approximately 16 ft west and 40 ft east of the RXB, respectively, and are structurally separated from the RXB. The RXB and RAB are functionally connected by a below grade non-integral corridor providing access to piping and other components of the IHT. The RXB and FHB are functionally connected above grade.

The below grade RXB substructure described in PSAR section 7.8.1.1 consists of a concrete cylindrical cavity and HAA. The foundations for the cylindrical cavity and the HAA are deeply embedded at approximately 118 ft and 55 ft below grade, respectively. The HAA foundation mat and shear wall support the grade slab, which is approximately 3.5 ft thick. The grade slab has a removable concrete plank to provide access to the HAA for refueling and other operational requirements.

The RXB substructure is classified as SR. The applicant determined functional requirements and design criteria for SR and NSRST SSCs through a plant-specific PRA described in PSAR section 3.1. The RXB supports SR passive RAC operation by maintaining the outer boundary of the RAC air flow path and NSRST functional containment barrier functions through the GV, HAA walls, and HAA HVAC isolation. The RXB substructure provides structural support and load path for the RAC stack and the RSS. The RSS transfers load from the RES to the RXB through seismic isolators. In addition, the RXB slabs and walls support NSRST SSCs. The RXB also provides temporary load paths for FHE SSCs that provide SR PSFs during refueling.

The above grade RXB superstructure described in PSAR section 1.1.4.4.2 is a steel-framed structure with metal siding. The rectangular layout is centered above the reactor, while the concrete foundation system is independent and not structurally connected to the RXB substructure. The superstructure is functionally connected to the FHB to allow conveyance of the EVHM for refueling and supports the overhead bridge crane used for conducting maintenance activities. The RXB superstructure is classified as NST.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the RXB:

- RG 1.142, Rev. 3;
- RG 1.243, Rev. 0;

- RG 1.199, Rev. 1;
- RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Rev. 3 (ML12220A043);
- RG 1.61, Rev. 1;
- RG 1.76, Rev. 1;
- RG 1.69, Rev. 1;
- RG 1.143, Rev. 2.

PSAR sections 7.8.1 and 5.3 identify the following PDC as applicable to the RXB: 1, 2, 3, 4, 16, 35, 36, 37, 81, 82.

7.8.1.1 Technical Evaluation

SE sections 7.8.1.1.1 through 7.8.1.1.9 provide the technical evaluation for the SR RXB substructure. The technical evaluation for the NST RXB superstructure is provided in section 7.8.1.1.10 of this SE.

7.8.1.1.1 Overall Design Requirements (PDC 1, 2, and 4)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. The design bases for these SSCs shall consider: (1) the most severe of the natural phenomena; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and (3) the safety significance of the functions to be performed. PDC 4 requires safety-significant SSCs to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, AOOs, and postulated accidents. PDC 4 also requires dynamic effects such as missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events outside the nuclear power unit.

PSAR section 7.8.1.1.2 states that RXB substructure is designed to withstand natural phenomena for the protection of SR and NSRST SSCs to perform their safety-significant functions. The seismic classification of the RXB substructure is SCS1. The RXB substructure is designed based on applicable codes and standards. The loads and load combinations used for the design include loads for construction, normal plant start-up, operations, shutdown, and design basis accidents.

USO's design basis hazards are developed based on site characterization as documented in PSAR chapter 2. The staff reviewed the DBHLs for external hazards discussed in PSAR section 6.1.1 and summarized in PSAR table 6.1-1 and documented its evaluation in SE chapter 2 and section 6.1.1. The staff reviewed the loads and load combinations and the applicable codes and standards including RGs for SR structural design in SE section 6.4.1.2.1.

PSAR section 7.8.1.1.3 references PSAR chapter 6 for SR structural characteristics, materials and construction techniques, design and analysis procedures, and structural acceptance criteria.

In SE sections 6.4.1.1.3 and 6.4.1.2.1, the staff reviewed and evaluated material properties; analysis procedures; input ground motion for seismic design; soil profile and properties; soil-structure interaction (SSI) modeling including interaction with nearby structures; structural modeling and response analysis methodology; in structure response spectra; and foundation stability for SR structures.

The staff notes that the descriptions and figures in PSAR chapter 1 provide a reasonable description of the structural configuration of the RXB substructure and PSAR section 7.8.1 provides a preliminary description of the functional supports, load path, and structural supports for SR and NSRST SSCs that is reasonable. SE section 6.4.1.2.1 provides the evaluation of the SR structural design bases, analysis and methodology and determined that the applicant adequately described (a) applicable codes and standards and regulatory guides for structural design; (b) load and load combinations; (c) seismic design motion; (d) methodology for modeling SSI and structure-soil-structure interaction (SSSI) with nearby structures; (e) proposed methodology for structural analysis and design; and (f) proposed materials for construction for SR structures. Based on the evaluation in SE section 6.4.1.2.1 and the other information noted above, the staff determined that the preliminary design information for the RXB substructure is consistent with PDC 1, 2, and 4.

7.8.1.1.2 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.8.1.1.2 states that the RXB substructure is designed with plant features that minimize the probability and effect of fires and explosions by the use of low combustible materials and physical separation. The staff's evaluation of the KU1 fire protections systems and program can be found in SE sections 7.5.2 and 8.3. Based on review of the information in PSAR section 7.8.1.1.2 and the evaluation in SE sections 7.5.2 and 8.3, the staff determined the preliminary RXB substructure design is consistent with PDC 3.

7.8.1.1.3 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. PSAR section 7.8.1.1.2 states that the HAA, which is part of the RXB substructure, provides a functional containment boundary that performs a radionuclide retention function. Based on the evaluation of the design and construction of the RXB substructure, including the HAA, in SE section 7.8.1.1.1, the staff determined that the preliminary design information for the RXB substructure is consistent with PDC 16.

7.8.1.1.4 Emergency Core Cooling (PDC 35)

PDC 35 requires sufficient core cooling during postulated accidents. PSAR section 7.8.1.1.2 states that the RXB substructure houses, protects, and supports the below grade portion of the RAC (evaluated in SE section 7.2.1) within the RAC enclosures and the reactor cavity. The RXB substructure reactor cavity walls form a portion of the RAC inlet air downcomer. The walls of the RXB substructure below the operating deck are part of the air plenum for the RAC. The ability of

the RAC to provide adequate core cooling and remove residual heat following postulated accidents is demonstrated by the LBE analyses in PSAR chapter 3. The staff determined that the preliminary RXB substructure design is consistent with PDC 35 because it supports the operation of the RAC to dissipate the necessary heat during LBEs. The staff will confirm that the final design of the RXB substructure supports adequate RAC performance at the OL stage.

7.8.1.1.5 Inspection and Testing of Emergency Core Cooling Systems (PDC 36 and 37)

PDC 36 and 37 address inspection and testing for systems required for emergency core cooling. PSAR section 7.8.1.1.2 states that the RXB substructure is designed to allow for access for the performance of MANDE activities for the RAC as a part of the RIM Program and to permit inspections of portions of the RXB that provide support for the RAC and form part of the RAC air flow path. Additionally, the PSAR states that the RXB provides access to permit periodic functional testing of the RAC. The staff determined that the preliminary RXB substructure design is consistent with PDC 36 and 37 because it allows for periodic inspections and testing to ensure the system's integrity and capability to perform its SR function. The staff will confirm that the final design of the RXB substructure supports periodic inspections and testing of RAC at the OL stage.

7.8.1.1.6 Reactor Building Design Basis (PDC 81)

PDC 81 requires the reactor building to be designed to structurally protect the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink during postulated accidents. PSAR section 7.8.1.1.2 states that the RXB substructure is designed to structurally support and protect the RES and RAC such that the geometry for passive heat removal from the reactor core is maintained during DBAs. Based on the evaluation of the structural design bases and the analysis methodology in SE section 6.4.1.2.1 and the evaluation of the design requirements in SE section 7.8.1.1.1, the staff determined that the preliminary design of the RXB substructure is configured to structurally protect the geometry for passive heat removal and is thus consistent with PDC 81.

7.8.1.1.7 Inspection of Reactor Building (PDC 82)

PDC 82 requires the reactor building to be designed to permit periodic inspection and surveillance of important structural areas. PSAR section 7.8.1.1.2 states the HAA and RAC enclosures are accessible for inspection and surveillance via removable access plugs on the ground floor. The cylindrical reactor cavity is accessible by remotely operated inspection equipment. Based on these abilities to inspect key areas, the staff determined that the preliminary RXB substructure design is consistent with PDC 82.

7.8.1.1.8 Safety Functions

PSAR section 7.8.1.1.2 identifies the following safety-significant functions associated with the RXB:

- DL3-HR4 Inherent RAC Operation (SR)
- DL4-RR1 Ex-RES Functional Containment Barrier (NSRST)
- DL4-RR1a HAA Barrier (NSRST)

- Load Path for SR SSCs (SR), and for NSRST SSCs (NSRST)
- Temporary Load Path for SR SSCs (NSRST)
- External Hazard Protection for SR SSCs (SR), and for NSRST SSCs (NSRST)

The RXB substructure supports SR DL3-HR4 “Inherent RAC operation” function by maintaining the outer boundary of the RAC air flow path. The RXB substructure houses the CCA and the inlet and outlet ducting and includes RAC enclosure structures that house the lower portion of the inlet and outlet stacks. The ability of the RAC to transfer heat during accident conditions is evaluated in SE section 7.2.1.

The NSRST barrier function DL4-RR1 refers to the functional containment boundary formed around the RES by the HAA barrier, the GV, and the HAA HVAC isolation. The sub-function DL4-RR1a represents the HAA portion of the boundary. The RXB substructure contributes to isolation of radionuclides to control the release of radionuclides to the environment.

The RXB substructure provides structural support and load path for the RAC stack and the RSS. The RSS transfers load from the RES to the RXB substructure through seismic isolators. The design and qualification methodology for the SIS are described in NAT-8922 and discussed in SE section 6.4.1.3. In addition, the RXB slabs and walls provide structural support load path for NSRST SSCs.

The RXB substructure also provides temporary load paths for FHE SSCs which provide SR PSFs during refueling. In addition, the RXB substructure provides external hazard protection for SR and NSRST SSCs located within the RXB substructure.

Based on the evaluation of the structural design bases and the analysis methodology in SE section 6.4.1.2.1 and the evaluation of the design description and requirements in SE section 7.8.1.1.1, the staff determined that the preliminary design of the RXB substructure is consistent with the six identified safety-significant functions listed at the beginning of this section.

7.8.1.1.9 Programmatic Special Treatments

PSAR section 7.8.1.1.4 states that the identification of special treatment for the SR RXB substructure is preliminary and is based on its safety classification (described in PSAR section 6.3) and construction codes and standards selected for design and construction (discussed in PSAR 6.4.3.2.1). The programmatic special treatments applied to the RXB include QAP, D-RAP and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the RXB contains SR and NSRST SSCs, the staff determined that these programs are thus applicable to the RXB. Plant programs are evaluated further in SE chapter 8.

7.8.1.1.10 RXB Superstructure

USO classified the RXB superstructure as NST because it does not perform safety-significant PSFs. Additionally, as stated in PSAR section 7.8.1, the RXB superstructure does not provide load path support or external hazards protection for safety-significant SSCs. There are no PDC

associated with the design of NST structures. The above-grade RXB is made of steel moment frames placed in two orthogonal directions supported on independent foundation and includes an overhead bridge crane. PSAR section 7.8.1 states that the superstructure is designed to prevent interactions or adverse impacts on SR SSCs (e.g., RXB substructure, RAC stacks, and fuel handling equipment), or nearby SR or NSRST structures. PSAR section 6.1.3 discusses the design requirements for NST structures for the prevention of seismic interaction at the design basis ground motion level and stated that the NST SSCs will be designed in accordance with ASCE 7-16 applying response parameters consistent with ASCE 43-19 Limit State B. While there is no special treatment associated with NST structures, seismic interaction sources, including NST SSCs, have design, analysis, and programmatic controls with a quality level consistent with that of NSRST SSCs, as stated in PSAR section 6.1.3.1. PSAR section 7.8.1.2 states that the RXB superstructure is a credible source for interaction due to proximity with the SR RXB substructure, SR RAC stacks, and SR fuel handling equipment during refueling operations. In addition, section 7.8.1.2 states that several aspects of the RXB superstructure design will be in accordance with ASCE 43-19, including evaluation of the RXB superstructure seismic force resisting system; load combinations for evaluating seismic interaction of RXB superstructure and concrete foundation; analytical models and methods for evaluating seismic interaction; and strength-based and deformation-based acceptance criteria for the RXB superstructure seismic force resisting system.

SE sections 6.1.1 and 6.4.1 address DBHL-related requirements for non-SR (NSRST or NST) SSCs and seismic interaction, respectively. The staff notes that the descriptions and the figures in PSAR chapter 1 and PSAR section 7.8.1.2 provide an adequate understanding of the structural configuration of the RXB superstructure. Based on the evaluation of DBHL-related requirements for non-SR SSCs and seismic interaction in SE chapter 6, the design descriptions in PSAR chapter 1 and PSAR section 7.8.1.2, and the plans for a seismic PRA to further inform the structural design at the OL stage, the staff determined that the preliminary design information for the RXB superstructure is acceptable at the CP stage.

7.8.1.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.8.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.8.2 Fuel Handling Building

PSAR section 7.8.2 describes the FHB. The FHB includes a below grade SR concrete substructure and an above grade NSRST superstructure which consists of a steel-framed structure and a reinforced concrete structure. The structural configuration of the FHB is shown in cross-sectional views in PSAR figures 1.1-7 to 1.1-9. The substructure provide load paths for structural support for SR and NSRST SSCs.

PSAR section 7.8.2.1 states that the below grade FHB concrete substructure contains several areas separated by thick concrete walls with foundations at different levels. The SFP located in the FHB substructure has 5 ft thick walls and a 4 ft thick foundation slab and includes a stainless-steel liner for leakage control. The foundation level of the SFP is approximately 51 ft below grade. The concrete walls and foundation slab for other sub-grade areas are 3 ft thick. The foundation for one subgrade area that supports the FHE EVST and radwaste processing system is approximately 30 ft below grade and the foundation level of the FHE PRC is

approximately 45 ft below grade. The FHB grade slab, which supports the FHE and cask transfer system, is 3 ft thick.

The NSRST superstructure consists of a steel framed building and an adjoining north wing concrete building. The high steel framed main building is supported by the FHB substructure concrete grade slab. The superstructure provides load paths for NSRST SSCs. The steel frame structure includes an overhead crane for handling spent fuel casks and performing maintenance activities. The north wing is supported by separate foundations and contains solid and liquid radwaste systems and additional equipment, such as for HVAC and water systems. The FHB is functionally connected to the RXB at the grade level by a NSRST steel framed corridor providing temporary structural support for the EVHM as it travels between the FHB and RXB. In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the FHB:

- RG 1.142, Rev. 3;
- RG 1.243, Rev. 0;
- RG 1.199, Rev. 1;
- RG 1.92, Rev. 3;
- RG 1.61, Rev. 1;
- RG 1.76, Rev. 1;
- RG 1.69, Rev. 1;
- RG 1.143, Rev. 2; and
- RG 1.13, Rev. 2.

PSAR sections 7.8.2 and 5.3 identify the following PDC as applicable to the FHB: 1, 2, 3, 4, and 16.

7.8.2.1 Technical Evaluation

7.8.2.1.1 Overall Design Requirements (PDC 1, 2, and 4)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions.

FHB substructure

PSAR section 7.8.2.1.2 states that the FHB substructure is seismically classified as SCS1. The FHB substructure is designed in accordance with applicable PDC including PDC 2 and PDC 4. PSAR section 7.8.2.1.2 states that the FHB substructure is designed to protect SR and NSRST SSCs from design basis external hazards. The FHB substructure is designed based on applicable codes and standards. The loads and load combinations used for the design include loads for construction, normal plant start-up, operations, shutdown, and design basis accidents.

USO's design basis hazards are developed based on site characterization. The staff reviewed the DBHLs for external hazards discussed in PSAR section 6.1.1 and summarized in PSAR table 6.1-1. The review findings are documented in SE chapter 2 and SE section 6.1.1. The staff reviewed the loads and load combinations and the applicable codes and standards including RGs for SR structural design in SE section 6.4.1.2.1.

PSAR section 7.8.2.1.3 provides information on the structural characteristics that support RSF, materials and construction techniques, design and analysis procedures, and structural acceptance criteria.

PSAR section 7.8.2.1.3 states that the structural design of the reinforced concrete FHB SR substructure and SR SFP are in accordance with ACI 349-13 and RG 1.142, and the SFP steel liner is designed to meet the requirements of ANSI/AISC N690-18.

In SE sections 6.4.1.1.3 and 6.4.1.2.1, the staff reviewed and evaluated structural material properties; analysis procedures; input ground motion for seismic design; soil profile and properties; SSI modeling, including interaction with nearby structures; structural modeling and response analysis methodology; in structure response spectra; and foundation stability for SR structures.

The staff notes that the descriptions and figures in PSAR chapter 1 provide a reasonable description of the structural configuration of the FHB substructure and PSAR section 7.8.2 adequately describes the functional supports, load path, and structural supports for SR and NSRST SSCs. SE section 6.4.1.2.1 provides the evaluation of the SR structural design bases, analysis, and methodology and determined that the applicant adequately described (a) applicable codes and standards and regulatory guides for structural design; (b) load and load combinations; (c) seismic design motion; (d) methodology for modeling SSI and SSSI with nearby structures; (e) proposed methodology for structural analysis and design; and (f) proposed materials for construction for SR structures. Based on the evaluation in SE section 6.4.1.2.1 and the other information noted above, the staff determined that the preliminary design information for the FHB substructure is consistent with PDC 1, 2, and 4.

FHB superstructure

The FHB superstructure is classified as NSRST. PSAR section 7.8.2.2 states that the seismic classification of the FHB superstructure is SCN1. The above grade FHB main structure consists of steel moment frames placed in two orthogonal directions. The north wing section of the FHB is a concrete structure consisting of shear walls. The steel frame structure is designed in accordance with ANSI/AISC 360-16 and the concrete structure is designed in accordance with ACI 318-19. The seismic demand is developed using ASCE 7-16. The FHB steel structure is founded on the grade slab of the FHB SR substructure. The design ground motion of SR FHB substructure is SSE. The in-structure floor response spectra for the at-grade slab is considered the risk targeted maximum considered earthquake modified by an importance factor and a response coefficient in accordance with ASCE 7-16 and the USO seismic classification.

The FHB superstructure is classified as NSRST. The NRC has not established regulatory guidance or positions for NSRST/NST structural review. The detailed findings on the applicant's approach for NSRST structural design are discussed in SE section 6.4.1.2.2. The applicant's proposed approach depends on defining the design performance targets and demonstrating that those targets are met using the seismic PRA (SPRA). According to USO, the SPRA will not be available until the OL stage, and therefore, the performance targets cannot be confirmed at this time. Nonetheless, the staff determined that the design described in the PSAR provides reasonable assurance that performance targets can be met at the OL stage. Additionally, the applicant has noted that they are developing and iterating on their design and will make any necessary modifications to achieve the performance targets.

In SE section 6.4.1.2.2, the staff reviewed and evaluated the structural material properties; loads and load combinations; design codes and standards; methodology for seismic analysis and structural modeling; and structural design for NSRST structures. Based on the evaluation in SE section 6.4.1.2.2 and the other information noted above, the staff determined that the preliminary design information for the FHB superstructure is consistent with PDC 1, 2, and 4.

7.8.2.1.2 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.8.2 states that the FHB substructure and superstructure are designed with plant features that minimize the probability and effect of fires and explosions including the use of noncombustible or fire-resistive materials. The staff evaluation of the KU1 fire protections systems and program can be found in SE sections 7.5.2 and 8.3. Based on review of the information in PSAR section 7.8.2 and the evaluation in SE sections 7.5.2 and 8.3, the staff determined the preliminary FHB substructure design is consistent with PDC 3.

7.8.2.1.3 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. PSAR section 7.8.2.2.2 states that the FHB superstructure is designed as an enveloping barrier to support maintaining a negative pressure for radionuclide retention. Based on the evaluation of the design and construction of the FHB superstructure in SE section 7.8.2.1.1, the staff determined that the preliminary design information for the FHB superstructure is consistent with PDC 16.

7.8.2.1.4 Safety Functions

The FHB substructure supports the following safety-significant functions:

- External Hazard Protection for SR SSCs (SR), and for NSRST SSCs (NSRST)
- Load Path for SR SSCs (SR), and for NSRST SSCs (NSRST)

The FHB substructure is designed to protect SR and NSRST SSCs from the effects of design-basis external hazards and provides the load path for structural support for SR and NSRST SSCs.

The FHB superstructure supports the following safety-significant functions:

- DL4-RR7 – Fuel Handling Building Barrier
- Temporary Load Path for SR SSCs
- Load Path for NSRST SSCs
- External Hazard Protection for NSRST SSCs

The FHB superstructure supports NSRST DL4-RR7 function by maintaining a slightly negative pressure as part of the enveloping barrier to reduce the amount of radionuclides escaping for fuel drop events. The FHB superstructure also provides the load path for temporary structural support for the EVHM as it performs its SR PSFs. In addition, the FHB superstructure is designed to protect NSRST SSCs from the effects of external hazards and provides the load path for structural support for NSRST SSCs.

Based on the evaluation of the structural design bases and the analysis methodology for SR structures (FHB substructure) in SE section 6.4.1.2.1 and for NSRST structures (FHB superstructure) in SE section 6.4.1.2.2 and the evaluation of the design description and requirements in SE section 7.8.2.1.1, the staff determined that the preliminary design of the FHB substructure and superstructure are consistent with the identified safety-significant functions.

7.8.2.1.5 Programmatic Special Treatments

Programmatic special treatments applied to the FHB substructure and FHB superstructure are summarized in PSAR sections 7.8.2.1.4 and 7.8.2.2.4 respectively. These special treatments include the QAP, D-RAP, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the FHB contains SR and NSRST SSCs, and the staff determined that these programs are thus appropriate to apply to FHB. Plant programs are evaluated further in SE chapter 8.

7.8.2.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.8.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.8.3 Reactor Auxiliary Building

PSAR section 7.8.3 describes the RAB. The RAB consists of a reinforced concrete below grade substructure and a steel framed above grade superstructure with metal siding and metal roof. The structural configuration of the RAB is shown in cross-sectional views in PSAR figures 1.1-10 and 1.1-11. Both below and above grade RAB structures support NSRST systems and are classified as NSRST. The RAB is connected to the RXB below grade for piping access between the two buildings and a personnel passage above grade.

The RAB substructure consists of concrete walls and slabs. As shown in PSAR figures 1.1-10 and 1.1-11, the substructure is divided into two levels. The larger upper level located approximately 30 ft below grade has a foundation slab that is 5 ft thick. The lower-level RAB

foundation is 3 ft thick and located approximately 62 ft below grade. The substructure consists of 4.5 ft thick exterior walls and a 3 ft thick roof slab at grade level. The structural design requirements are PDC 2 and PDC 4. The seismic classification of the RAB substructure is SCN1.

The RAB superstructure is a steel-framed building with metal siding and metal roof. The RAB superstructure is supported on the substructure concrete grade slab. The seismic classification of the RAB superstructure is SCN1.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the RAB:

- RG 1.142, Rev. 3;
- RG 1.243, Rev. 0;
- RG 1.199, Rev. 1;
- RG 1.92, Rev. 3
- RG 1.61, Rev. 1;
- RG 1.76, Rev. 1;
- RG 1.69, Rev. 1;
- RG 1.143, Rev. 2.

PSAR sections 7.8.3 and 5.3 identify the following PDC as applicable to the RAB: 1, 2, 3, 4, and 16.

7.8.3.1 Technical Evaluation

7.8.3.1.1 Overall Design Requirements (PDC 1, 2, and 4)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions.

USO states that the NSRST RAB superstructure and substructure are required to meet PDC 2 and PDC 4. In SE section 6.4.1.2.2, the staff reviewed and evaluated structural material properties; loads and load combinations, design codes and standards, methodology for seismic analysis and structural modeling; and structural design for NSRST structures. As discussed in SE section 7.8.2.1.1, the adequacy of the seismic design will be confirmed or adjusted at the OL to meet required performance targets based on the results of the SPRA, however, based on the

evaluation in SE section 6.4.1.2.2, the staff determined that the preliminary design information for the RAB substructure and superstructure are consistent with PDC 1, 2, and 4.

7.8.3.1.2 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.8.3.1.2 states that the RAB substructure and superstructure are designed with plant features that minimize the probability and effect of fires and explosions including the use of noncombustible or fire-resistive materials. The staff's evaluation of the KU1 fire protection systems and program can be found in SE sections 7.5.2 and 8.3. Based on review of the information in PSAR section 7.8.3.1.2 and the evaluation in SE sections 7.5.2 and 8.3, the staff determined the preliminary RAB substructure design is consistent with PDC 3.

7.8.3.1.3 Containment Design (PDC 16)

PDC 16 requires a reactor functional containment to control the release of radioactivity to the environment and ensure functional containment safety-significant design conditions are not exceeded during accidents. PSAR section 7.8.3.12 states that the RAB substructure SPS cells are designed as an enveloping boundary for functional containment for radionuclide retention. Based on the evaluation of the design and construction of the RAB substructure, including the SPS cell enclosures, in SE section 7.8.3.1.1, the staff determined that the preliminary design information for the RAB substructure is consistent with PDC 16 and further information can reasonably be left for later consideration. Additionally, as noted in SE sections 3.2.1.4 and 5.1, additional design information to support the bases for the nominal and degraded performance of the SPS cell barrier used in the source term analysis is expected to be provided at the OL stage.

7.8.3.1.4 Safety Functions

The RAB substructure supports the following safety-significant functions:

- DL4-RR3 SPS Cells Barrier
- External Hazard Protection for NSRST SSCs
- Load Path for NSRST SSCs

The RAB superstructure supports the following safety-significant functions:

- External Hazard Protection for NSRST SSCs
- Load Path for NSRST SSCs

DL4-RR3 requires that the RAB substructure provides an enveloping boundary for radionuclide retention in the SPS cells. The preliminary design includes limited design information on the SPS cell enclosures. The DL4-RR3 function and its associated LBE are discussed in greater detail in SE sections 3.2.1, 3.6.1.3, 3.8.3.4, 5.1, and 5.5, which note USO is expected to provide bases for the nominal and degraded performances of radionuclide retention barriers used in the source term analyses at the OL stage. Based on the evaluation of this function in these sections

as well as the evaluation of the overall RAB substructure design in SE section 7.8.3.1.1, the staff determined that the preliminary design information for DL4-RR3 is sufficient at the CP stage.

Based on the evaluation of the structural design bases and the analysis methodology for NSRST structures in SE section 6.4.1.2.2 and the evaluation of the design description and requirements in SE section 7.8.3.1.1, the staff determined that the preliminary design of the RAB is consistent with the identified safety-significant functions for external hazard protection and load path for NSRST SSCs.

7.8.3.1.5 Programmatic Special Treatments

Programmatic special treatments applied to the RAB substructure and superstructure are summarized in PSAR sections 7.8.3.1.4 and 7.8.3.2.4, respectively. These special treatments include the QAP, D-RAP, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the RAB contains NSRST SSCs, the staff determined that these programs are thus appropriate to apply to RAB. Plant programs are evaluated further in SE chapter 8.

7.8.3.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.8.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.8.4 Nuclear Island Control Building

PSAR section 7.8.4 describes the NCB. The NCB consists of a below grade reinforced concrete substructure and steel framed superstructure. The general arrangements in plan and cross-sectional views are shown in PSAR figures 1.1-12 to 1.1-15. The below grade substructure and above grade superstructure are classified as SR and NSRST, respectively. The below grade concrete structure is divided into several compartments and has 2 ft thick interior walls and 3 ft thick exterior walls. The foundation is located 33 ft below grade and is 3 ft thick and the roof slab is 2 ft thick. The superstructure houses the MCR and is a steel framed metal building. The MCR is surrounded by a reinforced enclosure to protect the MCR computer and other components from chemical, radiological, and weather-related events.

In addition to the regulations and guidance provided at the beginning of this chapter, the staff used the following guidance in its evaluation of the NCB:

- RG 1.142, Rev. 3;
- RG 1.243, Rev. 0;
- RG 1.199, Rev. 1;
- RG 1.92, Rev. 3;
- RG 1.61, Rev. 1;

- RG 1.76, Rev. 1;
- RG 1.69, Rev. 1;
- RG 1.143, Rev. 2.

PSAR sections 7.8.4 and 5.3 identify the following PDC as applicable to the NCB: 1, 2, 3, 4, and 19.

7.8.4.1 Technical Evaluation

7.8.4.1.1 Overall Design Requirements (PDC 1, 2, and 4)

PDC 1 requires safety-significant SSCs to be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. PDC 2 requires safety-significant SSCs to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. PDC 4 requires safety-significant SSCs to withstand the effects of environmental conditions associated with normal operations and postulated accidents without loss of capability to perform their safety functions.

NCB Substructure

PSAR section 7.8.4.1.2 states that the NCB substructure is designed in accordance with applicable PDC including PDC 2 and PDC 4. The seismic classification of the NCB substructure is SCS1.

In SE section 6.4.1.2.1, the staff reviewed and evaluated material properties; design basis hazard levels; analysis procedures; input ground motion for seismic design; soil profile and properties; SSI modeling including interaction with nearby structures; structural modeling and response analysis methodology; in structure response spectra; and foundation stability for SR structures.

The staff notes that the descriptions and figures in PSAR chapter 1 provide a reasonable description of the structural configuration of the NCB substructure and PSAR section 7.8.4 adequately describes the functional supports, load path, and structural supports for SR and NSRST SSCs. SE section 6.4.1.2.1 provides the evaluation of the SR structural design bases, analysis and methodology and determined that the applicant adequately described (a) applicable codes and standards and RGs for structural design; (b) load and load combinations; (c) seismic design motion; (d) methodology for modeling SSI and SSSI with nearby structures; (e) proposed methodology for structural analysis and design; and (f) proposed materials for construction for SR structures. Based on the evaluation in SE section 6.4.1.2.1 and the other information noted above, the staff determined that the preliminary design information for the NCB substructure is consistent with PDC 1, 2, and 4.

NCB Superstructure

PSAR section 7.8.4.2.2 states that the NSRST NCB superstructure is required to meet PDC 2 and PDC 4. The seismic classification is SCN1. In SE section 6.4.1.2.2, the staff reviewed and evaluated structural material properties; loads and load combinations, design codes and standards, methodology for seismic analysis and structural modeling; and structural design for

NSRST structures. As discussed in SE section 7.8.2.1.1, the adequacy of the seismic design will be confirmed or adjusted at the OL to meet required performance targets based on the results of the SPRA, however, based on the evaluation in SE section 6.4.1.2.2, the staff determined that the preliminary design information for the NCB superstructure is consistent with PDC 1, 2, and 4.

7.8.4.1.2 Fire Protection (PDC 3)

PDC 3 requires safety-significant SSCs to be designed and located to minimize the probability and effect of fires and explosions. PSAR section 7.8.4 states that the NCB substructure and superstructure are designed with plant features that minimize the probability and effect of fires and explosions including the use of noncombustible or fire-resistive materials. The staff evaluation of the KU1 fire protections systems and program can be found in SE sections 7.5.2 and 8.3. Based on review of the information in PSAR section 7.8.4 and the evaluation in SE sections 7.5.2 and 8.3, the staff determined the preliminary NCB substructure design is consistent with PDC 3.

7.8.4.1.3 Control Room (PDC 19)

PDC 19 requires, in part, that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions and adequate habitability measures be provided to permit access and occupancy of the control room during normal operations and under accident conditions. PSAR section 7.8.4.1.2 states that the NCB substructure includes the RSC that contains instrumentation and controls for monitoring of the plant if the MCR is evacuated. PSAR section 7.8.4.2.2 identifies that the NCB superstructure provides radiation and external hazards protection for the MCR. Based on the evaluation of the design and construction of the NCB substructure and superstructure in SE section 7.8.4.1.1, the staff determined that the preliminary design information for the NCB substructure and superstructure is consistent with PDC.

7.8.4.1.4 Safety Functions

The NCB substructure supports the following safety-significant functions:

- External Hazard Protection for SR SSCs (SR), and for NSRST SSCs (NSRST)
- Load Path for SR SSCs (SR), and for NSRST SSCs (NSRST)

The NCB superstructure supports the following safety-significant functions:

- External Hazard Protection for NSRST SSCs
- Load Path for NSRST SSCs

Based on the evaluation of the structural design bases and the analysis methodology for SR structures (NCB substructure) in SE section 6.4.1.2.1 and for NSRST structures (NCB superstructure) in SE section 6.4.1.2.2 and the evaluation of the design requirements in SE section 7.8.4.1.1, the staff determined that the preliminary design of the NCB substructure and superstructure are consistent with the identified safety-significant functions.

7.8.4.1.5 Programmatic Special Treatments

Programmatic special treatments applied to the NCB substructure and superstructure are summarized in PSAR sections 7.8.4.1.4 and 7.8.4.2.4 respectively. These special treatments include the QAP, D-RAP, and PITAP. The applicant stated in PSAR chapter 8 that these programs generally apply to all SR and NSRST SSCs. Because the NCB contains SR and NSRST SSCs, and the staff determined that these programs are thus appropriate to apply to NCB. Plant programs are evaluated further in SE chapter 8.

7.8.4.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 7.8.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section and at the beginning of this chapter for the issuance of a CP.

7.9 References

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8 PLANT PROGRAMS

This chapter of the safety evaluation (SE) describes the staff's review and evaluation of the Kemmerer Power Station Unit 1 (KU1) preliminary safety analysis report (PSAR) chapter 8, which contains a preliminary description of plant programs, quality assurance, and fire protection programs.

The applicable regulatory requirements for the evaluation of plant programs covered in this chapter at the construction permit (CP) stage are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(3) – Contents of applications; technical information;
- 10 CFR 50.34(a)(7);
- 10 CFR 50.34(f)(2)(xxvi)
- 10 CFR 50.35, "Issuance of construction permits;"
- 10 CFR 50.40, "Common standards;"
- 10 CFR 50.55a(h) – Codes and standards;
- 10 CFR 50.55(f) – Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses;
- 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors;" and
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B.

The following regulatory requirements relevant to KU1 plant programs are applicable at the operating license (OL) stage:

- 10 CFR 50.34(b)(6)(iv);
- 10 CFR 50.43(e)(1) – Additional standards and provisions affecting class 103 licenses and certifications for commercial power;
- 10 CFR 50.48(a) – Fire protection;
- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants;"
- 10 CFR 50.65(a) – Requirements for monitoring the effectiveness of maintenance at nuclear power plants; and
- 10 CFR 50.65(b).

Though the staff did not make determinations regarding these OL requirements, the staff ensured the preliminary CP information would support compliance with these requirements at the OL stage. The staff notes that US SFR Owner, LLC (USO) requested an exemption to 10 CFR 50.65 which would alter the scope of the required maintenance program, as documented in enclosure 4, "Regulatory Exemptions," of the KU1 CP application. However, as discussed in Appendix B of this SE, the staff is not granting the requested exemption because the requirements of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," do not apply to an applicant for or holder of a CP. As appropriate, the staff would consider a future request for an exemption from 10 CFR 50.65 for KU1 if requested.

The relevant guidance for the evaluation of the plant programs is as follows:

- Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor internals During Preoperational and Startup Testing," Revision (Rev.) 4 (ML16056A338);
- RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Rev. 2 (ML22325A263);
- RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Rev. 4 (ML19312C533);
- RG 1.189, "Fire Protection for Nuclear Power Plants," Rev. 5 (ML23214A263);
- RG 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants," Rev. 0 (ML20166A001);
- RG 1.246, "Acceptability of ASME Code, Section XI, Division 2, 'Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,' for Non-Light Water Reactors," Rev. 0 (ML22111A087);
- RG 1.253, "Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Advanced Reactors," (ML23269A222);
- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," (ML23277A139);
- DANU-ISG-2022-05, "Advanced Reactor Content of Application Project chapter 11, 'Organization and Human-System Considerations' Interim Staff Guidance," March 2024 (ML23277A143); and
- DANU-ISG-2022-07, "Risk-informed Inservice Inspection/Inservice Testing Programs for Non-LWRs," (ML23277A231).

Definitions of Principal Design Criteria (PDC) are included in PSAR section 5.3. The following PDC were identified as applicable to the plant programs as follows:

- PDC 1, “Quality Standards and Records”
- PDC 3, “Fire Protection”
- PDC 4, “Environmental and Dynamic Effects Design Bases”
- PDC 14, “Primary Coolant Boundary”
- PDC 15, “Primary Coolant System Design”
- PDC 19, “Control Room”
- PDC 30, “Quality of Primary Coolant Boundary”
- PDC 31, “Fracture Prevention of Primary Coolant Boundary”
- PDC 32, “Inspection of Primary Coolant Boundary”
- PDC 36 “Inspection of Emergency Core Cooling System”
- PDC 37, “Testing of Emergency Core Cooling System”
- PDC 45, “Inspection of Structural and Equipment Cooling System”
- PDC 46, “Testing of Structural and Equipment Cooling System”
- PDC 73, “Sodium Leakage Detection and Reaction Prevention and Mitigation”
- PDC 77, “Inspection of Intermediate Coolant Boundary”

8.1 Plant Programs

PSAR section 8.0 describes plant programs considered at the CP stage as special treatments for safety-significant structures, systems, and components (SSCs). Programs that apply to safety-related (SR) or non-safety-related with special treatment (NSRST) SSCs are governed under the quality assurance program description (QAPD) that is evaluated in section 8.2 of this SE.

8.1.1 Technical Evaluation

As discussed in RG 1.253, PSAR chapter 8.0 should provide an overview of the plant programs relied upon to support the Licensing Modernization Project (LMP)-based safety analysis, addressing these programs’ purpose, scope, and performance objectives, as well as applicability to SR SSCs, NSRST SSCs, and operations activities. In addition, RG 1.253 states that at the CP stage applicants should include general descriptions in the safety analysis report (SAR) regarding any programs needed to implement special treatments and meet reliability and performance targets for SR SSCs and NSRST SSCs. The staff evaluated the information

provided in the CP application related to plant programs consistent with the guidance in RG 1.253 as well as the requirements in 10 CFR 50.34(a) for the preliminary design to adequately describe how the design bases relate to the PDC.

The staff's evaluation of the human factors engineering program and associated technical information for the facility is in SE section 11.2. The staff's evaluation of the emergency preparedness program and associated technical information for the facility is in SE section 11.3. The staff's evaluation of the post-construction inspection, testing, and analysis program and associated technical information for the facility is in SE chapter 12.

8.1.1.1 Comprehensive Vibration Assessment Program

PSAR table 8.0-1 provides an overview of the comprehensive vibration assessment program (CVAP). The PSAR references RG 1.20 and specifies that the CVAP should be applied to unique or first-of-its-kind designs and includes all reactor internals (except for the core) and SSCs that may transfer vibration to the internals. The applicant states that the results of the CVAP will be documented in a series of technical reports submitted to the U.S. Nuclear Regulatory Commission (NRC). The applicant will use these reports to demonstrate compliance with regulatory requirements and to ensure safe operation of the reactor. The CVAP is comprised of a number of subprograms as summarized in table 8.0-1 of the PSAR.

In PSAR section 7.1.2.2, the applicant states that reactor vessel internal components are classified as prototypes in accordance with RG 1.20 section C.1 and are screened and analyzed for effects due to potential excitation mechanisms in accordance with RG 1.20 section C.2. In PSAR chapter 7, the applicant identifies the systems that will be included in the CVAP to evaluate the effects of flow-induced vibration. In PSAR section 12.5.3, the applicant states that flow-induced system vibration testing is included as part of the power ascension testing.

The staff evaluated the CVAP program description in chapter 8 of the PSAR as well as its proposed application in chapter 7 of the PSAR and compared it to the guidance in RG 1.253 (content of application) and RG 1.20 (CVAP program development). Additionally, the staff conducted an audit that included the supporting documentation for the CVAP program to verify the associated information in the PSAR. Based on this review and supported by confirmation in the audit, the staff determined that the preliminary information related to the CVAP program is consistent with the guidance in RG 1.253 and RG 1.20 as well as the applicable requirements in 10 CFR 50.34(a)(3) for the preliminary design. Therefore, the preliminary CVAP program is acceptable to support the CP application for KU1.

The staff notes that, in table 8.0-1, the applicant references 10 CFR 50.55a(b)(3)(iii)(C), and regulatory guidance in RG 1.20. As published in the Federal Register (FR), the NRC removed 10 CFR 50.55a(b)(3)(iii)(C) from the NRC regulations in August 2024 (89 FR 70449), following receipt of the KU1 CP application. As discussed in the preamble for the final rule, this condition is required by other regulations. As such, leaving the reference to 10 CFR 50.55a(b)(3)(iii)(C) in the PSAR, would not require the applicant to take actions inconsistent with the current regulatory regime.

8.1.1.2 Equipment Qualification Program

PSAR table 8.0-1 states that the equipment qualification (EQ) program provides direction on establishing and performing EQ, and defines how to ensure systems, components, and designs comply with applicable technical requirements, codes, standards, and regulatory requirements.

The EQ program is applied to safety-significant SSCs to ensure performance of design functions independent of safety classification. USO's EQ program contains three qualification disciplines: electrical, mechanical, and instrumentation and controls (I&C) equipment qualification. The EQ program includes development of testing and qualification specifications, processes, procedures, and reports.

PSAR table 8.0-1 references the 10 CFR Part 50 regulations in Appendix B, 10 CFR 50.49, 10 CFR 50.55a(h), and 10 CFR 50.34(a)(3)(i). The staff notes that 10 CFR 50.49 does not apply at the CP stage. However, Appendix B and 50.55a(h) do apply at the CP stage. In addition, 10 CFR 50.34(a)(3) states that the CP application must describe how the design bases relate to the PDC. The EQ program is used to support several KU1 PDC, including PDC 1, 4, 14, 15, and 30.

Mechanical Equipment Qualification

PSAR table 8.0-1 references RG 1.100, which endorses with certain exceptions and clarifications Institute of Electrical and Electronics Engineers (IEEE) 344-2013, American Society of Mechanical Engineers (ASME) QME-1-2017, and IEEE C37.98-2013 for seismic qualification of electrical and active mechanical equipment and the functional qualification of active mechanical equipment for nuclear power plants. The staff evaluated the mechanical EQ program description in the PSAR and compared it to the relevant guidance in RG 1.253 and RG 1.100. Additionally, the staff conducted an audit that included applicant's mechanical EQ program supporting documents to better understand the proposed EQ program for mechanical equipment (ML25302A443).

Based on this review and supported by confirmation in the audit, the staff determined that the preliminary information related to the mechanical EQ program is consistent with the guidance in RG 1.253 and RG 1.100 as well as the applicable requirements in 10 CFR 50.34(a)(3) to describe how the design bases relate to the PDC. Therefore, the preliminary mechanical EQ program is acceptable for the KU1 CP. The staff notes that the NRC has not endorsed the ASME QME-1-2023 standard referenced in the KU1 PSAR. However, the staff reviewed ASME QME-1-2023 and considers it appropriate if implemented consistent with the regulatory positions specified in Rev 4 of RG 1.100.

Electrical Equipment Qualification

PSAR table 8.0-1 references RG 1.89, which endorses, with certain clarifications, IEEE/International Electrotechnical Commission (IEC) 60780-323-2016 for satisfying regulatory requirements for environmental qualification of certain electric equipment important to safety for nuclear power plants. The staff reviewed the electrical EQ program description in chapter 8 of the PSAR and compared that information to relevant guidance in RG 1.253 and RG 1.89. Additionally, the staff conducted an audit that included USO's electrical EQ program documents to better understand the proposed EQ program for electrical equipment (ML25302A443).

Based on this review, as confirmed through audit, the staff determined that the preliminary information related to the electrical EQ is consistent with the guidance in RG 1.253 and RG 1.89 as well as applicable requirements in 10 CFR 50.34(a)(3) to describe how the design bases relate to the PDC and 10 CFR 50.55a(h). Therefore, the preliminary electrical EQ program is acceptable to support the CP application for KU1.

Instrumentation and Controls Equipment Qualification

PSAR table 8.0-1 references RG 1.209, which endorses, with certain enhancements and exceptions, IEEE 323-2003 for satisfying the environmental qualification of SR computer-based instrumentation and control systems for service in mild environments at nuclear power plants. The staff reviewed the I&C EQ program description in chapter 8 of the PSAR and compared that information to relevant guidance in RG 1.253, RG 1.209, and RG 1.180. Additionally, the staff audited the applicants' I&C EQ preliminary documentation to better understand their proposed EQ program for I&C equipment (ML25302A443).

Based on this review, as confirmed through audit, the staff determined that the preliminary information related to the I&C EQ program is consistent with the guidance in RG 1.253, RG 1.209 and RG 1.180 as well as applicable requirements in 10 CFR 50.34(a)(3) to describe how the design bases relate to the PDC and 10 CFR 50.55a(h). Therefore, the preliminary I&C EQ program is acceptable to support the CP application for KU1.

8.1.1.3 Testing Program

PSAR table 8.0-1 indicates that the purpose of the testing program is to provide reasonable assurance that testing demonstrates that SR and NSRST SSCs perform satisfactorily in service. The applicant states that the testing program also identifies the programs and groups responsible for proof tests, preoperational tests, and operational tests needed to demonstrate compliance with the design. The applicant also states that the testing will comply with written test procedures. In PSAR table 8.0-1, USO references the regulations in 10 CFR 50.43(e)(1) and 10 CFR Part 50, Appendix B, and guidance in ASME NQA-1-2015, which is endorsed in RG 1.28, with appropriate regulatory positions.

The staff reviewed the testing program description in chapter 8 of the PSAR and compared that information to relevant guidance in RG 1.253 and RG 1.28. Additionally, the staff conducted an audit that included the applicant's testing program documents (ML25302A443). During the audit, the staff discussed the importance of early development of the testing program to ensure the performance of each safety feature of KU1 is appropriately demonstrated at the OL stage, in accordance with 10 CFR 50.43(e)(1). Based on this review, as confirmed through audit, the staff determined that the preliminary testing program is consistent with the guidance in RG 1.253 and RG 1.28 as well as applicable requirements in 10 CFR Part 50, Appendix B and 10 CFR 50.43(e)(1). Therefore, the preliminary testing program is acceptable to support the CP application for KU1.

8.1.1.4 Reliability and Integrity Management Program

PSAR table 8.0-1 identifies the RIM program as a special treatment program supporting the design and construction phase. The PSAR also identifies the RIM program as supporting PDC 4, 31, 32, 36, 37, and 77 for various safety-significant SSCs. The objective of the RIM program is to ensure that passive components within the RIM program scope achieve an acceptable level of reliability to support the plant probabilistic risk assessment (PRA) throughout the life of the plant. The RIM program involves design interaction, performance monitoring, inspections, tests, maintenance, and replacements, as strategies to ensure the SSCs meet their reliability targets.

The applicant stated that the RIM program will be developed following the 2019 Edition of ASME Boiler and Pressure Vessel Code (BPVC) Section XI, Division 2 (XI-2) and RG 1.246,

which endorses XI-2. The RIM program scope includes applicable SSCs whose failure could adversely affect plant safety and reliability. The RIM program requires conducting a degradation mechanism assessment (DMA) for each component in the scope of the RIM program, determining the reliability targets for the SSCs, and identifying RIM strategies to ensure reliability targets can be met.

8.1.1.4.1 Program Scope and Degradation Mechanism Assessment

The staff reviewed the applicant's preliminary RIM program scope and the criteria used to determine the SSCs to be included in the RIM program as described in section 6 of NAT-13478, "Natrium Demonstration Plant Long-Lived Passive Structural Materials of Construction Selection and Development," (ML25274A130), which was provided by the applicant as a supplement to the CP application. The applicant states that the RIM program scope includes SR and NSRST passive mechanical components. The staff notes that SSCs in the scope of the RIM program contribute to several functions that affect plant safety and reliability, including forming the primary coolant pressure boundary, being credited as part of the functional containment, and being necessary to ensure a flow path for natural circulation in the primary system for emergency core cooling. The preliminary RIM program scope is consistent with the ASME BPVC XI-2 scope requirement that passive SSCs whose failure could adversely affect plant safety and reliability be included in the RIM program. Based on the consistency of the preliminary RIM program scope with ASME BPVC XI-2, the staff finds the preliminary design information relative to RIM program scope and criteria used to determine SSCs to be included in the program acceptable at the CP stage.

The staff also reviewed the preliminary DMA and the screening criteria for the RIM program. Section 7 of NAT-13478 describes the DMA process and the preliminary DMA screening criteria. The applicant states that the DMA process is iterative, with the criteria being updated and expanded upon as testing completes and the design matures. The staff's audit observations regarding the preliminary DMA are included in enclosure 2 of the audit report (ML25302A448). The staff evaluation of the preliminary DMA for SSCs exposed to sodium or cover gas is in section 7.1.2.1.4 of this SE.

8.1.1.4.2 Reliability Targets

The staff reviewed the preliminary process for allocating and confirming reliability targets as described in section 6 of NAT-13478. The applicant stated that reliability targets are set as the mean value of the failure rates associated with the component type and failure mode from the failure data used in the PRA. If the particular SSC failure mode is not included in the PRA data, then the target reliability is set based on a conservative failure effect and the associated failure rates for impacted mitigation functions, or the frequencies for SSCs whose failure causes an initiating event.

The staff recognizes that, given the preliminary nature of the PRA and the plant design, it is reasonable to not establish final reliability targets at the CP stage. Accordingly, though the staff reviewed the reliability target allocation and confirmation process and the preliminary reliability target and provided observations from that review below, the staff did not make any determinations on the acceptability of the reliability target allocation or confirmation process or the preliminary reliability targets. The staff expects the applicant to provide, during the OL stage, the final reliability targets for SSCs that are in the RIM program, including the technical basis on how the reliability targets are derived as well as the associated uncertainties. For additional

evaluation of the reliability and capability targets for safety-significant SSCs, including those that are not passive mechanical SSCs and thus not subject to RIM, see section 6.2.1 of this SE.

During the review of the CP application, the staff noted several areas where additional information will be needed to support review of the reliability targets related to SSCs in the RIM program at the final design stage. The staff provides these observations and feedback for information. Three particular areas are (1) the treatment of passive SSCs that are not typically modeled in a PRA, (2) how uncertainties associated with the reliability targets are considered, and (3) the process for confirming reliability targets. For the first area, ASME BPVC XI-2 RIM-2.4.3(b) indicates that for a component to be in the scope of the RIM program it must be included in the PRA. However, USO identified some SSCs that are not modeled in the PRA but will be included in the RIM program. The staff will review such deviations from the endorsed RIM process for these SSCs and the justification for the alternative approach to identifying reliability targets and RIM strategies for these SSCs to ensure they reliably perform their functions through the life of the plant at the OL stage.

For the second area, ASME BPVC XI-2 RIM-2.4.2(c) states: "The allocation of SSC-level Reliability Targets shall consider the uncertainties inherent in the prediction of SSC reliability." However, the staff notes that the reliability target allocation process and preliminary reliability targets provided did not appear to include consideration of uncertainty. This is particularly important given the level of uncertainty associated with passive SSC performance-based on the limited knowledge and operating experience in sodium fast reactor conditions. The staff will review how the RIM process addresses uncertainty associated with establishing and meeting reliability targets consistent with the requirements of ASME BPVC XI-2 (including RIM-2.4.2(c)) and the LMP guidance at the OL stage.

For the third area, ASME BPVC XI-2 RIM-2.5.2 states: "The impact of each RIM strategy... on the reliability of each SSC in the scope of the RIM Program shall be assessed for comparison against the SSC-level Reliability Targets." ASME BPVC XI-2 RIM-2.5.1(c) similarly indicates that RIM strategies shall be selected to achieve and maintain SSC reliability consistent with their reliability targets. However, the staff noted that the applicant's RIM process did not include any step to assess, demonstrate or confirm that the reliability targets are met by the identified RIM strategies. This aspect of the process to confirm reliability targets consistent with the requirements of ASME BPVC XI-2 and the LMP guidance, is expected to be included at the OL stage and the staff is open to qualitative or semi-quantitative approaches based on engineering judgement and appropriate consideration of uncertainties. The staff will review how the RIM process ensures reliability targets are met consistent with the requirements of ASME BPVC XI-2 and the LMP guidance at the OL stage.

8.1.1.4.3 RIM Strategy Development

The staff reviewed the applicant's proposed methodology to identify preliminary RIM strategies during the design and construction phase as described in section 6 of NAT-13478. The methodology provides prioritization for developing RIM strategies using a graded risk evaluation process utilizing the results of the reliability target allocation and DMA. The RIM risk evaluation includes considerations such as the design margin, component function, the risk and consequences of a postulated failure of the SSC, susceptibility of each applicable degradation mechanism for the SSC, feasibility and efficacy of monitoring and non-destructive examination (MANDE) for detecting degradation, and the power generation impact of the mechanism.

One aspect of the RIM strategy development the staff noted is the RIM strategy significance (RIMSS) categories, in which RIM strategies are binned into four categories. The staff notes that the lowest bin (RIMSS I) identifies that no MANDE would be provided for components least susceptible to a degradation mechanism or with a low probability or consequence of failure. The staff expectation is that any SSC in the scope of RIM that is deemed to have potentially active degradation should be covered by either a primary or expansion (with appropriate justification and connection to the primary RIM strategy) RIM strategy that is demonstrated to achieve and maintain the specified reliability target for that SSC. The scope of SSCs with potentially active degradation mechanisms binned into RIMSS I will be evaluated at the OL stage if this approach to RIM strategy binning is maintained to ensure those SSCs to which no MANDE is assigned are appropriate.

The staff also reviewed the preliminary screening criteria and RIM strategies and made several observations including the need to consider long-term environmental effects on material properties in the design analysis, management of SSCs needed for SR core cooling, the combined effects of multiple degradation mechanisms, and the need for testing to characterize materials performance (see section 7.1.2.1.4 of this SE and audit report (ML25302A443)). The applicant stated that the development of RIM strategies based on uncertainties under ASME BPVC XI-2 RIM-2.6 is a planned future activity and likely to lead to additional MANDE being identified.

The staff will review the final RIM program, including the final RIM strategies covering SSCs that are in the RIM program and technical basis on how the reliability targets are met with appropriate consideration of uncertainties at the OL stage.

8.1.1.4.4 Research and Development Item on Assuring Materials Performance

PSAR chapter 13 and section 12.1 of NAT-13478 describe a research and development (R&D) item under 10 CFR 50.34(a)(8) associated with the development of information needed to provide assurance of adequate structural material performance for safety-significant SSCs included in the RIM program. The identified activities include research to improve understanding of the effects of high-temperature, chemistry exposure, and irradiation on materials; determination of requirements for further materials testing for environmental compatibility and activities to support the RIM program implementation. To provide assurance that the materials will perform as designed and to confirm that the applicant will develop and implement appropriate RIM strategies to address degradation mechanisms in safety-significant SSCs, the staff is proposing two permit conditions in the KU1 CP related to providing updates associated with this R&D item. These permit conditions are discussed in greater detail in chapter 13 of this SE.

Based on the applicant's use of ASME BPVC XI-2, which is endorsed by the staff in RG 1.246 and the staff's review of the preliminary RIM implementation documents and the associated permit conditions, the staff finds the applicant's preliminary RIM program acceptable to support the design and construction stage. The applicant stated it will continue to update and develop the RIM program as the design matures and detailed design information become available.

8.1.1.5 Design Reliability Assurance Program

The applicant included a design reliability assurance program (D-RAP) according to section 4.4.5 of NEI 18-04, "Risk-Informed Performance-Based Technology-Inclusive Guidance

for Non-Light Water Reactor Licensing Basis Development,” Rev.1 (ML19241A472), which states:

All safety-significant SSCs, including those in the SR and NSRST categories, should be included in a Reliability Assurance Program (RAP) similar to that described in SRP [Standard Review Plan] 17.4. The reliability and availability targets established in the RAP are used to focus the selection of special treatments that are necessary and sufficient to achieve these targets and to assure they are maintained for the life of the plant.

The objective of the D-RAP, as described in section 17.4, “Reliability Assurance Program,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” (ML13296A435) and interim staff guidance DC/COL-ISG-18 “Reliability Assurance Program” (ML103010113), is to ensure that the plant is designed and constructed in a manner consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability) derived from probabilistic, deterministic, and other analyses used to identify and quantify risk. This objective is achieved through implementation of the essential elements of the D-RAP, including organization, design control, procedures and instructions, records, and corrective actions during design and construction activities.

As stated in NUREG-0800, the purpose of the RAP is to provide reasonable assurance that:

- The plant is designed, constructed, and operated consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability);
- RAP SSCs do not degrade to an unacceptable level of reliability, availability, or condition during plant operations;
- Transients that challenge RAP SSCs are minimized; and
- RAP SSCs will function reliably when challenged.

The applicant confirmed during the audit (ML25302A443) that the quality assurance (QA) programs (e.g., design, procurement, fabrication, construction, inspection, and testing activities) will provide control over activities affecting the quality of the RAP SSCs. QA controls for SR SSCs are established under 10 CFR Part 50, Appendix B.

As noted above, NEI 18-04 states that the D-RAP should include SR and NSRST SSCs. The KU1 D-RAP includes SR and NSRST SSCs, consistent with NEI 18-04. To identify SSCs included in the scope of D-RAP, the KU1 D-RAP uses the absolute risk metric approach of NEI 18-04, without deviation. For SSCs identified as safety-significant (i.e., SR or NSRST) based on defense-in-depth (DID) or deterministic considerations and not modeled in the PRA, the reliability target is a conservative value established by the USO system designer.

The staff notes that availability targets cannot be finalized until the design is completed and testing, maintenance, and inspection requirements are identified. Therefore, the staff will review USO’s calculations and incorporation of anticipated out-of-service times of SSCs into the availability targets at the OL stage.

Based on the staff's review of the applicant's D-RAP description, methodology, and QA implementation, as confirmed through an audit of the supporting documentation, the staff determined that the preliminary information regarding the KU1 D-RAP is appropriate at the CP stage because it assures the design and construction of RAP SSCs will be consistent with the key assumptions and risk insights derived from the PRA and other supporting analyses. Because the applicant stated in section 6.2 of the PSAR that the D-RAP will confirm at the OL stage that the reliability performance targets assumed in the PRA have been met, the staff expects the applicant to continue implementing and updating the KU1 D-RAP as the design matures during the construction phase, using the RAP expert panel to enhance the RAP process and SSCs.

8.1.1.6 In-service Testing Program

PSAR table 8.0-1 indicates that the in-service testing program (IST) provides the requirements for IST activities for SR and NSRST pumps, valves, and dynamic restraints to assess their operational ability to perform their specified functions. PSAR table 8.0-1 states that the IST program describes baseline testing, IST activities, examination, and monitoring needed to provide reasonable assurance of the operational readiness of the components. In PSAR table 8.0-1, USO references the regulations in 10 CFR 50.34(b)(6)(iv), and guidance in the draft ASME OM-2 Code.²⁶ The staff notes that 10 CFR 50.34(b)(6)(iv) does not apply at the CP stage. However, 10 CFR 50.34(a)(3) states that the CP application must describe how the design bases relate to the PDC. There are several KU1 PDC that involve in-service testing supported by the IST program.

The staff reviewed the PSAR description of the IST program. Additionally, the staff audited the Natrium IST program documents (see audit report ML25302A443) to verify the description of the IST program in the PSAR. Based on the staff's review and supported by confirmation in the audit, the staff determined the preliminary information on the IST program is consistent with the requirements in 10 CFR 50.34(a)(3) to describe how the design bases relate to the PDC. Therefore, the preliminary information related to the IST program is acceptable to support the CP application for KU1.

8.1.1.7 Leakage Program

PSAR table 8.0-2 identifies a leakage program as a special treatment program to be provided at the OL stage to address 10 CFR 50.34(f)(2)(xxvi). The staff notes that 10 CFR 50.34(f)(2) states a CP application "shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage." Design provisions for monitoring leakage from SSCs are addressed for each SSC as appropriate in chapter 7 of the PSAR and evaluated by the staff in chapter 7 of this SE. PSAR section 8.2.1.8 provides additional detail on leakage detection equipment to inform plant operators in the event of a leak, which is evaluated in SE section 8.3.1.3. Based on the identification of a leakage program to be provided at the OL stage and the description of leakage detection equipment, the staff determined that the applicant described how the design will satisfy 10 CFR 50.34(f)(2)(xxvi) to an extent that is reasonable for a PSAR.

²⁶ The ASME OM-2 code was in a draft form at the time of CP submittal. OM-2 has since been finalized as OM-2-2024 and is being considered for endorsement by the staff.

8.1.2 Conclusion

Based on the review and evaluations described above, the staff concludes that the information in PSAR section 8.0 regarding preliminary plant programs is consistent with the applicable guidance and, therefore, meets regulatory requirements identified in this chapter for the issuance of a CP. Further information on plant programs can reasonably be left for later consideration at the OL stage since this information is not necessary to be provided as part of a CP application.

8.2 Quality Assurance

Section 8.1²⁷ of the PSAR provides the QAPD to be applied to the design, fabrication, construction, and testing of the safety-significant (which includes SR and NSRST) SSCs of the facility. PSAR section 8.1 states that the QAPD is submitted to the NRC as the TerraPower, LLC (TerraPower), QA topical report (TR), TP-QA-PD-0001, "TerraPower QA Program Description," Rev. 14-A²⁸ (ML23213A199), and is incorporated by reference as shown in PSAR table 1.4-2.

The applicable regulatory requirements for the evaluation of QAPD are as follows:

- 10 CFR 50.34(a)(7);
- 10 CFR 50.35;
- 10 CFR 50.40;
- 10 CFR 50.55(f); and
- Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".

Definitions of the PDC are included in PSAR section 5.3. The staff identified PDC 1 as applicable to QA in PSAR section 5.3.

8.2.1 Technical Evaluation

TP-QA-PD-0001 addresses the activities associated with the design and construction of TerraPower's advanced nuclear reactor (Natrium), and describes the policies, processes, and controls that satisfy the requirements established in Appendix B to 10 CFR Part 50. As described in the staff's SE for TP-QA-PD-0001, the staff determined that the TR adequately describes the quality assurance program (QAP) for a CP application (subject to specified

²⁷ Subsequent to the submittal of Rev. 1 of the PSAR, USO made additional modifications to section 8.1 and 11.1 that supersede the sections 8.1 and 11.1 of PSAR Rev.1. This modification is found in ML25323A489.

²⁸ Rev. 14-A of TP-QA-PD-0001 is an approved update to Rev. 12-A of TP-QA-PD-0001 (ML22110A233). The staff's review of Rev. 14-A was limited to the changes from Rev. 12-A. Therefore, the staff's findings during its review of Rev. 12-A are applicable to USO's incorporation by reference of Rev. 14-A of TP-QA-PD-0001.

limitations and conditions) and that the QAP complies with the applicable regulations and can be used for the design, fabrication, construction, and testing activities associated with a nuclear power reactor. As mentioned above, USO incorporated TP-QA-PD-0001 by reference to satisfy the regulatory requirements for the issuance of a CP. While incorporation of an approved TR by reference is one acceptable way for an applicant to establish a QAP and meet the regulatory requirements for Appendix B to 10 CFR Part 50, the SE approving TP-QA-PD-0001 approves the TR for use by TerraPower. Although USO is a wholly owned subsidiary of TerraPower, it is a distinct legal entity. Therefore, the staff evaluated USO's ability to implement the QAPD as described in TP-QA-PD-0001 to meet the regulatory requirements for the issuance of a CP.

The staff reviewed TP-QA-PD-0001 to identify the aspects of the QAPD which can be incorporated by USO without additional evaluation. The staff determined that section 3 through section 20 of TP-QA-PD-0001 are able to be incorporated by USO without additional evaluation. Additionally, the staff confirmed during the general audit that USO will adopt all commitments, policies, procedures, and guidance outlined in TP-QA-PD-0001. Sections 3 through 20 of TP-QA-PD-0001 correlate the requirements of Criterion III through Criterion XVIII of Appendix B to 10 CFR Part 50. The staff determined that USO's incorporation by reference of TP-QA-PD-0001 complies with Criterion III through Criterion XVIII of Appendix B to 10 CFR Part 50 should the applicant adequately address the limitations and conditions included in the SE for TP-QA-PD-0001. USO provided justifications to address the limitations and conditions in section 8.1.1 of the PSAR, which the staff evaluated below in SE section 8.2.1.3. Furthermore, the staff identified Criterion I and Criterion II of 10 CFR Part 50 Appendix B as requiring additional evaluation to determine if USO satisfied these criteria through the incorporation by reference of TP-QA-PD-0001 sections 1 and 2 subject to the limitations and conditions of the SE for TP-QA-PD-0001.

8.2.1.1 Criterion I "Organization"

The primary consideration in determining USO's ability to implement the QAP as described in TP-QA-PD-0001 is the organizational structure of USO and how it relates to TerraPower as USO's direct and controlling owner. The staff reviewed USO's corporate structure, as described in PSAR section 11.1, to ensure that it follows the corporate structure described in TP-QA-PD-0001 and meets the requirements of Criterion 1 of Appendix B to Part 50. Additional evaluation on the staff's analysis regarding USO's organization is in section 11.1 of this SE.

PSAR section 8.1.2 states that "the USO Quality Execution Plan (QEP) governs the implementation of the TerraPower QAPD for the Kemmerer Unit 1 project." During the general audit, the USO QEP was provided for audit, which confirmed that high level controls are documented for the design, construction, and initial testing of KU1. The staff notes that implementation of the USO QEP is planned to be verified during a future QA implementation inspection.

PSAR section 11.1.1.1 states that "[t]he President and Chief Executive Officer of USO has overall responsibility for functions involving design, construction, and operation. At present, the President and CEO of USO is the same individual as the President and CEO of TerraPower. In their role as the President and CEO of USO, they are responsible for establishment and execution of quality assurance program requirements and ensuring information and resources are shared between the nuclear-oriented part of the organization and the rest of the corporate organization in a manner that prioritizes nuclear safety and security." Additionally, PSAR section 11.1 lists the following as USO positions, which at present are held by individuals serving in equivalent roles for TerraPower:

- Executive Responsible for Project Execution
- Executive Responsible for Regulatory Affairs
- Director Responsible for Quality
- Director of Plant Delivery

The staff compared these positions, considering their descriptions in PSAR section 11.1, to those described in section 1 of TP-QA-PD-0001 and determined that the USO organization is consistent with the organizational description in the TP-QA-PD-0001, given the dual nature of the key positions outlined in the TR. Specifically:

- USO's CEO and President provides top-level leadership and is responsible for the implementation and execution of the QAP and its subordinate documents.
- USO's Executive Responsible for Project Execution is responsible for executing parts of the QAP and, per the May 2, 2024, letter (ML24123A242), to provide guidance and direction such that for the SR activities, including engineering, construction, operations, maintenance, procurement, and planning during the design and construction phase are, performed following the guidelines of the QAP.
- USO's Executive Responsible for Regulatory Affairs is responsible for the records management and document control activities to meet the requirements of the QAP.
- USO's Director of Quality will have the primary responsibility for verifying that the QAP is in place and effective.

In addition to ensuring USO has a corporate structure consistent with TP-QA-PD-0001, the staff reviewed USO's corporate structure to ensure that it complies with the requirements of Criterion I of Appendix B to 10 CFR Part 50.

Criterion I of Appendix B to 10 CFR Part 50 requires, in part, that,

[t]he applicant shall be responsible for the establishment and execution of the quality assurance program. The applicant may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part thereof, but shall retain responsibility for the quality assurance program.

[t]he persons and organizations performing quality assurance functions shall report to a management level so that the required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided.

Criterion I clarifies that the quality assurance functions are,

- (1) assuring that an appropriate quality assurance program is established and effectively executed; and (2) verifying, such as by checking, auditing, and inspecting, that activities affecting the safety-related functions have been correctly performed.

The staff notes that currently the Director Responsible for Quality is responsible for both the implementation of the QAPD for TerraPower and USO. Although the letter dated May 2, 2024, states USO will retain responsibility for the execution of the QAP, the staff determined that a permit condition was necessary to ensure sufficient independence between USO and TerraPower to support meeting the requirements of Criterion I of Appendix B:

Prior to the start of construction, as defined by 10 CFR 50.10, USO shall establish and document within the PSAR a dedicated position responsible for overseeing the execution of the Quality Assurance Program Description (QAPD). This position shall have full authority to make decisions related to QAPD implementation, independent of cost or schedule considerations, including responsibility for overseeing and ensuring the implementation of audit findings. This position shall not be assigned responsibilities for implementing the quality assurance programs of other organizations supporting the design and construction of KU1.

With this condition along with the key positions necessary for implementation of the QAPD program established within USO, as described in PSAR section 11.1, the staff has reasonable assurance that the individual and organization responsible for QA satisfies Criterion I of Appendix B to 10 CFR Part 50.

Based on review of the organizational structure, key management position functions within USO, and implementation of QAPD for both TerraPower and USO, the staff finds that the applicant is responsible for the execution of the QAP. Therefore, the staff finds that incorporation by reference of section 1 of TP-QA-PD-0001 satisfies Criterion I of Appendix B to 10 CFR Part 50.

8.2.1.2 Criterion II “Quality Assurance Program”

The staff evaluated USO’s incorporation by reference of section 2 of TP-QA-PD-0001 in order to address Criterion II of 10 CFR 50 Appendix B. Specifically, the staff evaluated compliance with the requirement of Criterion II of Appendix B to 10 CFR Part 50 that, “[t]he applicant shall regularly review the status and adequacy of the quality assurance program. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing.”

PSAR section 11.1.1.5, states that, “[t]he director responsible for quality is responsible for ensuring quality audits are performed in accordance with the QAPD as well as ensuring TerraPower performs functions safely and in accordance with environmental regulations.” In addition, PSAR section 8.1.2 states that the USO QEP, “governs the implementation of the TerraPower QAPD for the Kemmerer Unit 1 project, including audits of the execution of the QAPD.” As noted above, the staff will verify implementation of the QEP during a future QA inspection. The staff finds that USO has adequately described how the effectiveness of the QAPD will be evaluated and how management of other organizations participating in the QAP is conducted. Therefore, the staff finds that incorporation by reference of section 2 of TP-QA-PD-0001 satisfies Criterion II of Appendix B to 10 CFR Part 50.

8.2.1.3 Evaluation of TP-QA-PD-0001 Limitations

Section 8.1.1 of the PSAR addresses the two limitations on the usage of the TP-QA-PD-0001. The staff evaluated the limitations and conditions identified in the TP-QA-PD-0001 to confirm

the limitations and conditions were met in the PSAR. The staff's SE for Rev. 12-A of TP-QA-PD-0001 imposed two limitations and conditions.

Limitation 1 states:

TerraPower did not include a commitment to conform with NRC's GL 85-06 and Regulatory Position 3.5 in RG 1.155 in the QAPD. The NRC staff acknowledges that with the substantial differences between TerraPower's plant design and a light water reactor design, a direct commitment to these quality guidance positions may not be practical. However, TerraPower will have to justify in its application why these two quality guidance positions are not applicable to its advanced reactor design.

Limitation 2 states:

TerraPower stated it did not include a commitment to conform with the applicable regulatory guidance in the following RGs: 1.26, 1.29, 1.54, 1.164, 1.189, and 1.231. The NRC staff acknowledges that with the substantial differences between TerraPower's plant design and a light water reactor design, a direct commitment to these RGs may not be appropriate at this time. However, TerraPower will have to address in its application its conformance and/or exceptions to the applicable regulatory position guidance provided in the RGs.

The staff's evaluation of these limitations with respect to the QAPD's incorporation by reference into the KU1 CP application is provided in the following sections.

GL 85-06 and Regulatory Guide 1.155

PSAR section 8.1.1 states KU1 will not conform to GL 85-06, "Quality Assurance Guidance for ATWS Equipment that is Not Safety-Related," (ML031140390), and RG 1.155, "Station Blackout," (ML003740034), because GL 85-06 and RG 1.155 are methods to satisfy the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." and 10 CFR 50.63, "Loss of all alternating current power," which are not applicable regulations to non-LWR designs. PSAR section 8.1.1 states risk of loss of offsite power will be reduced by applying the methodologies described in NEI 18-04 as endorsed by RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Rev. 0 (ML20091L698). The staff finds the approach acceptable because GL 85-06 and RG 1.155 are not applicable to non-LWR designs and because KU1 will follow an endorsed methodology used to reduce risk of loss of offsite power.

Regulatory Guide 1.26

PSAR section 8.1.1 states that KU1 does not conform to RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste Containing Components of Nuclear Power Plants," Rev. 6 (ML21232A142). PSAR section 8.1.1 states that KU1 conforms to NEI 18-04 as endorsed by RG 1.233 for determining the plant design SSC classifications. The staff evaluated KU1 SSC classifications in SE chapter 5 and determined that conformance to NEI 18-04, as endorsed by RG 1.233, is acceptable.

Regulatory Guide 1.29

PSAR section 8.1.1 states that KU1 partially conforms to RG 1.29, “Seismic Design Classification for Nuclear Power Plants,” Rev. 6 (ML21155A003). PSAR section 8.1.1 states that KU1 conforms to RG 1.29 with the exception that the requirements for “seismic category I” SSCs are applied to SR SSCs instead. The staff finds this acceptable because seismic category 1 is applied to the most safety-significant SSCs in a currently operating plant, but “seismic category 1” does not exist under the LMP framework, so an equivalent grouping in LMP of the most safety-significant SSCs is SR SSCs. A more detailed evaluation of the applicant’s seismic categorization framework is provided in section 6.4.1.1.1 of this SE.

Regulatory Guide 1.54

PSAR section 8.1.1 states that KU1 design does not use service level I, II, or III protective coatings, therefore RG 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” Rev. 3 (ML17031A288), is not applicable. In addition, PSAR section 8.1.1 states that KU1 will identify and conform to regulatory positions in RG 1.54 if service level I, II, or III protective coatings are used in the design. The staff finds this approach acceptable because service level I, II, or III protective coatings are not used in the KU1 design. If service level I, II, or III protective coatings are used in the design, the staff will confirm USO’s conformance to the regulatory positions in RG 1.54 during a future licensing submittal.

Regulatory Guide 1.164

PSAR section 8.1.1 states that KU1 conforms to RG 1.164, “Dedication of Commercial-Grade Items for Use in Nuclear Power Plants,” Rev. 0 (ML17041A206), as shown in PSAR table 1.4-1. The staff finds conformance with RG 1.164 acceptable.

Regulatory Guide 1.189

PSAR section 8.1.1 states that the nuclear island fire protection system of KU1 conforms to RG 1.189 as shown in PSAR table 1.4-1. The staff finds conformance with RG 1.189 acceptable. Additional evaluation of the applicant’s fire protection program is provided in sections 7.5.2 and 8.3 of this SE. The staff notes that RG 1.189 provides an acceptable method to address 10 CFR 50.48(a) which is applicable at the OL stage, therefore conformance to RG 1.189 is not required at the CP stage.

Regulatory Guide 1.231

PSAR section 8.1.1 states that KU1 conforms to RG 1.231, “Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants,” Rev. 0 (ML16126A183) as shown in PSAR table 1.4-1. The staff finds conformance with RG 1.231 acceptable.

8.2.2 Conclusion

Based on the staff findings above, the staff determined that the TP-QA-PD-0001, as incorporated by USO, is sufficient and complies with the requirements of 10 CFR 50.34(a)(7) and Appendix B to 10 CFR Part 50. In addition, the staff finds that USO’s implementation of TP-QA-PD-0001 conforms with PDC 1, which requires, in part, that a QAP shall be established

and implemented in order to provide adequate assurance that safety-significant SSCs will satisfactorily perform their safety functions.

The staff notes that a holder of a CP is subject to the terms and conditions of 10 CFR 50.55, including 10 CFR 50.55(f) which requires, in part, that

[E]ach construction permit holder described in paragraph (f)(1) of this section may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report, provided the change does not reduce the commitments in the program description previously accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to NRC within 90 days. Changes to the quality assurance program description that do reduce the commitments must be submitted to NRC and receive NRC approval before implementation...

The staff has concluded that regulatory requirements and license commitments for QA will be adequately included in the design, procurement, and construction of the KU1. In addition, the staff has concluded that changes to the organizational structure between USO and TerraPower will be documented adequately and submitted to the NRC, as appropriate.

8.3 Fire Protection

PSAR section 8.2 describes general fire protection design considerations, sodium leakage concepts, fire analyses, and the fire protection program. This section also states that the design maintains adequate fire protection DID in all fire areas and takes into account prevention, detection, containment, suppression, and the ability to maintain safe shutdown in accordance with 10 CFR 50.48(a), PDC 3, and RG 1.189. PSAR section 8.2 further states that fire protection requirements are considered in the design of the Natrium reactor plant for both conventional and sodium fires and that the conventional fire protection design uses approved guidance, methodologies, and codes and standards. For sodium fire protection, the KU1 design uses sodium fast reactor industry information to develop Natrium-specific guidance, requirements, and criteria.

8.3.1 Technical Evaluation

As discussed in RG 1.189, the CP application should provide assurance of that the fire protection systems will meet the relevant requirements in 10 CFR 50.48(a).

8.3.1.1 Cells and Guard Vessels

PSAR section 8.2.1 discusses various methods of limiting sodium interactions with air and concrete including seals and inert gases, cells, guard pipes, guard enclosures, clamshells, catch pans, inerted guard vessels, leakage detection, materials, and fire areas and barriers. The applicant states that seals and inert gases are used as part of individual sodium-containing system design and that inerting is used in areas or cells containing radiological sodium to mitigate sodium-air reactions and the airborne release of radionuclides. In addition, cells are rooms with a limited leak rate and known oxygen volume which is either actively inerted or becomes de facto inerted upon consumption of the limited available oxygen after a sodium leak. Finally, PSAR section 8.2.1.7 describes how inerted guard vessels are used to prevent sodium-air and sodium-concrete reactions in the event a leak from another vessel containing

sodium (e.g., the reactor vessel-guard annulus) and that leak detection is provided within the inerted space.

The staff reviewed the information provided with respect to inerted or limited oxygen spaces, such as cells and guard vessels, and notes that these are generally effective means of preventing or mitigating sodium-air reactions and are reasonable to use in various areas of the plant. Based on the effectiveness of well-designed and monitored cells and guard vessels, the staff finds that the preliminary design of cells and guard vessels is consistent with PDC 3 and 73.

At the preliminary design stage, there are limited details of how these SSCs will be designed to perform their function to prevent or mitigate sodium fire. Given that sodium fire hazards are highly dependent on the final design configuration and a fire PRA will be performed as part of the OL application, further information can reasonably be left for later consideration. In the OL review, the staff will evaluate the final design of these SSCs and their ability to perform their functions reliably in the event of postulated sodium leakage.

8.3.1.2 Guard Pipes and Enclosures

The applicant described guard pipes, which are currently applied for systems containing radiological sodium and act as a full secondary pressure boundary in the event of a leak designed to the same conditions as the protected system pipe. In addition, guard pipes may be inerted or not inerted. If not inerted, the conditions within the annulus are considered for the design rating of the guard pipe. Guard enclosures are similar to guard pipes, with the addition of a drainage to an acceptable location such as a drain tank cell via sodium leak detection, Collection, and Containment System (NNA) drain line or within the process cell to catch pans. For cells which have an inerted environment, the catch pans will not have a suppression deck. For cells without an inerted environment, the cell will either be evaluated for combustion or a suppression deck will be included. Guard enclosures are qualified to withstand design basis sodium leaks and are applied to high stress piping, nozzles, and components in locations where sodium leak interactions with safety-significant components cannot be tolerated, such as the Head Access Area (HAA).

The applicant further describes a clamshell as a form of guard enclosure placed specifically where piping inspections are expected to be needed. These will be used for Intermediate Heat Transport System (IHT) SSCs located within the HAA and form the fire area and boundary between the Reactor Auxiliary Building (RAB) pipe chase room and the HAA. The applicant states that this feature of the clamshell enclosures to allow leaked sodium to flow into the RAB pipe chase helps ensure pressurized sodium in the clamshell cannot spray into the HAA. PSAR section 7.5.2.3.4 states that first-of-a-kind aspects of the clamshell design will be tested to ensure it can perform its required function with testing methods and acceptance criteria to be identified in the OL application. Pre-service and in-service testing as required by the RIM program will be performed. During audit discussions, the applicant indicated that testing will be performed on the clamshell enclosures to ensure they can maintain effectiveness when exposed to design temperatures and pressures and is expected to include leak-tight testing and pressure testing after installation.

The staff reviewed the information provided with respect to guard pipes and enclosures, including clamshell enclosures. The staff notes that guard pipes and enclosures are generally an effective means of preventing or mitigating sodium-air reactions and are reasonable to use in various areas of the plant based on the risk of sodium leakage and fire. In particular, the staff

notes that the proposed use of clamshell enclosures to prevent sodium leakage from IHT piping from entering the HAA and causing a sodium fire in a critical area of the plant is very important. The staff notes that the applicant plans to perform testing of the clamshell design as well as leak-tightness and pressure testing of these SSCs after installation. Based on the information noted above, the staff finds that the preliminary design of guard pipes and enclosures, including clamshell enclosures, is consistent with PDC 3 and 73.

At the preliminary design stage, there are limited details of how these SSCs will be designed to perform their function to prevent or mitigate sodium fire. Given that sodium fire hazards are highly dependent on the final design configuration and a fire PRA will be performed as part of the OL application, further information can reasonably be left for later consideration. At the final design stage in the OL review, the staff will evaluate the final design of the guard pipes and enclosures, including the clamshell enclosures, to ensure their ability to perform their functions reliably in the event of postulated sodium leakage (which is discussed in SE sections 8.3.1.5 and 8.3.1.6 below).

8.3.1.3 Catch Pans, Leakage Detection, and Materials

PSAR section 8.2.1.6 states that catch pans are steel-lined volumes designed to retain the design basis sodium leak volume plus margin and are used as drain locations that collect leaked sodium from guard enclosures or sloped floors. Catch pans are also placed in areas where operational leakage may occur (e.g., a sodium-containing heat exchanger). The applicant also states that suppression decks are applied to catch pans where the environmental oxygen volume cannot be limited.

PSAR section 8.2.1.8 states that leakage detection equipment for liquid sodium, sodium aerosols, and reaction products, is provided to inform plant operators in the event of a leak and that leakage detectors conform to PDC 73 and the applicable guidance in RG 1.189. The PSAR further clarifies that leak detection is provided at multiple points between pipe supports along all normally sodium-containing systems (i.e., drain lines excepted), within either the guard pipe, guard enclosure, or pipe insulation and that the leak detectors are capable of detecting leaks within the entire length of pipe it covers. In addition, conventional fire detection is installed in non-inerted areas to provide supplemental sodium leak detection.

PSAR section 8.2.1.9 states that the design of SSCs containing sodium considers the use of materials compatible with sodium to minimize the effects of sodium interactions, including using insulation that has low chemical reactivity with sodium. The applicant also states that in consideration of conventional fires, less combustible materials (e.g. Spectrasyn 10 for Intermediate Sodium Pump (ISP) and Primary Sodium Pump (PSP) lubrication and cooling fluid due to its high flash point) are chosen in the design of SSCs and that additional information regarding the materials being used for the final design will be further described at the OL stage.

The staff reviewed the information provided with respect to catch pans, leakage detection, and materials. The staff notes that catch pans and suppression decks have been used in past sodium reactor designs to mitigate and manage sodium-air reactions but require careful consideration of sizing and design to address heat loads from leaked sodium and the ensuing sodium-air reaction. The staff notes that the use of less combustible materials is a prudent approach to reduce the hazard from conventional and sodium fires. In addition, the staff notes the proposed approach to leakage detection for guard vessels, pipes, enclosures, and pipe insulation is thorough and important to enable early detection and response to sodium leaks.

Based on the information noted above, the staff finds that the preliminary design of catch pans, leakage detection, and materials is consistent with PDC 3 and 73.

At the preliminary design stage, there are limited details of how these SSCs will be designed to perform their function to prevent or mitigate sodium fire. Given that sodium fire hazards are highly dependent on the final design configuration and a fire PRA will be performed as part of the OL application, further information can reasonably be left for later consideration. At the final design stage in the OL review, the staff will evaluate the final design of the catch pans, leakage detection, and materials to ensure their ability to perform their functions reliably in the event of postulated sodium leakage.

8.3.1.4 Fire Areas and Barriers

PSAR section 8.2.1.10 states that fire areas are established and that redundant trains of safety-significant and safe shutdown SSCs are located in separate fire areas per the guidance from RG 1.189 section C.8.2. Separation by physical barriers and selective positioning of safety-significant SSCs ensure a fire is contained within a single fire area and a safe shutdown path is maintained regardless of the location of fire initiation. The applicant further states that fire barriers are designed to withstand the most restrictive hazard conditions (temperatures, pressures, etc.) postulated in the established fire areas with fire scenarios determined and fire area separation design requirements for each fire area validated through modeling and analysis. Fire modeling will be used to run sensitivity analyses to determine which leak fire scenarios create the most stringent design conditions for each area and ensure the mitigation features provide adequate DID for the hazards in that fire area.

The staff reviewed the information provided with respect to fire areas and barriers and notes the planned use of RG 1.189 to establish appropriate separation by physical barriers and physical separation ensure a safe shutdown path is maintained. The staff notes that fire scenarios will be modeled and analyzed with sensitivity analyses to validate fire area separation design requirements are met for each fire area. Based on conformance with RG 1.189 and the information noted above, the staff finds that the preliminary design information related to fire areas and barriers is consistent with PDC 3 and 73.

At the preliminary design stage, there are limited details of how the plant and specific fire protection SSCs will be designed to perform their function to prevent or mitigate sodium fire. Given that fire hazards are highly dependent on the final design configuration and a fire PRA will be performed as part of the OL application, further information can reasonably be left for later consideration. At the final design stage in the OL review, the staff will evaluate the final design of the fire areas and barriers to ensure their ability to perform their functions reliably in the event of postulated sodium leakage.

8.3.1.5 Sodium Leakage Concepts

PSAR section 8.2.2 discusses sodium leakage concepts and design provisions for sodium fire protection including design basis leakage into a guard enclosure, design basis leakage within a process cell, beyond design basis leakage, and operational leakage. PSAR section 8.2.2.2 states that design basis sodium leakage within a guard enclosure is designed to be contained within the enclosure and then directed into a catch pan through a drain line. The PSAR states that sodium-air reactions may occur within guard enclosures as they are not inerted but the guard enclosure will be designed based on analyses of potential reactions within the enclosure.

In addition, the drain lines will be heat traced, where needed, to minimize sodium freezing and sized to accommodate the design basis leak rate.

PSAR section 8.2.2.3 describes design basis leakage within process cell and states that each cell is designed to contain or drain design basis leakage through sloped floors and steel plates to a catch pan. For process cells that have an inerted environment, the catch pan does not include a suppression deck. For cells without an inerted environment, the cell is either evaluated for sodium-air reactions based on available oxygen, or a suppression deck is included. The staff notes that PSAR section 8.2.1.6 states that this type of catch pan will include suppression decks where the environmental oxygen volume cannot be limited.

PSAR section 8.2.2.4 describes beyond design basis leakage. Leaks postulated during beyond design basis events will be evaluated for acceptable consequences with respect to the Quantitative Health Objectives as part of the integrated risk assessment performed as described in chapter 3 of the PSAR, considering worst case, maximum leak sizes as applicable. The PSAR states that additional leak mitigation features may be added to address unacceptable consequences.

The staff reviewed the information provided by the applicant with respect to sodium leakage concepts. At a conceptual level, the staff notes that the approach to addressing sodium leakage within guard enclosures and process cells appears reasonable. However, this conceptual approach is very preliminary with many details of how the design, modeling, and analysis of sodium leakage and the associated fire hazard to be determined in the OL application. Similarly, the staff notes the limited description of how design basis and beyond design basis leakage from sodium-containing SSCs (piping or vessels) will be established and analyzed, which will also need to be justified at the OL stage.

PSAR section 8.2.2.1 states that catch pans are provided for operational leakage only for asset protection and ease of cleanup. Since this concept and these features are not credited for safety purposes, the staff did not evaluate this information further. The staff notes that the description of operational leakage identifies it as being assumed to be less than the design basis leak rate. Therefore, the staff focused its review on design basis and beyond design basis leakage and the features credited to detect, prevent, and mitigate design basis and beyond design basis leakage.

8.3.1.6 Sodium Leak Postulation

PSAR section 8.2.2 states that leaks are postulated for all piping that normally contains sodium. PSAR section 8.2.3.3 further describes the approach to sodium leak postulation. Postulated design basis leakage crack sizes from piping areas subject to high stress are established as a $Dt/4$ (diameter multiplied by thickness divided by 4) crack for seismically qualified piping and a full-circumferential break for non-seismically qualified piping. This approach is adapted from the guidance found in Branch Technical Positions (BTPs) 3-3 (ML070800027) and 3-4 (ML16085A315) from NUREG-0800 for moderate energy piping for light water reactors. The PSAR further states that design basis leak locations are determined using additional criteria for high-temperature creep considerations from ASME BPVC Section III Division 5 in addition to the guidance in BTP 3-4.

PSAR section 8.2.3.3 also describes the approach taken to postulating leakage in areas identified as lower stress. The PSAR identifies two exceptions based on stress criteria derived from BTP 3-4 with a third exception tied to not assuming leakage from sealed guard piping

simultaneously with leakage from the system piping. PSAR section 8.2.3.3 indicates that lower stress areas meeting any of the three exceptions are excluded from considering a $Dt/4$ or full-circumferential break as their design basis leakage size. However, design basis leakage cracks will be postulated in these lower stress locations with the postulated crack sizes developed using a methodology to be provided at the OL stage.

The staff reviewed the information provided by the applicant with respect to sodium leakage and sodium leak postulation. The staff notes that postulating leaks for all piping that normally contains sodium is a reasonable approach. However, the staff also notes that the PSAR does not describe an approach to postulation of sodium leaks from vessels. The specific justification for sodium leak postulation will need to be provided at the OL stage; alternatively, without an accepted approach, the staff expects leaks to be postulated from any sodium-containing SSC with a single barrier, including vessels.

The staff notes the use of a graded approach to sodium leak postulation for design basis and beyond design basis leakage based on the probability of leakage is generally reasonable. However, the staff also observe that BTP 3-3 and 3-4 are part of NUREG-0800 guidance that is specific to LWRs. In particular, BTP 3-3 and 3-4 are focused on large leaks and pipe ruptures in LWRs creating dynamic effects that could damage other SSCs. However, in a sodium system even relatively small leaks can create a significant fire hazard that needs to be managed effectively. Therefore, the staff's general observation with applying BTP 3-3 and 3-4 to KU1 is that the criteria and basis of leak postulation are for a fundamentally different purpose than in LWRs and additional justification will need to be provided at the OL stage. In addition, there are significant differences between KU1 and LWRs, such as operating pressure and temperature, coolant type and potential degradation modes that must be considered in applying these LWR criteria to a sodium reactor design. Therefore, the staff observes these conceptual approaches to be a reasonable way to analyze and address sodium leakage which is acceptable at the CP stage, but the specific criteria and basis have not yet been developed and justified for the staff to make a finding on the use of BTP 3-3 and 3-4, adapted to the Natrium design at this time.

Further, the staff notes that applicant proposes to develop a methodology to postulate design basis leakage from lower stress areas of piping. This methodology will need to be developed with appropriate technical basis and justified for how it is applied to the KU1 design.

Based on the discussion above, the staff determined that the preliminary approach related to sodium leak postulation to be consistent with PDC 3 and 73. However, at the preliminary design stage, there are limited details of how sodium leakage will be managed and analyzed to ensure SSCs will be designed appropriately to manage the sodium fire hazard. Given that fire hazards are highly dependent on the final design details and further information related to sodium leak postulation and analysis, as well as a fire PRA will be performed as part of the OL application, further information can reasonably be left for later consideration. At the final design stage in the OL review, the staff will evaluate the final design for appropriate sodium leak postulation approaches and analysis to ensure the plant and specific SSCs can perform their functions reliably consistent with PDC 3 and 73.

8.3.1.7 Fire Analyses

PSAR section 8.2.3 discusses the fire analyses and indicates that fire modeling, fire PRA, fire hazards analysis, and fire safe shutdown analysis will be performed as part of the OL application. The PSAR states that as the Natrium design progresses, strategies to manage the hazards identified will be developed and may include manual firefighting, automatic or manual

suppression with agents compatible with sodium, inerting, purging, using incombustible materials, and SSC specific design features. As discussed above, the KU1 fire protection design strategy follows the guidance in RG 1.189 and focuses on the designation of fire areas supported by the design of fire barriers such that a protected path of safe shutdown is maintained while assuming all equipment in any one fire area is rendered inoperable by fire.

PSAR section 8.2.3.1 states that the KU1 design uses NUREG/CR-6850 “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” (ML15167A411) to provide a structure for developing a fire PRA, including to develop the sodium frequency assessment, scenario development, and fire modeling. The PSAR states that a sodium fire modeling methodology with complete verification and validation and inputs to support the fire PRA will be developed at the OL stage. PSAR section 8.2.3.2 states that the design ensures safe shutdown capability is maintained in the event of a fire by limiting fire damage to a single train of safety-significant SSCs so that the capability to shut down the plant safely is ensured. The PSAR also states that the design uses methodologies approved by the NRC to demonstrate DID and safe shutdown margin is available.

The staff reviewed the information provided with respect to the fire analyses, including the fire modeling, fire PRA, fire hazards analysis and safe shutdown analyses. The staff notes that the fire analysis methods will follow RG 1.189 and NUREG/CR-6850. Additionally, the staff notes that a sodium fire modeling methodology with complete verification and validation for inputs to support the fire PRA will be developed to meet the ASME/ANS RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Nuclear Power Plants.” Based on conformance with RG 1.189, NUREG/CR-6850, and the plan to develop a sodium fire modeling methodology with complete verification and validation for inputs to support the fire PRA, the staff finds that the preliminary design information related to fire analyses is consistent with PDC 3 and 73.

At the preliminary design stage, there are limited details of how these SSCs will be designed to perform their function to prevent or mitigate sodium fire. Given that sodium fire hazards and conventional fire hazards are highly dependent on the final design configuration and a fire PRA will be performed as part of the OL application, further information can reasonably be left for later consideration. At the final design stage in the OL review, the staff will evaluate the final methodology development, verification and validation effort, fire hazards analysis and the safe shutdown analysis to ensure their ability to perform their functions reliably in the event of postulated sodium leakage and conventional fire risk. This will include a comprehensive review of the inputs used for the scoping fire models that will be used to inform design decisions and fire conditions expected for each postulated sodium leak scenario.

8.3.1.8 Fire Protection Program

PSAR section 8.2.4 indicates that a fire protection program will be developed at the OL stage to meet 10 CFR 50.48(a) using the guidance in RG 1.189. PSAR section 8.2.4.1 states that there is no active or manual fire suppression for large sodium fires and that sodium leaks and any resulting sodium fires are mitigated by passive means through NNA design features.

USO states that firefighting strategies for conventional deep-seated fires will be described in the fire protection plan submitted at the OL stage. USO also states that deep-seated sodium fire are contained within the NNA features and are passively self-suppressing. USO also states that open sodium leaks in the HAA are prevented by containing leakage from the primary Sodium Processing System with guard piping of the system and containing leakage from the

Intermediate Heat Transport System within a leak guard enclosure that drains to the RAB pipe chase.

USO states that the Reactor Building (RXB) and RAB do not contain water-based fire suppression systems. Water-based fire suppression systems are contained in the Fuel Handling Building (FHB), which also has sodium-containing SSCs, but USO also states that the FHB establishes fire barriers or other mitigative features to separate areas containing sodium from areas protected by a water-based suppression system.

The staff reviewed the information provided with respect to the fire protection program and notes that the fire protection program will meet 10 CFR 50.48(a) and the guidance in RG 1.189. The staff review noted that some aspects of the conventional firefighting strategies cannot be utilized in the RXB, RAB and portions of the FHB due to the limitations surrounding the use of water for concerns surrounding sodium-water interactions. The firefighting strategies for both incipient and deep-seated fire conditions will be further developed during the OL stage. The staff notes that incipient firefighting strategies considered may include one or some combination of manual firefighting, automatic or manual suppression with agents compatible with sodium, inerting, purging, using incombustible materials, and SSC specific design features. Based on plans to meet 10 CFR 50.48(a), follow the guidance in RG 1.189, and identification of various firefighting strategies, the staff finds that the preliminary design information related to the fire protection program is consistent with PDC 3 and 73. The staff particularly notes that deep-seated firefighting strategies may require offsite response with specialized training, which will require significant description and justification at the OL stage. Additional discussion regarding fire protection can be found in SE section 7.5.2.

At the preliminary design stage, there are limited details of how the fire protection program will be designed to perform its design function to prevent or mitigate conventional and sodium fires in areas where water cannot be utilized. Given that sodium fire hazards and conventional fire hazards are highly dependent on the final design configuration and a fire PRA will be performed as part of the OL application, further information can reasonably be left for later consideration. At the final design stage in the OL review, the staff will evaluate the fire protection program to ensure the ability to perform its functions reliably in the event of postulated sodium leak and/or conventional fire. This will include a review of the installed fire protection suppression systems, the firefighting strategies which will be used for the various fire conditions, the design features which will enable passive and self-suppressing sodium fires and any reliance on off-site fire brigade support to ensure the ability to achieve safe shutdown.

8.3.2 Conclusion

Based on the review described above, the staff concludes that the information in PSAR section 8.2 is sufficient and meets applicable guidance and regulatory requirements identified in this chapter for the issuance of a CP. Further information as may be required to complete the review can reasonably be left for later consideration when the final design with the full scope fire protection plan, the fire PRA, and the safe shutdown analysis have been completed at the OL stage

8.4 References

American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel Code, Section XI Rules for Inservice Inspection of Nuclear Reactor Facility Components, Division 2,

Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities,” ASME: New York, NY, 2019 edition.

----. ASME NQA-1-2015, “Quality Assurance Program Requirements for Nuclear Facilities,” New York, NY, dated February 20, 2015.

----. ASME Standard QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” ASME, 2017 edition.

----. ASME, “Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Facility Components, Division 5, High Temperature Reactors,” ASME, 2017 edition.

Institute of Electrical and Electronic Engineers (IEEE) Standard (Std.) 344, “Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations,” IEEE, 2013 edition.

----. IEEE Std. 323, “Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” IEEE, 2003 editions.

----. IEEE Std C37.98, “Standard for Seismic Qualification Testing of Protective Relays and Auxiliaries for Nuclear Facilities,” IEEE, 2013 edition.

Institute of Electrical and Electronics Engineers/ International Electrotechnical Commission (IEEE/IEC) Std. 60780-323, “International Standard - Nuclear facilities - Electrical equipment important to safety - Qualification,” IEEE/IEC, 2016 edition.

9 CONTROL OF PLANT RADIOACTIVE EFFLUENTS, PLANT CONTAMINATION AND SOLID WASTE

This chapter of the safety evaluation (SE) describes the staff's review and evaluation of the Kemmerer Power Station Unit 1 (KU1) preliminary safety analysis report (PSAR) chapter 9, which contains the preliminary descriptions for release of liquid and gaseous effluents, the liquid and gaseous waste treatment systems, plans to minimize plant contamination, and the control of solid radioactive waste.

The applicable regulatory requirements for the evaluation of the KU1 PSAR chapter 9 are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation;"
- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive materials in effluents – nuclear power reactors;"
- 10 CFR 50.35, "Issuance of construction permits;"
- 10 CFR 50.40, "Common standards;"
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents;" and
- 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

The applicable guidance for the evaluation of the KU1 PSAR chapter 9 are as follows:

- U.S. Nuclear Regulatory Commission (NRC), branch technical position (BTP) 11-6, "Postulated Radioactive Releases due to Liquid-Containing Tank Failures," Rev. 4. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15027A401);
- DANU-ISG-2022-03, "Advanced Reactor Content of Application Project, Chapter 9, 'Control of Routine Plant Radioactive Effluents, Plant Contamination and Solid Waste'" (ML23277A141);
- RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Rev. 1 (ML003740384);
- RG 1.143, "Design Guidance for Radioactive Waste Management Structures, Systems, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Rev. 2 (ML013100305);

- RG 4.1, “Radiological Environmental Monitoring for Nuclear Power Plants,” Rev. 2. (ML091310141); and
- RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” Rev. 0 (ML080500187).

The following principal design criteria (PDC), as defined in section 5.3 of the KU1 PSAR, apply to the review of the control of plant radioactive effluents, plant contamination and solid waste:

- PDC 60, “Control of Releases of Radioactive Materials to the Environment;”
- PDC 61, “Fuel Storage and Handling and Radioactivity Control;”
- PDC 63, “Monitoring Fuel and Waste Storage;” and
- PDC 64, “Monitoring Radioactivity Releases.”

9.1 Liquid and Gaseous Effluents

PSAR section 9.1 describes the structures, systems, and components (SSCs) associated with the control of liquid waste and gaseous effluents. During normal operations of the KU1 plant, the primary source of gaseous radiological effluent is from the sodium cover gas (SCG) system, which contains xenon (Xe), krypton (Kr), activated argon (Ar) and hydrogen. SCG is processed by the gaseous radwaste processing (RWG) system. Sources of gaseous effluent include the RWG, nuclear island heating, ventilation and air conditioning systems (NHV), reactor air cooling (RAC) system, and assumed releases of tritium from the cold salt storage tank.

During normal operation of the KU1 plant, spent fuel assemblies are the primary source of radioisotopes in the liquid radwaste processing (RWL) system. The RWG and RWL are in the fuel handling building within the nuclear island. The staff’s evaluation of the design of RWG is provided in section 7.4.1 of this SE, so this section focuses on the design of the RWL.

9.1.1 Technical Evaluation

As discussed in DANU-ISG-2022-03, section 9.1, the construction permit (CP) application should provide assurance that the limits on the release of radioactive liquid and gaseous effluents will meet the relevant requirements in 10 CFR Part 20 and Part 50.

9.1.1.1 Liquid and Gaseous Waste Management System Description

Paragraph 10 CFR 50.34a(a) requires, in part, that a CP application include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. PDC 60 specifies that systems shall be designed to control the release of radioactive materials in gaseous and liquid effluents and ensure that the waste systems have sufficient holdup capacity for retention of gaseous and liquid effluents containing radioactive materials.

US SFR Owner (USO) provides the descriptions for the RWG in PSAR section 7.4.1 and the RWL in PSAR section 9.1.3. The RWG overview is provided in PSAR figure 7.4.1-1 and the

RWL overview is provided in PSAR figure 9.1-1. PSAR table 9.1-6 provides information on the Ar, Kr, and Xe hold up times for the KU1 facility.

PSAR section 9.1.3, states the purpose of the RWL, provides a description of the components that comprise the RWL, and states that the system is classified as a non-safety related with no special treatment (NST) system. PSAR table 9.1-2 provides information on the sources of liquid waste, capacity of tanks, and preliminary yearly activities for the components specified.

The staff reviewed the description of the RWL in PSAR section 9.1 and determined that it is sufficient to meet the preliminary design information required in 10 CFR 50.34a because RWL holdup capacity for liquid waste was discussed and functional arrangement information was provided in PSAR figure 9.1-1.

The staff conducted an audit (ML25302A443) of design documents and calculations to confirm that the RWL design has sufficient storage space for liquid waste and that the tanks were sized based on conservative input assumptions. Based on its review of the PSAR, as confirmed through audit, the staff determined that the preliminary design of the RWL is consistent with PDC 60 because it has sufficient capacity to store and process radioactive liquid waste generated by the KU1 facility.

As documented in SE section 7.1.4, the staff also determined that the RWG description is sufficient to meet the preliminary design information required in 10 CFR 50.34a and demonstrate consistency with PDC 60.

9.1.1.2 Equipment Design Requirements

PDC 61 requires that the radioactive waste systems be designed to ensure adequate safety under normal and postulated accident conditions.

PSAR sections 9.1.3 and 9.1.4 state that the RWL is an NST system that is designed, constructed, and tested to the requirements set forth in the applicable codes and standards listed in RG 1.143. The PSAR states that the KU1 partially conforms to RG 1.143 with use of section C.1 and methodologies in section C.5 and C.6 are replaced by the methodologies discussed in PSAR sections 5.1 and 6.1.3. The RWL is stated to meet or exceed the guidance for RW-IIc classified systems or components. PSAR table 9.1-7 lists the codes and standards applied to the design of the RWL. PSAR section 9.1.4 also states that the RWL processing filtering systems and storage tanks will have suitable shielding where appropriate. PSAR section 5.1 discusses safety classifications of SSCs and PSAR section 6.1.3 discusses the design basis hazard level (DBHL) requirements for non-SR (i.e., NSRST and NST) SSCs, by providing reference to American Society of Civil Engineers (ASCE) 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures."

PSAR sections 9.1.3 and 9.1.4 also state that RWL components that contain radioactivity and are in controlled shielded areas and that the spent resin storage tanks are provided with an enclosure that satisfies shielding requirements.

PSAR section 2.5.3.2 describes the accident release scenario of a liquid radwaste processing waste holdup tank with a capacity of 4,000 gallons, and an assumed release of 80 percent of the tank's contents. This PSAR section describes the analysis performed to calculate the radionuclide concentrations in ground water at the site boundary that a member of the public could be exposed to. PSAR section 2.5.3.2.5 reports the total effective dose equivalent (TEDE)

dose results from this accidental release to a member of the public to be 5.60×10^{-3} millirem (mrem) per year.

The staff reviewed the RWL equipment design requirement information in PSAR sections 5.1, 6.1.3, 9.1, and in PSAR table 9.1-7 and determined it is acceptable because the equipment and components are designed, constructed, and tested to the requirements set forth in the applicable codes and standards listed in RG 1.143.

The staff review of an accidental release of liquid effluents to the ground and surface water is found in SE section 2.5.1.3.2. As described in SE section 2.5.1.3.2, the staff's confirmatory analysis using the guidance contained in the NRC's BTP 11-6, determined that the TEDE dose result of 5.60×10^{-3} mrem per year is reasonable and that the proposed design would prevent or minimize public exposure in the event of an accidental release below the public dose limit of 100 mrem per year as specified in 10 CFR 20.1301, "Does limits for individual members of the public." The staff observes that preliminary accidental releases of liquid effluent with no mitigative measures are below the 10 CFR 20.1301 limits.

As stated in PSAR section 9.1.3, the RWL components containing radioactivity are in shielded areas. The staff also understands that the design of the RWL will apply As Low as Reasonably Achievable (ALARA) design principles which will be fully evaluated in the operating license (OL) application.

The staff determined that the preliminary design of the RWL is consistent with PDC 61 because USO applies the design guidance contained in RG 1.143, with appropriate modifications, and describes methods to maintain occupational doses ALARA. The staff will review additional details regarding radiation shielding and specific component sources terms at the OL stage.

As documented in SE section 7.1.4, the staff also determined that the preliminary design of the RWG is consistent with PDC 61.

9.1.1.3 Effluent Sources and Dose Analysis

Paragraph 10 CFR 50.34a(a), in part, specifies the design objectives for equipment to control releases of radioactive material in effluents in nuclear power reactors by reference to the numerical design objectives contained in 10 CFR Part 50, Appendix I. Paragraph 10 CFR 50.34a(b)(2) specifies an estimate of the quantity of radionuclides in both gaseous and liquid waste that are expected to be released annually. The requirements, applicable at the OL stage, contained in 10 CFR 20.1301 specify the 1 millisievert (mSv) (100 mrem) annual public dose limit. Paragraph 10 CFR 20.1301(e) specifies that licensees shall also comply with the standards contained in 40 CFR 190.

PSAR section 9.1.1 and table 9.1-1 identify the primary sources of gaseous effluents produced at KU1. These sources are gases from the SCG which contains Xe, Kr, Ar, and tritium which is processed by the RWG and effluent being emitted from the NHV, the RAC, and tritium sources from the cold salt storage tank cover gas and steam generator.

PSAR section 9.1.2, discusses that KU1 is a zero radioactive liquid discharge facility processing all liquid waste through filtration, demineralization, and ion exchange to remove or reduce radioactivity prior to the liquid being evaporated and directed to the NHV. The gaseous effluent release points are from the plant effluent stack, tritium releases from the cold salt storage tank

cover gas and steam generator, and the RAC. The results of the dose analysis using the GASPARD II computer code are contained in PSAR tables 9.1-3, 9.1-4, and 9.1-5.

PSAR section 2.4.4.2, provides the X/Q and D/Q values used for the normal operational dose analysis. The information used for the GASPARD II analysis is found in PSAR table 2.4-81. This provides the X/Q and D/Q information needed for the GASPARD II computer code to evaluate doses at the specified locations.

In addition to reviewing the information in the PSAR sections 9.1.1, 9.1.2, and 2.4.4.2, the staff conducted an audit of the effluent source term and effluent dose calculations performed by USO (ML25302A443). In the audit, the staff reviewed the inputs and justifications made into the GASPARD II computer code used for the results located in PSAR tables 9.1-3 to 9.1-5. The staff also reviewed additional supporting information related to how the gaseous effluent source terms were determined for the RWG, the RAC, and the spent fuel pool source terms. The audit review enhanced staff understanding of the methods used to develop initial source term estimates and confirmed that the assumptions reflected anticipated plant operations and accounted for removal using decay and holdup times appropriately.

Based on its review of the source term information provided in PSAR table 9.1-1, as confirmed in audit, the staff determined that the estimated source term information is appropriate because it considers the principal radionuclides expected from operations and appropriately accounted for decay and holdup times consistent with the systems available at the facility.

The staff also performed a confirmatory calculation using GASPARD II to verify the results provided in PSAR tables 9.1-3 to 9.1-5.

The staff's analysis verified that the dose results reported in the PSAR reflect the use of finite dose correction factors that adjust the dispersion model to account for elevated releases of Ar-41 noble gas due to the heat of the gas released from the RAC. The staff's calculations confirmed the dose results accounting for elevated releases from the RAC incorporating the methodology provided by USO in their audit documentation, which applied finite dose correction factors taken from the MACCS theory manual. The basis of this analysis is found in the MACCS theory manual and SAND2021-13183. A similar calculation to determine the same geometry factors I_1 and I_2 are also found in RG 1.109 appendix F to account for elevated releases of noble gases. The staff's evaluation of the applicant's calculation determined that the MACCS theory manual applies models consistent with RG 1.109. Because the methodology used is consistent with a staff approved methodology in RG 1.109, the staff determined that the method of evaluating elevated releases of Ar-41 is acceptable.

The staff determined that the dose results provided in PSAR tables 9.1-3 to 9.1-5 are acceptable because the use of GASPARD II, in addition to the method for accounting for elevated releases summarized above, reflects the use of RG 1.109 which provides an acceptable method for determining the doses from gaseous effluent releases. In addition, the staff's review of the audit information confirmed the use of site-specific assumptions as input for use in the GASPARD II code which is appropriate for a CP application with a designated site location. The results of the dose analysis show that gaseous releases are below the dose objectives contained in 10 CFR Part 50 Appendix I, and as a result, these calculated annual doses also reflect that doses will not exceed the public dose limits provided in 10 CFR 20.1301 or 40 CFR 190 at the OL stage.

The staff determined that preliminary information about the effluent sources and dose analysis on the facility's proposed effluent releases meet the CP requirements for 10 CFR 50.34a because proposed effluent releases from the KU1 facility use site specific information and are below the dose objectives specified in 10 CFR Part 50 Appendix I are described. The staff will review final effluent releases at the OL stage to ensure requirements in 10 CFR 20.1301, 40 CFR Part 190, and 10 CFR 50.34a are met.

9.1.1.4 Effluent Radiation Monitoring

PDC 63 specifies that appropriate systems shall be provided for radioactive waste systems to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions. PDC 64 specifies the monitoring requirements for the effluent discharge paths for radioactivity that may be released from normal operations. For the KU1 facility this pathway is only for the release of gaseous effluents.

PSAR section 9.1.3 states that process radiation monitoring instrumentation is provided on the evaporator skid to monitor liquid for radioactivity prior to reuse, evaporation, or recycling. PSAR section 7.5.1.2 states exhaust from RXB, FHB, RAB, and fuel auxiliary building are combined through the main plant stack for effluent monitoring by the radiation monitoring system described in PSAR section 7.6.6. PSAR section 7.2.1 describes the monitoring and sample process for the RAC.

The KU1 Environmental Report (ER) (ML24088A072) section 2.9 and PSAR section 9.1 provide information on the radiological and environmental monitoring program (REMP) and the offsite dose calculation manual (ODCM). These sections summarize that the REMP and ODCM will be a component of the KU1 process and effluent monitoring and sampling program and will be consistent with Nuclear Energy Institute (NEI) 07-09A, "Generic final safety analysis report (FSAR) Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," (ML091460258) and RG 4.1. The ODCM and REMP will be developed at the OL stage and will describe effluent and environmental monitoring with PSAR section 9.1 stating that details of the effluent and environmental radiation monitoring, including locations, monitoring ranges, sampling methods, and specific functions will be described in the ODCM.

The staff reviewed the information contained in the PSAR sections 7.2.1, 7.5.1.2, 7.6.6, and 9.1.3 related to effluent releases from the plant stack, the RAC, and the energy island. In addition to the information reviewed in the PSAR, the staff conducted an audit to better understand how releases from the RAC and the cold salt storage tanks would be quantified by sampling and what details for this sampling methods would be contained within the ODCM that will be provided as part of the OL application.

The staff determined that preliminary information about effluent monitoring of the facility's releases is acceptable because USO identified the expected release points at the facility and provided proposed methods for tracking radioactive effluent releases from the facility by providing monitoring or a description of the sampling methods to quantify those releases.

Based on its review and evaluation above, the staff determined the preliminary design of the RWL is consistent with PDC 63 and 64 because USO provided preliminary information on the gaseous monitoring for the facility. As documented in SE section 7.1.4, the staff also determined that the RWG is consistent with PDC 63 and 64.

9.1.2 Conclusion

Based on its review and evaluation above, the staff concludes the information in KU1 PSAR section 9.1 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a CP in accordance with 10 CFR 50.34a, 10 CFR 50.35 and 10 CFR 50.40.

9.2 Contamination Control

Section 9.2 of the PSAR describes design features and procedures that minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste consistent with the requirements in 10 CFR 20.1406, "Minimization of contamination."

9.2.1 Technical Evaluation

As discussed in DANU-ISG-2022-03 section 9.2, the purpose of the staff's review of contamination control is to ensure that the application describes sufficient design and programmatic information to address 10 CFR 20.1406, which at the OL stage, requires that the design and operation minimize contamination of the facility and the environment. To meet provisions related to contamination control in 10 CFR 20.1406 an applicant must describe how it intends to minimize, to the extent practicable, the contamination of the facility, the environment and the generation of radioactive waste. As provided in 10 CFR 20.1002, "Scope," the regulations in 10 CFR Part 20 apply to "persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material or to operate a production or utilization facility...." USO has applied for a CP, which does not provide a license to operate the facility. In the CP application, USO also has not applied for licenses to receive, possess, use, transfer, or dispose of any byproduct, source, or special nuclear material at the facility. Therefore, the staff did not evaluate whether the requirements in 10 CFR Part 20 would be met in support of the CP. Instead, the staff assessed whether USO had identified the relevant requirements for an operating facility and provided descriptions of how the preliminary facility design will, to the extent practicable, minimize contamination of the facility and the environment, as required for operating facilities by 10 CFR 20.1406. This is consistent with 10 CFR 50.40(a), which provides that in determining whether a CP may be issued, the Commission will be guided by consideration of reasonable assurance that USO will comply with the NRC's regulations, including the regulations in 10 CFR Part 20, and that the health and safety of the public will not be endangered.

PSAR section 9.2 states that design features, programmatic elements, and procedures consider the guidance in RG 4.21. PSAR section 9.2 also states that a description of the design features and procedures for the control of radioactive contamination for the facility, including consideration of RG 4.21, will be provided at the OL stage.

The staff determined that the PSAR information on contamination control is acceptable for a CP application because USO has committed to using RG 4.21, consistent with the guidance for contamination control in DANU-ISG-2022-03. Further details on the design features as well as procedures for contamination control will be provided at the OL stage for staff review to evaluate conformance with 10 CFR 20.1406.

9.2.2 Conclusion

Based on the evaluation above, the staff concludes the information in PSAR section 9.2 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a CP in accordance with 10 CFR 50.35 and 10 CFR 50.40.

9.3 Solid Radwaste Processing

Section 9.3 of the PSAR describes the solid radwaste processing system (RWS) for the unit that is designed to process both wet and dry solid waste resulting from normal plant operation and abnormal operational occurrences. The RWS is located within the fuel handling building and consists of three key design features: a dewatering skid, a compactor skid and a storage area.

9.3.1 Technical Evaluation

As discussed in DANU-ISG-2022-03 section 9.3, the purpose of the staff's review of solid radwaste processing is to ensure that an application provides information on the design of the solid radwaste management system and assurance that the requirements in 10 CFR Part 20, 10 CFR Part 50, and applicable storage and transportation requirements will be met.

9.3.1.1 Solid Waste Management System Description

As specified in 10 CFR 50.34a, a CP application shall include a general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from the treatment of gaseous and liquid effluents and from other sources. PDC 60 specifies that systems shall be designed to control the release of radioactive materials in gaseous and liquid effluents and ensure that the waste systems have sufficient holdup capacity for retention of gaseous and liquid effluents containing radioactive materials as it applies to the generation of solid waste.

PSAR section 9.3.1 describes the ability of the RWS to process wet and dry solid waste and states that the RWS is in the fuel handling building and consists of a dewatering skid, a compaction skid, and a storage area for waste. PSAR figure 9.3-1, shows the high-level functional arrangement of the dewatering skid and the compaction skid.

ER section 6.2.2.3 states that the average annual shipment of radwaste over 40 years is 2,070 cubic feet with ER table 6.2-6 giving the estimated source term for radwaste at the time of shipment. ER section 3.4.2.3.1 states that the solid waste storage area is sized to accommodate one fuel cycle's worth of waste. PSAR figure 1.1-7 shows the RWS drum skid and proposed storage area for radwaste.

The staff reviewed the information contained in PSAR section 9.3 and ER section 6.2.2.3 to verify that the RWS is adequately described and has sufficient holdup capacity for solid waste. The staff identified that the description of the components of the RWS includes the components of the RWS with storage area locations identified and includes a high-level functional arrangement of the RWS that illustrates the RWS interfaces.

The staff determined the preliminary information on the RWS's solid waste generation rates and available storage area for solid waste is adequate to meet the CP requirements for 10 CFR 50.34a because the PSAR and ER provide a general description of the provisions for

packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from the treatment of gaseous and liquid effluents and from other sources.

The staff also determined that the preliminary RWS design is consistent with PDC 60 because the PSAR describes the means for packaging, storing, and shipping waste containing radioactive materials and specifics that the design will have equipment available to collect, segregate, process, sample, and monitor waste generated from normal operations and will control and confine radionuclides against uncontrolled releases.

9.3.1.2 Equipment Design Requirements

PDC 61 specifies that the RWS that handle radioactivity shall be designed to assure adequate safety under normal and postulated to not exceed the dose limits in 10 CFR Part 20. The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. PDC 61 also states that the RWS shall be designed with suitable shielding for radiation protection and appropriate containment, confinement, and filtering systems. PSAR sections 9.3.1 and 9.3.2 states that the RWS is an NST system that is designed, constructed, and tested to the requirements set forth in the applicable codes and standards listed in RG 1.143 methodologies section C.5 and C.6 are replaced by the methodologies discussed in PSAR sections 5.1 and 6.1.3. PSAR section 5.1 discusses the methodology for safety classifications of SSCs. PSAR section 6.1.3 discusses the DBHL requirements for non-SR (i.e., NSRST and NST) SSCs, by providing reference to ASCE 7-16. The RWS is stated to meet or exceed the guidance for RW-IIc classified systems or components. In addition, PSAR table 9.3-1 lists the codes and standards applied to the design of the RWS.

The staff reviewed the information summarized above to understand how the RWS design information partially conforms with the guidance contained in RG 1.143. The staff determined that the information in PSAR table 9.3-1 is acceptable because the equipment and components are design, constructed, and tested to the requirements set forth in the applicable codes and standards listed in RG 1.143.

As stated in PSAR section 9.3.2 the RWS is designed to maintain occupational dose ALARA and states that the RWS and its system components are in a shielded area. The staff notes that this level of detail is appropriate because the information provided is consistent with the information required by DANU-ISG-0222-03 for a CP application and detailed design information is not available or expected to be available at the CP stage.

The staff determined that the preliminary design of the RWS is consistent with PDC 61 because USO applies the design guidance contained in RG 1.143, with appropriate modifications, and describes methods to maintain occupational doses ALARA. The staff will review additional details regarding radiation shielding and specific component source terms at the OL stage.

9.3.1.3 Radiation Monitoring

PDC 63 specifies that appropriate systems shall be provided for radioactive waste systems to detect excessive radiation levels and to initiate appropriate safety actions.

PSAR section 9.3.1 provides information on the monitoring of waste to state that dry solid waste is monitored and classified in accordance with 10 CFR 61.55, "Waste classification." The RWS includes process sampling as well as local area radiation monitoring. PSAR section 9.3.2

describes the presence of area radiation monitoring and airborne radioactive material monitoring for the compaction skid. PSAR section 7.6.6 provides information on the types of radiation monitoring at the facility.

The staff reviewed the information contained in PSAR sections 7.6.6 and 9.3.1 related to radiation monitoring and determined that preliminary information contained in these sections is appropriate for the CP application review because USO has described their intent to monitor and classify waste to meet the requirements in 10 CFR 61.55.

The staff determined that the preliminary design of the RWS is consistent with PDC 63 because USO provided preliminary information for RWS monitoring that will be further evaluated in the OL application.

9.3.1.4 Operational Controls and the Process Control Program (PCP)

The following regulations summarize the storage and transportation OL requirements:

- 10 CFR Part 61, "Licensing requirements for land disposal of radioactive waste;"
- 10 CFR 71.43 "General standards for all packages;"
- 10 CFR 71.47 "External radiation standards for all packages;"
- 10 CFR 71.87 "Routine determinations;" and
- 10 CFR 20.1904 "Labeling containers," as it relates to classifying, processing, and disposing of radioactive waste in approved containers and packaging methods.

PSAR section 9.3.2 states that a process control plan will be developed consistent with the guidance contained in NEI 07-10A, "Generic FSAR Template Guidance for Process Control Program," (ML091460236) to meet 10 CFR Part 61 and will be provided at the OL stage.

As stated in DANU-ISG-2022-03 section 9.3, the NRC has approved the PCP template via the SE for NEI 07-10A. An applicant who wishes to employ the NEI 07-10A PCP template for the portion of the chapter 9 application content it covers should explain why the template applies to its proposed facility, including how the conditions for use of the template are satisfied. The PCP presents methods by which liquid and wet waste may be processed and disposed of in accordance with Part 61 and packaged and transported in accordance with 10 CFR Part 20, 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," and 49 CFR Part 171, "General Information, Regulations and Definitions," through 49 CFR Part 180, "Continuing Qualification and Maintenance of Packagings."

The staff determined that the applicant's use of NEI 07-10A as a PCP template is acceptable for a CP application because it aligns with the guidance in DANU-ISG-2022-03 section 9.3 and the SE for NEI 07-10A explains its appropriate use.

9.3.2 Conclusion

Based on its evaluation above, the staff concludes the information in PSAR section 9.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this section for the issuance of a CP in accordance with 10 CFR 50.34a, 50.35 and 50.40.

9.4 References

American Society of Civil Engineers (ASCE) Standard 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," ASCE, 2016 edition.

Bixler, Nathan, and Nosek, Andrew, "MACCS Theory Manual," SAND2021-11535, (2021), <https://doi.org/10.2172/1820907>

Clayton, Daniel J., "Update to the Finite Cloud Dose Correction Factors in MACCS," SAND2021-13183, (2021), <https://doi.org/10.2172/1827253>

10 CONTROL OF OCCUPATIONAL DOSE

This chapter of the safety evaluation (SE) describes the staff's review and evaluation of Kemmerer Power Station Unit 1 (KU1) preliminary safety analysis report (PSAR) chapter 10, which contains preliminary descriptions of the radiation protection program (RPP), the as low as is reasonably achievable (ALARA) program, the facility design features and design considerations to maintain exposures ALARA, and the preliminary dose assessment performed for the construction permit (CP) application.

The applicable regulatory requirements for the evaluation of the control of occupational dose are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(6).

The applicable guidance for the evaluation of the control of occupational dose is as follows:

- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," section 12 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23277A139).
- DANU-ISG-2022-04, "Advanced Reactor Content of Application Project Chapter 10, 'Control of Occupational Dose,'" (ML23277A142).
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," section 12.5 (ML13155A232).

The following principal design criterion (PDC), as defined in section 5.3 and 10.3 of the PSAR, applies to the review of control of occupational dose:

- PDC 19, "Control Room"
- PDC 61, "Fuel Storage and Handling and Radioactivity Control"

As part of its CP application, US SFR Owner, LLC (USO) did not apply for licenses to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material in accordance with 10 CFR Part 30, "Rules of general applicability to domestic licensing of byproduct material," 10 CFR Part 40, "Domestic licensing of source material," and 10 CFR Part 70, "Domestic licensing of special nuclear material," respectively. Therefore, occupational dose requirements associated with these regulations were not evaluated by the staff at this time. The staff's review of occupational dose requirements associated with those regulations would occur as part of its review of those applications when received. The staff notes that USO is not permitted to engage in any activity for which these licenses are required until it applies for and receives these licenses.

10.1 Technical Evaluation

As discussed in DANU-ISG-2022-04 a CP application should include (1) a description of its plans to develop, at the operating license (OL) application stage, comprehensive worker protection programs, organizational structure, training, and monitoring to ensure the requirements are met in 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations;" and 10 CFR Part 20, "Standards for Protection Against Radiation;" (2) a description of the design-specific PDC necessary to control occupational exposures, and (3) a preliminary description of design provisions necessary to control

occupational exposure, including ensuring that occupational doses are ALARA, such as shielding, ventilation, area radiation and airborne radioactivity monitoring instrumentation, and dose assessment for expected occupancy.

Given this guidance the staff will only make findings related to 10 CFR 50.34(a)(6) and PDC 19 and 61 during the CP review. The staff's review of the requirements contained 10 CFR Part 20 will be conducted during the OL review consistent with the requirement in 10 CFR 50.34(b)(3) which provides the OL requirements to provide the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

As provided in 10 CFR 20.1002, "Scope," the regulations in 10 CFR Part 20, apply to "persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material or to operate a production or utilization facility...." USO has applied for a CP, which does not provide a license to operate the facility. In the CP application, USO also has not applied for licenses to receive, possess, use, transfer, or dispose of any byproduct, source, or special nuclear material (SNM) at the facility. Therefore, the staff did not evaluate whether the requirements in 10 CFR Part 20 would be met in support of the CP. Instead, the staff assessed whether USO had identified the relevant requirements for an operating facility and provided descriptions of how the preliminary facility design will, to the extent practicable, meet the 10 CFR Part 20 requirements as discussed in the technical evaluation. This is consistent with 10 CFR 50.40(a), which provides that in determining whether a CP may be issued, the Commission will be guided by consideration of reasonable assurance that USO will comply with the NRC's regulations, including the regulations in 10 CFR Part 20, and that the health and safety of the public will not be endangered.

10.1.1 Radiation Protection Program

As discussed in DANU-ISG-2022-04, the CP application should include, among other things, a description of the applicant's plans to develop, at the OL stage, worker protection programs, organizational structure, training, and monitoring to meet the requirements of 10 CFR Parts 19 and 20. Consistent with this guidance 10 CFR 50.34(a)(6) addresses a requirement to provide a preliminary plan for the applicant's organization, training of personnel, and conduct of operations. As such, the staff reviewed the PSAR to determine whether the applicant had adequately described its plans to develop the RPP which would address the 10 CFR 50.34(a)(6) requirements as related to the RPP. The evaluation of the overall organization as it meets the requirements of 10 CFR 50.34(a)(6) is addressed in section 11.1 of this SE.

The PSAR chapter 10 introduction states that an RPP will be developed and implemented prior to receipt of radioactive materials on-site which will include comprehensive worker protection practices, an organizational structure to support radiation protection, training, and monitoring, and administrative programs. In addition, the PSAR states that the requirements of 10 CFR 19.12, "Instruction to workers," and 10 CFR Part 20 will be met by USO's RPP. PSAR section 10.1 provides a list of Regulatory Guides (RGs) that will be implemented as part of USO's RPP. USO also stated that it will submit procedures and programs that incorporate ALARA principles during plant operations, including the RPP, at the OL stage. The staff will review these submissions at the OL stage.

PSAR section 11.1.4 and figure 1.4-1 provide information on the operating organization which will include the radiation protection organization. Radiation protection training is addressed by PSAR table 1.4-1, which states that conformance with RG 8.27, "Radiation Protection Training

for Personnel at Light-Water-Cooled Nuclear Power Plants,” Rev. 0 (ML003739628), and RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants” Rev. 4 (ML19101A395), will be addressed at the OL stage.

The staff reviewed PSAR sections 10.1.1, 10.1.2, 10.1.3, 11.1.4, and PSAR table 1.4-1 which describe what will be included in the RPP. DANU-ISG-2022-01 provides reference to NUREG-0800 section 12.5, which provides additional guidance for a radiation program which includes a list of RGs that the RPP will be consistent with, and, although LWR-based, may provide useful insights for staff review of non-LWR applications. Consistent with the guidance referenced in NUREG-0800 section 12.5, the staff’s review determined that the high-level description of the RPP and commitments to follow applicable guidance are appropriate at the CP stage because USO will address applicability of RG 1.8, RG 8.13, “Instruction Concerning Prenatal Radiation Exposure,” Rev. 3 (ML003739505), RG 8.15, “Acceptable Programs for Respiratory Protection,” Rev. 1 (ML003739528), RG 8.27, RG 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure,” Rev. 1 (ML003739438), RG 8.35, “Planned Special Exposure,” Rev. 1 (ML101370008) and RG 8.36, “Radiation Dose to the Embryo/Fetus,” Rev. 0 (ML003739548) during the OL application, and committed to follow a set of RGs (i.e., RG 8.2, “Administrative Practices in Radiation Surveys and Monitoring,” Rev. 1 (ML110460093), RG 8.7, “Instructions for Recording and Reporting Occupational Radiation Exposure Data,” Rev. 4 (ML17221A245), RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable,” Rev. 3 (ML003739549), RG 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program,” Rev. 1 (ML003739554), RG 8.10, “Operating Philosophy for Maintaining Occupational and Public Radiation Exposures as Low as is Reasonably Achievable,” Rev. 2 (ML16105A136), RG 8.19, “Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates,” Rev. 1 (ML003739550), RG 8.25, “Air Sampling in the Workplace,” Rev.1 (ML003739616), RG 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses” Rev. 1 (ML22132A083), and RG 8.38, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants,” Rev. 1 (ML061350096) consistent with the guidance to develop an RPP in DANU-ISG-2022-01 and DANU-ISG-2022-04. The staff will review the KU1 RPP information at the OL stage to ensure compliance with 10 CFR 19.12, and 10 CFR Part 20.

The staff determined that the applicant has provided sufficient information with references to RGs 1.8 and RG 8.27 which provide staff with approved methods to qualify and train personnel for nuclear power plants and radiation protection training for personnel at nuclear power plants. Therefore, the staff determined that the applicant has provided sufficient information with references to the cited RGs that establish the preliminary plan for the applicant’s organization, training of personnel, and conduct of operations as required by 10 CFR 50.34(a)(6).

10.1.2 ALARA Program

As discussed in DANU-ISG-2022-04, the CP application should include, among other things, a description of the applicant’s plans to develop, at the OL stage, worker protection programs, organizational structure, training, and monitoring to meet the requirements of 10 CFR Part 20.

PSAR section 10.2 states that USO will implement an ALARA program consistent with RG 8.10. RG 8.10 describes methods and procedures that the staff considers acceptable for maintaining radiation exposures to occupational workers and the public ALARA. The staff determined the plan to implement an ALARA program is acceptable because USO states that its plan will

conform to RG 8.10. The staff will review the KU1 ALARA program at the OL stage to ensure compliance with 10 CFR 20.1101, "Radiation protection programs."

10.1.3 Design Considerations

As described in DANU-ISG-2022-04, a CP application should describe the design-specific PDC necessary to control occupational exposures and provide a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA. As specified in 10 CFR 50.34(f)(2)(vii), an applicant shall perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access. As discussed in 10 CFR 50.34(f)(2)(vii), in order to satisfy this regulation, CP applications must provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. As discussed in DANU-ISG-2022-01, compliance with 10 CFR 50.34(f) is not required for 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," applicants. Commission direction in SRM-SECY-15-0002, "Staff Requirements – SECY-15-0002 – Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications," (ML15266A023), however, confirmed earlier directions related to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," should be applied consistently to 10 CFR Part 50 new power reactor applications. Per DANU-ISG-2022-01, the staff should ensure that an applicant addresses the technically relevant Three Mile Island-Related items during the review process and propose license conditions requiring the appropriate items in the interim.

The PDC for KU1 are incorporated by reference into the PSAR via topical report (TR), NATD-LIC-RPRT-0002-A, "Principal Design Criteria for the Natrium Advanced Reactor," Rev. 1 (ML24283A066). PSAR section 5.3, lists the PDC from the TR and provides a brief description of how each PDC is addressed by the design. The staff found the PDC for the Natrium plant to be acceptable, as documented in NATD-LIC-RPRT-0002-A. The staff's review of PDC is also contained in SE section 5.3. PSAR section 10.3.1 cites relevant radiation protection aspects of PDC 19 and 61, which are the design-specific PDC necessary to control occupational exposures. PSAR section 10.3.2 and table 10.3-1 provide preliminary information on the major radiation sources expected at the facility. PSAR table 1.4-4 states that 10 CFR 50.34(f)(2)(vii) will have its compliance achieved at the operating license stage.

The staff reviewed PSAR section 10.3.2 and table 10.3-1 to understand the locations of USO's stated major radiation sources. An audit review (ML25302A443) of the facility's preliminary dose estimates included detailed radiation zone mapping projections which helped the staff understand preliminary radiation areas at the KU1 facility. The staff understands that these preliminary estimates for radiation zoning are the goals being established at the facility, and the final estimates for radiation zoning would be provided for review during the OL application. In addition, the staff understands that given the preliminary nature of the radiation zoning, the source term and shielding information will be finalized for review in the OL application. As a result of the audit, the staff gained an understanding of the preliminary sources of radiation exposures that will require shielding at the KU1 facility.

Given the commitments to follow the RGs described above for the RPP and the ALARA program, and in consideration of the information reviewed in the audit, the staff determined that USO will include design considerations such as shielding to ensure that the dose estimates and radiation zones are maintained to levels as desired for the KU1 facility. These RGs provide the staff the understanding that design considerations as needed will be incorporated into the

design to meet PDC 19 and 61. The staff determined that the information on radiation sources in PSAR section 10.3 is acceptable because a description of the design-specific PDC necessary to control occupational exposures, and a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA, such as shielding, ventilation, area radiation and airborne radioactivity monitoring instrumentation, and dose assessment for expected occupancy was discussed, consistent with the guidance for CPs in DANU-ISG-2022-04. This also included a description of the major radiation sources; with final details to be reviewed at OL stage. The staff will review the radiation sources and the final shielding design at the OL stage for conformance with 10 CFR 50.34(b)(3), 10 CFR 50.34(f)(2)(vii), PDC 61, and PDC 19.

10.1.4 Design Features

As described in DANU-ISG-2022-04, a CP application should provide a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA.

PSAR table 1.4-1 and section 9.2 state that USO will be in full conformance with RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," Rev. 0 (ML080500187). PSAR table 1.4-1 and PSAR section 10.3 state the facility will be in full conformance with RG 8.8 for ALARA design features. In PSAR section 10.3 USO provides a list of ALARA principles that apply to its facility and states that additional details on design considerations to maintain occupational exposures ALARA will be provided at the OL stage.

The staff's review of the information provided in PSAR section 10.3 determined that the principles referenced provided insight into how RGs 4.21 and 8.8 are being considered by the design. The information contained in RG 8.8 provides information relevant to attaining goals and objectives for planning, designing, constructing, operating, and decommissioning a light-water reactor (LWR) nuclear power station to meet the criterion that exposures of station personnel to radiation during routine operation of the station will be ALARA, and also states that much of the information in the RG is applicable to non-LWR designs. The information contained in RG 4.21 describes a method acceptable to the staff for use in implementing 10 CFR 20.1406, "Minimization of contamination," and includes a number of design features to consider for the control of contamination. References to these RGs provide the staff with a preliminary description of design provisions necessary to control occupational exposure that would address the CP information contained in DANU-ISG-2022-04.

The staff reviewed the information contained in the PSAR for determining the applicable radiation protection design features and determined it is acceptable because USO committed to follow RG 4.21 and using RG 8.8 which describe acceptable design features to maintain doses ALARA. The staff's audit also confirmed that design features in RG 4.21 and RG 8.8 are being considered, with details regarding specific design features to be finalized during the OL application. USO's commitments to RGs 4.21 and 8.8 give the staff confidence that the design will appropriately consider those design features referenced in each RG for inclusion during the OL application.

The staff determined that the radiation protection design features and commitments to conform with RGs 4.21 and 8.8 are appropriate for the issuance of a CP because USO has committed to use NRC approved methods for determining applicable design features to consider at the KU1 facility and has provided the staff a preliminary description of design provisions necessary to

control occupational exposure that would address the CP information contained in DANU-ISG-2022-04.

10.1.5 Preliminary Dose Assessment for Expected Occupancy

As described in DANU-ISG-2022-04, a CP application should provide a preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA.

The staff notes that PSAR section 10.5 provides preliminary dose estimates for occupational workers. This section states that the preliminary design target dose rates are not based on expected source terms and are not representative of actual dose rates that may be encountered. Thus, initial radiation zones were developed to provide a basis for the preliminary design of the plant. The staff reviewed the design considerations and design features as discussed in SE sections 10.1.3 and 10.1.4, which provide information on the preliminary design considerations and features that would inform the occupational dose assessment that will be performed in the OL. Together with the design considerations and design feature commitments reviewed above, the staff also examined available details of major radiation sources and radiation protection design features through audit. Consistent with the discussions in the SE sections above, the staff understood the preliminary nature of this information and that subsequent dose assessments using this information are also preliminary in nature. While the staff did not review or make a determination related to preliminary dose assessments for expected occupancy, it acknowledges inclusion of these estimates in PSAR section 10.5. With USO's commitment to follow RG 8.8 and the identification of major radiation sources, the staff determined that USO has identified the major radiation sources and will consider ALARA design considerations and features as appropriate to ensure occupational exposures ALARA. The commitments to following the RGs discussed in SE section 10.1.3 and 10.1.4 provide the preliminary description of design provisions necessary to control occupational exposure, including ensuring that occupational doses are ALARA.

The staff notes that the occupational dose assessment will be reviewed at the OL stage when details required to assess dose such as radiation sources will be provided, as required, by 10 CFR 50.34(b)(3). The staff will review the KU1 occupational dose assessment at the OL stage to ensure compliance with 10 CFR 20.1101(b) and 10 CFR 20.1201, "Occupational dose limits for adults."

10.2 Summary and Conclusions on Control of Occupational Dose

Based on its findings above, the staff concludes the information in PSAR chapter 10 is sufficient and meets the applicable guidance and, therefore, the regulatory requirements identified in this chapter for the issuance of a CP in accordance with 10 CFR 50.34(a)(6), 10 CFR 50.35, "Issuance of construction permits," and 10 CFR 50.40, "Common standards."

11 CONDUCT OF OPERATIONS

11.1 Organization

Section 11.1 of the Kemmerer Power Station Unit 1 (KU1) Preliminary Safety Analysis Report (PSAR)²⁹ describes the organizational structure and key management positions in the organizations that are responsible for facility design, design review, design approval, construction management, testing, and plant operation.

The applicable regulatory requirements for the evaluation of the design, construction, and operating organization are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of applications; technical information," paragraph (a), "Preliminary safety analysis report;"
- 10 CFR 50.34(a)(6);
- 10 CFR 50.34(a)(9);
- 10 CFR 50.40, "Common standards;"
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Criterion II, "Quality Assurance Program," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants;"
- 10 CFR 26.205, "Work hours;" and
- 10 CFR 26.4(f).

The applicable guidance for the evaluation of this section of the PSAR is as follows:

- DANU-ISG-2022-05, "Advanced Reactor Content of Application Project Chapter 11, 'Organization and Human-System Considerations' Interim Staff Guidance," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23277A143);
- NUREG-0800 section 13.1.1, "Management and Technical Support Organization," Revision (Rev.) 6, (ML15005A449);
- NUREG-0800 sections 13.1.2 - 13.1.3, "Operating Organization," Rev. 7, (ML15007A296);
- NUREG-0800 Chapter 13.5.1.1, "Administrative Procedures – General," Rev. 2, (ML15006A205);

²⁹ Subsequent to the submittal of Rev. 1 of the PSAR, USO made additional modifications to section 8.1 and 11.1 that supersede the sections 8.1 and 11.1 of PSAR Rev.1. This modification is found in ML25323A489.

- NUREG-0800 Chapter 13.5.2.1, “Operating and Emergency Operating Procedures,” Rev. 2 (ML070100635);
- Regulatory Guide (RG) 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Rev. 4, (ML19101A395);
- RG 1.33, “Quality Assurance Program Requirements (Operation),” Rev. 3, (ML13109A458);
- RG 1.114, “Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit,” Rev. 3 (ML082380236);
- American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.1-2014, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants,”
- ANSI/ANS-3.2-2012, “Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants.”

11.1.1 Technical Evaluation

Submittal Overview

Enclosure 1 to the KU1 construction permit (CP) application, “General and Financial Information” (ML24088A059), section 1.1 identifies US SFR Owner, LLC (USO) as the applicant for the CP application and states, in part,

USO is a special purpose entity that was created on July 10, 2020, to serve as a vehicle for TerraPower’s US Department of Energy Advanced Reactor Demonstration Project bid for the Natrium™ Demonstration Project. USO is focused on the development and funding of Kemmerer Unit 1. USO will own 100 percent of Kemmerer Unit 1 and is responsible for the safe operation of the plant from pre-operational testing through commercial operation...TerraPower is the direct and controlling owner of USO. Pursuant to the terms of its agreement with USO, TerraPower is responsible for the administration and development of Kemmerer Unit 1 during the design and construction phase of the project, including support services in connection with filings by USO with any government authority, specifically the U.S. Nuclear Regulatory Commission.

Section 11.1 of the PSAR describes the organizational responsibilities of the applicant and TerraPower from design to commercial operation as follows:

- Design: TerraPower has overall ownership, control, oversight, and authority of the design.
- Construction: TerraPower partners with BPC [Bechtel Power Corporation] to perform the construction of Kemmerer Unit 1. BPC executes this responsibility through the construction organization. TerraPower coordinates with, and provides oversight of, BPC through the Natrium Project Organization, which continues to be the design authority.

- Pre-operational testing: USO conducts pre-operational testing, which includes fuel load and startup testing, through the Operating Organization. Overall technical support functions are provided through the oversight of TerraPower and USO.
- Commercial operation: USO executes responsibilities for the daily operation of Kemmerer Unit 1 through the Operating Organization and shares technical support functions with TerraPower.

PSAR section 11.1.1 describes the executive management organization for USO, which includes the President and Chief Executive Officer (CEO) of USO; and the following executives who report directly to the President and CEO of USO: the executive responsible for project execution - USO, the executive responsible for regulatory affairs - USO, the director responsible for quality - USO, and the Chief Nuclear Officer (CNO) - USO.

Section 11.1.4 of the PSAR states that USO is responsible for the safe operation of KU1 and establishes an operating organization prior to pre-operational testing, which will include the managers, supervisors, and staff needed to support the functional areas required for safe plant operations. The functions of the operating organization are described as including, in part, engineering, quality assurance, chemistry, radiation protection, training, maintenance, and operations. PSAR section 11.1.4 details the scope of responsibilities and reporting relationships associated with the following roles within the operating organization that will report to the CNO: the Plant General Manager - USO)) (who has overall responsibility for occupational and public radiation safety), the Director of Operations - USO (who oversees on-site operations and operations support personnel), the Director of Maintenance - USO (who oversees on-site maintenance personnel), the Station Sciences Manager - USO (who oversees the chemistry and radiation protection programs), the Director of Training - USO (who is responsible for the qualification and training of operations, maintenance, station sciences, and engineering personnel), the Security Manager - USO (whose responsibilities include physical security plan implementation, access authorization, and the fitness-for-duty (FFD) programs), Director of Nuclear Oversight - USO (who is responsible for the overall management and coordination of independent nuclear oversight to include audits, quality control, and employee concerns), and the Director of Organizational Effectiveness - USO (who is responsible for emergency planning, licensing, regulatory assurance, and occupational health). PSAR section 11.1.4 also states that personnel in the operating organization meet the qualification and experience requirements of ANSI/ANS-3.1-2014, as endorsed by RG 1.8.

Review Approach

The staff evaluated section 11.1 of the PSAR, as supplemented, using the guidance and criteria of DANU-ISG-2022-05. The objective of this review was to ensure that the application included sufficient information for the staff to understand organizational considerations. The same DANU-ISG-2022-05 criteria are intended to support reviews of 10 CFR Part 50 CP applications and operating license (OL) applications. DANU-ISG-2022-05 notes that the criteria in section 11.2, discussed in the following subsections, are broadly applicable to, among other things, both CP applications and OL applications and that “it may be necessary for the reviewer to adjust the review depth associated with individual criteria in light of the specific type of application under consideration.” In this case, the application is a CP application, therefore requiring less detail than an OL application. As such, the staff adjusted the review depth to appropriately apply the DANU-ISG-2022-05 criteria within the context of a PSAR review.

11.1.1.1 Assignment of Plant Operating Responsibilities

Pursuant to 10 CFR 50.34(a)(6), a PSAR must include a preliminary plan for the applicant's organization, training of personnel, and conduct of operations. DANU-ISG-2022-05 criterion 11.2(a) states a staff review should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the applicant describes the assignment of plant operating responsibilities, the reporting chain up through the CEO of the applicant, the proposed size of the regular plant staff, the functions and responsibilities of each major plant staff group, and the proposed shift crew complement, and the qualification.

Section 11.1.4 of the PSAR describes the operating organization and associated technical support positions. The staff reviewed the applicant's description of the organizational structure, reporting relationships, and functional responsibilities in the PSAR. While the proposed size of plant staff is not provided in numerical terms, the applicant has described the assignment of plant operating responsibilities, the reporting chain up to the CEO, the functions and responsibilities of each major plant staff group, the proposed shift crew complement, and personnel qualifications.

The staff determined the level of detail is appropriate for the CP stage. The staff's evaluation is consistent with the guidance in DANU-ISG-2022-05, criterion 11.2(a), which states that a safety finding may be reached if the applicant provides sufficient information on these organizational elements. Based on its technical judgement and the information provided, the staff determined that the applicant has met the guidance DANU-ISG-2022-05, criterion 11.2(a) and the requirements in 10 CFR 50.34(a)(6). The staff notes that further details regarding the applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualification requirements must be provided as part of an OL application in accordance with 10 CFR 50.34(b)(6).

11.1.1.2 Technical Qualification of Applicant

Pursuant to 10 CFR 50.34(a)(9), an applicant for a CP must include in the PSAR information regarding, "the technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter." Under 10 CFR 50.40(b), the U.S. Nuclear Regulatory Commission (NRC) must consider, in part, the technical qualifications of the applicant to engage in the proposed activities in accordance with the NRC's regulations, when determining that a CP will be issued. DANU-ISG-2022-05 instructs staff reviewers under criterion 11.2(b) to, in part, address whether the applicant is technically qualified. PSAR section 11.1.2 states that the executive responsible for project execution - USO leads the Natrium Project Organization, which is responsible for the design and construction of KU1. PSAR section 11.1 states that TerraPower has over 15 years of experience in designing and laboratory testing nuclear technologies, as well as a Board of Directors that has more than 50 years of combined nuclear executive experience. It is further stated that TerraPower is partnered with both GE-Vernova Hitachi Nuclear Energy Americas, LLC (GVH) (as a co-developer) and BPC (as an engineering and construction partner). GVH and BPC are described as each having over 60 years of experience in the design and construction of commercial nuclear reactors, with BPC noted as having participated in the design and construction of over 150 nuclear power projects.

Based upon the aggregate technical experience that will be available through the collaboration of USO, TerraPower, GVH, and BPC, in addition to the historical nuclear development

experience associated with GVH and BPC, the staff conclude that the applicant, through its agreements with TerraPower and organizational structure as described in the CP application, PSAR section 11.1, and the November 19, 2025 letter, is technically qualified to undertake the KU1 construction project, consistent with DANU-ISG-2022-05 criterion 11.2(b), 10 CFR 50.34(a)(9), and 10 CFR 50.40(b).

11.1.1.3 Key positions for ensuring the safe operation of the plant

DANU-ISG-2022-05 criterion 11.2(c) states that the reviewer should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the applicant includes the following information: the key positions for ensuring the safe operation of the plant are in the operating organization, consistent with the quality assurance program and ANSI/ANS-3.2, and on-shift personnel have the capability to provide initial facility response in the event of an emergency. Section 11.1.4 of the PSAR states that USO will establish an operating organization prior to pre-operational testing, with the PGM and Director of Operations having responsibility for that organization. The PSAR depicts the structure of the operating organization within the following chart from the PSAR:

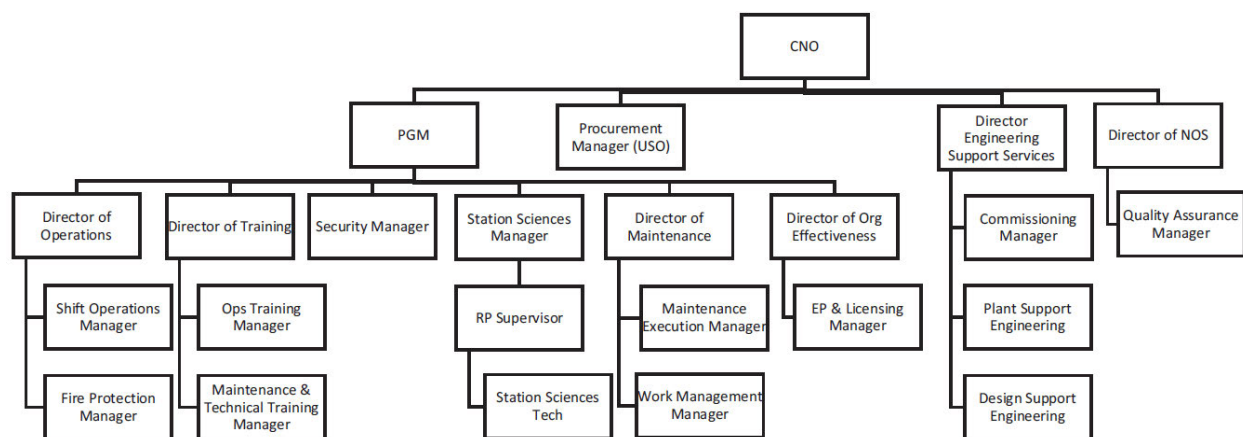


Figure 11.1-1: Operating Organization and Technical Support

Section 11.1.4 of the PSAR further states that the organization of managers, functional managers, supervisors, and staff necessary to support the engineering, quality assurance, chemistry, radiation protection, training, maintenance, and operations functions will be established before the commencement of pre-operational testing. The staff notes that the PSAR does not describe whether the operating organization will comply with either the organizational provisions of ANSI/ANS-3.2 or with an alternative approach. However, given that this criterion specifically addresses the facility organization during the operating phase and 10 CFR 50.34(a)(6) only requires a preliminary plan for the applicant's organization, training of personnel, and conduct of operations, the staff determined that the complete evaluation of this criterion, including any related commitment to ANSI/ANS-3.2, is not required until the review of an OL application. The staff notes that PSAR table 1.4-1 states that conformance with RG 1.33, which endorses with condition ANSI/ANS 3.2, will be addressed as part of an OL application.

11.1.1.4 Initial Test Program and Technical Support

DANU-ISG-2022-05 criterion 11.2(d) states, in part, the staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the application includes: adequate descriptions of groups and key positions responsible for implementing the initial test program (consistent with the quality assurance program and ANS 3.2) and for providing technical support for the operation of the facility. Section 11.1.4 of the PSAR states that operating and technical support organizations will be established prior to the start of pre-operational testing. The staff notes that the structure of this organization is depicted in SE figure 11.1-1, which illustrates reporting relationships beginning with the CNO and continuing through the managers of various functional areas of the organization. PSAR section 11.1.4 states that the operating organization will establish the operational phase quality assurance (QA) program, procedures for implementing the QA program, review and audit programs for activities affecting plant safety, and programs and "procedures for rules of practice as described in section 3.5 of ANSI/ANS 3.2-2012... that are applicable to Kemmerer Unit 1." Additionally, it is further described that TerraPower and USO will each provide the operating organization with technical support, with TerraPower supplying engineering support and USO providing technical support functions.

The staff notes that ANSI/ANS-3.2 addresses the operational phase of plant life, which is entered prior to initial fuel loading. As such, the staff determined that evaluation of the consistency of the test organization with either ANSI/ANS-3.2 or an alternative approach is not required until the future review of an OL application.

11.1.1.5 Plans for the Construction, Testing, and Operating Organizations

Under 10 CFR 50.34(a)(6), a PSAR must include a preliminary plan for the applicant's organization, training of personnel, and conduct of operations. DANU-ISG-2022-05 criterion 11.2(e) states that the staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the application includes plans for the construction, testing, and operating organizations that are consistent with the relevant portions of NUREG-0800 sections 13.1.1 through 13.1.3. These sections of NUREG-0800 provide guidance on, among other things, the staff's review of both CP and OL applications. To conduct its review, the staff reviewed the guidance in NUREG-0800 and determined which criteria in the referenced sections are relevant for a CP application review based on the information required by 10 CFR 50.34(a)(6). These criteria are evaluated within sections 11.1.1.5.1 through 11.1.1.5.6 of this safety evaluation (SE), as follows.

11.1.1.5.1 Integrated Management of Activities

NUREG-0800 section 13.1.1, criterion III.1(a)(ii) states that the application should describe how the organization provides for the integrated management of activities that support the construction, operation, and maintenance of the facility. Section 11.1.1 of the PSAR describes the integration of organizational management at the executive level and states that "executive management establishes expectations for quality, safety, and efficiency in plant operations and support activities..."

PSAR section 11.1.1.1 explains that the President and CEO of USO has overall responsibility for functions involving design, construction, and operation. This section states that design and construction responsibilities are assigned to the executive responsible for project execution,

with control of nuclear plant operation subsequently being assigned to the CNO prior to pre-operational testing. PSAR section 11.1.1.5 states that the director responsible for quality reports to the President and CEO of USO and also has responsibility for ensuring that design and construction of KU1 is consistent with the Quality Assurance Program Description (QAPD). The staff notes that currently the director responsible for quality is responsible for both the implementation of the QAPD for TerraPower and USO. As described in section 8.2 of this SE, the staff determined that a permit condition was necessary to ensure sufficient independence between USO and TerraPower to support meeting the requirements of Criterion I of Appendix B

Section 11.1.1.4 of the PSAR states that the CNO of USO "...has direct authority and responsibility for the management, control, and supervision of the nuclear power plant and for the execution of nuclear programs, policies, and decisions" and is assisted in these management activities by the PGM. PSAR section 11.1.4.1 states that the PGM is a direct report of the CNO with responsibility for the daily operations of KU1. The PSAR further states that both the USO Director of Operations and Director of Maintenance, in turn, report to the PGM. During the staff's audit (ML25302A443), the staff verified that the organizational reporting relationship between the CNO and PGM is consistent with that shown in the organizational charts in the PSAR. Further details on the flow of responsibility and decision making as it relates to the operating organization are not required to be provided until the OL stage. Based on the staff's review of PSAR sections 11.1.1.1 and 11.1.1.4, the staff determined that the applicant has described how the organization structure will provide for integrated management of construction, operation, and maintenance activities, consistent with criterion III.1(a)(ii) presented in NUREG-0800 section 13.1.1 and at a level appropriate for a PSAR.

11.1.1.5.2 Obtaining Personnel

NUREG-0800 section 13.1.1, criterion III.1(a)(iii) states that the applicant should describe how it will obtain personnel with sufficient experience to provide management and technical support for the facility. PSAR section 11.1.2 states that the Natrium Project Organization, led by the USO executive for project execution, is responsible for design and construction of KU1, and partners with BPC through the construction organization described in PSAR section 11.1.3. As stated in PSAR section 11.1, BPC has participated in the design and construction of over 150 nuclear power projects in 25 countries. Section 11.1.3 of the PSAR describes that BPC will perform the construction of KU1 and will utilize specialty contractors to support work that requires unique capabilities or knowledge. During the audit, the staff verified that BPC plans to use a direct hire with targeted specialty subcontracting approach to ensure appropriate staffing for construction. PSAR section 8.1.2 states that TerraPower has oversight of project partners through a project quality execution plan and procurement documents that flow quality and regulatory requirements down to the project partners and require flow down to the project's sub-contractors.

PSAR section 8.1.3 further states that the QAPD (ML23213A199) describes both the organization and its responsibilities during design and construction of KU1. The QAPD addresses the qualifications of personnel in technical and management roles. Based on the staff's review described above, as confirmed during an audit, the staff determined that the PSAR addresses how staffing will be obtained to support construction of the facility at a level of detail appropriate for a PSAR, and that further evaluation of how this will be addressed for the operating organization can be left until the review of an OL application.

11.1.1.5.3 Control of Major Contractors

NUREG-0800 section 13.1.1, criterion III.1(a)(vii) states that the application should describe how the organization will carry out its responsibilities to control major contractors and has committed to consider safety first, with due consideration of risk insights, in design and construction of the facility and during the transition from construction through testing to operation. As discussed above, through an agreement between USO and TerraPower, TerraPower provides services to USO for the design and construction of KU1. Section 11.1 of the PSAR states that TerraPower will partner with GVH as a co-developer and with BPC to perform construction, TerraPower remains responsible for the design and construction of KU1, with overall ownership, control, oversight, and authority of the design. Specifically, TerraPower is described as overseeing BPC via the Natrium Project Organization, which serves as the design authority. The CNO of USO is stated to have responsibility for plant management and execution of nuclear programs. The USO director responsible for quality, who at present is the same individual as the director responsible for quality of TerraPower, ensures the design and construction of KU1 is consistent with the QAPD. The staff included a permit condition to ensure that prior to construction, this role will be independent of other organizations supporting the design and construction of KU1 (see SE section 8.2.1). Additionally, the TerraPower executive responsible for design maintains responsibility for both configuration management and ensuring that technical and quality standards are met. Furthermore, the TerraPower Director of Safety and Engineering Analysis oversees development of the probabilistic risk assessment (PRA), defense-in-depth, and design basis event analyses and is supported in doing so by the Senior Manager, Safety Analysis and Risk who, in turn, manages the team performing the risk analyses. PSAR section 11.1.2.5 states that the team performing risk analysis activities is supplemented with external contractor personnel when necessary to ensure that the appropriate risk analysis experience is available.

Section 11.1.4 of the PSAR further discusses that, “USO is responsible for the safe operation of Kemmerer Unit 1 and establishes an Operating Organization prior to pre-operational testing.” Specifically, the CNO of USO will establish the organization of managers, supervisors, and staff needed to support the functional areas associated with plant operations. The staff determined that the applicant has described how the organization will, beginning in the design and construction phases and continuing through testing to operation, control major contractors while incorporating consideration of both safety and risk, consistent with criterion III.1(a)(vii) presented in NUREG-0800 section 13.1.1.

11.1.1.5.4 Safety Responsibility

NUREG-0800 section 13.1.1, criterion III.1(b) states that the applicant should identify the corporate officer responsible for nuclear activities and safety matters and should make a commitment that this individual will have no ancillary responsibilities that might detract attention from nuclear safety matters. Section 11.1.1.4 of the PSAR states that “the CNO has direct authority and responsibility for the management, control, and supervision of the nuclear power plant and for the execution of nuclear programs, policies, and decisions.” Additionally, section 11.1.1.4 of the PSAR also states that the “...CNO is the corporate executive with overall responsibility for nuclear safety and compliance. The CNO assumes these duties prior to the commencement of pre-operational testing.” The staff determined that the corporate officer responsible for nuclear activities and safety matters is identified within the application, consistent with criterion III.1(b) presented in NUREG-0800 section 13.1.1.

11.1.1.5.5 Construction Responsibilities

NUREG-0800 section 13.1.1, criterion III.1(c) states that the application should define design and construction responsibilities for applicant and vendor personnel, both in terms of numbers of personnel and necessary position qualifications and experience. As discussed above, through an agreement between USO and TerraPower, TerraPower provides services to USO for the design and construction of KU1. PSAR section 11.1.2 states that the Natrium Project Organization, led by the USO executive for project execution, is responsible for design and construction of KU1, and partners with BPC through the construction organization described in PSAR section 11.1.3. Oversight of the construction organization is discussed as being implemented by TerraPower in the PSAR by means of a project quality execution plan. It is stated that TerraPower oversees BPC via the Natrium Project Organization, which serves as the design authority. The USO director responsible for quality ensures the design and construction of KU1 is consistent with the QAPD. Additionally, the TerraPower executive responsible for design maintains responsibility for both configuration management and ensuring that technical and quality standards are met.

PSAR section 11.1.3 provides a description of the construction organization and states that the BPC Project Manager leads the construction organization. This section also states that the construction of KU1 itself is performed by BPC, with BPC being supported by contractors in areas which require unique capabilities or knowledge. As stated in PSAR section 11.1, BPC has participated in the design and construction of over 150 nuclear power projects in 25 countries. The structure of the construction organization is depicted in the following figure from the PSAR:

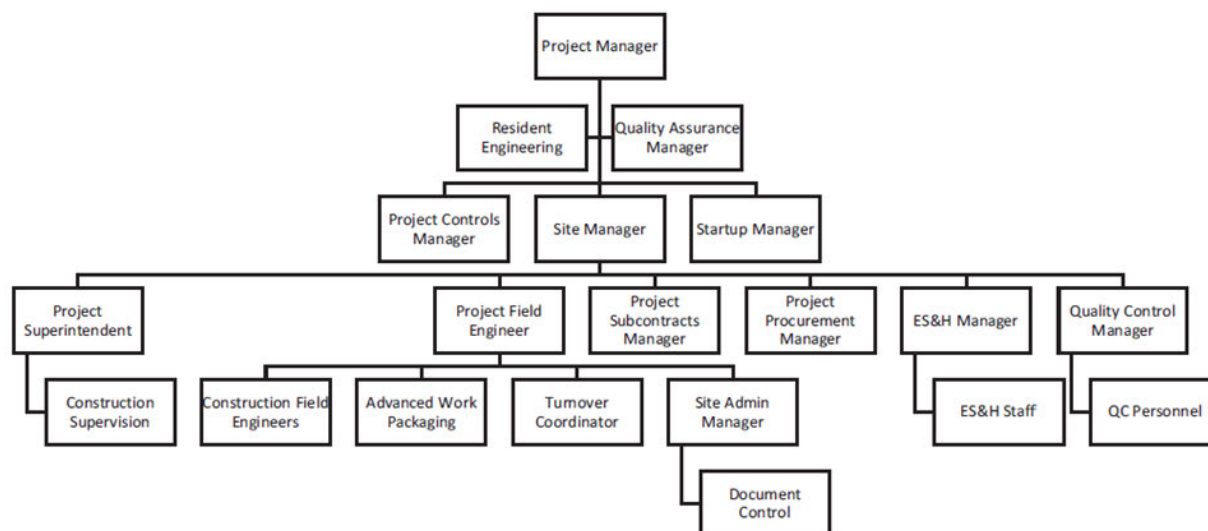


Figure 11.1-2: Construction Organization

PSAR section 11.1.3 provides an overview of certain BPC positions that report to the Project Manager, which include the Site Manager, Project Quality Manager, and Project Field Engineer. The Site Manager is described as being responsible for the management of the total site, including oversight of environmental safety, security, emergency response, incident reporting, quality, and subcontract management. The Project Quality Manager is stated to be responsible for implementing the project quality plan and "...supervises a team of personnel that performs oversight of construction activities, including audits, to ensure construction is performed in accordance with the BPC QA program." Lastly, the Project Field Engineer is stated to be the

primary construction interface across for BPC and GVH engineering groups, as well as for the Natrium Project Organization. The Project Field Engineer is further described as leading a team of field engineers who are accountable for quality and technical compliance.

PSAR section 8.1.3 states that the QAPD describes the QA organization and responsibilities during design and construction of KU1. The staff notes that the QAPD addresses the qualifications of personnel in technical and management roles. While the PSAR does not describe the specific numbers of personnel to be included within the construction organization, during the audit, the staff verified that BPC plans to use a direct hire with targeted specialty subcontracting approach to ensure appropriate staffing for construction. Based on its review described above, the staff determined that the description of the organization, key personnel, reporting responsibilities, and construction responsibilities in general in the PSAR was sufficient and that specific number of personnel may vary during the design and construction activities and are not necessary to be included in the PSAR. The staff determined that the PSAR adequately defines applicant and vendor responsibilities for design and construction, as well as key personnel roles, qualifications, and experience associated with design and construction, consistent with criterion III.1(c) presented in NUREG-0800 section 13.1.1.

11.1.1.5.6 Responsibilities for Human Factors Engineering

The staff review guidance in NUREG-0800 section 13.1.1, criterion III.1(d) states that the application should clearly define management and organizational responsibilities to address human factors engineering (HFE) considerations. Section 11.2.1.5 of the PSAR states that licensing topical report (TR) NAT-2965-A, "Human Factors Engineering Program Plan and Methodologies," Rev. 1 (ML25211A232), is incorporated by reference and describes the HFE organization. The staff's evaluation of the TR found, in part, that the described management and organizational responsibilities were adequate to address HFE considerations. It should be noted that PSAR section 11.2 is addressed separately under section 11.2 of this SE. The staff determined that management and organizational responsibilities are clearly defined as they relate to the addressing of HFE considerations, consistent with criterion III.1(d) presented in NUREG-0800 section 13.1.1.

11.1.1.6 Plans for Conduct of Operations

NUREG-0800 section 13.1.1, criterion III.1(d) states that the application should clearly define management and organizational responsibilities to address HFE considerations. The staff's evaluation of NAT-2965-A determined, in part, that the described management and organizational responsibilities were adequate to address HFE considerations. Based on the staff's analysis in the SE approving NAT-2965-A, which is incorporated by reference in this CP application, the staff determined that management and organizational responsibilities are clearly defined as they relate to the addressing of HFE considerations, consistent with criterion III.1(d) presented in NUREG-0800 section 13.1.1.

11.1.1.7 Education and Experience Requirements for Key Personnel

DANU-ISG-2022-05 criterion 11.2(g) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the applicant has proposed education and experience requirements for key personnel that conform to the qualifications and experience guidance endorsed by RG 1.8, which, itself, endorses with clarifications and exceptions ANSI/ANS 3.1. These should include provisions for individual experience which may not be entirely applicable to be weighed against

the requirements of the position. This criterion also notes that any alternative to, or exceptions from, this guidance should be adequately justified.

Section 11.1.1.4 of the PSAR states the CNO of USO will ensure that the collective experience of the KU1 senior leadership team will meet ANSI/ANS-3.1-2014, as endorsed by RG 1.8. PSAR section 11.1.4 further states that USO will establish an operating organization prior to pre-operational testing, the personnel of which will meet the qualification and experience requirements of ANSI/ANS 3.1-2014. Additionally, PSAR section 11.1.5.1 states that the qualifications of managers, supervisors, operators, technicians, and other support staff will also meet ANSI/ANS 3.1-2014. The staff determined that the PSAR proposes education and experience requirements for key personnel that are consistent with the guidance of RG 1.8 and therefore that the application is consistent with criterion 11.2(g).

11.1.1.8 Proposed Operating Organization

Under 10 CFR 50.34(a)(6), a PSAR must include a preliminary plan for the applicant's organization. DANU-ISG-2022-05 criterion 11.2(h) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if an applicant's proposed organization conforms to the guidance of RG 1.33, or the applicant has provided justification for an alternative. The staff notes that RG 1.33 endorses ANSI/ANS 3.2. Section 11.1.4 of the PSAR states that USO will establish an operating organization prior to pre-operational testing. The staff note that the PSAR does not describe whether the operating organization will comply with the organizational provisions of ANSI/ANS 3.2 or with an alternative approach. The staff notes that RG 1.33 specifically applies to applicants for or holders of OLs, and PSAR table 1.4-1 accordingly states that conformance with RG 1.33 will be addressed as part of an OL application. As such, the staff determined that the PSAR contains a preliminary plan for the operating organization, and that the final evaluation of this criterion, including any related commitment to ANSI/ANS 3.2, can reasonably be left until the review of an OL application.

11.1.1.9 Compliance With Minimum Staffing Requirements

Under 10 CFR 50.34(a)(6), a PSAR must include a preliminary plan for the applicant's organization. DANU-ISG-2022-05 criterion 11.2(i) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if an adequate number of licensed operators will be available at all required times to satisfy the minimum staffing requirements of 10 CFR 50.54(m), or the applicant has provided justification for an exemption.

Section 11.1.4 of the PSAR describes that the CNO of USO will establish the organization of managers, supervisors, and staff necessary to support operations prior to the commencement of pre-operational testing. PSAR section 11.1.5.3 states that the rules of practice for control room activities, as described by administrative procedures, incorporate the guidance of RG 1.114. This includes both the conduct and relief requirements for the control room supervisor and operator at the controls. Additionally, it is further stated that manipulation of reactivity controls will be limited to being performed only by licensed operators, except for training situations as allowed by regulation. Additionally, PSAR section 11.1.5.3 states that "[a]pparatus and mechanisms other than controls which may affect reactivity or power level of the reactor shall be operated only with the knowledge and consent of the operator or the Control Room Supervisor present at the controls." The staff notes that section 11.1.5.3 further states

that “the staffing requirements in 10 CFR 50.54(k) and (m) will be provided at the operating license stage.”

The staff determined that the applicant has described preliminary plans for licensed operator staffing to the extent warranted for a PSAR as required under 10 CFR 50.34(a)(6) and that further review of compliance with the staffing requirements of 50.54(m) would occur as part of the staff’s review of a future OL application.

11.1.1.10 Engineering Expertise on Shift

DANU-ISG-2022-05 criterion 11.2(j) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant’s organization and human-system interfaces are met if the engineering expertise on-shift will be consistent with the Commission policy (50 FR 43621) and within the guidelines of Three Mile Island Action Plan Item I.A.1.1, “Shift Technical Advisor,” of NUREG-0737, “Clarification of TMI Action Plan Requirements” (ML083040141), or the applicant provided justification for an alternative. Section 11.2 of the PSAR incorporates by reference, in its entirety NAT-2965-A. In the staff’s evaluation of NAT-2965-A, the staff noted that USO has indicated an intent to request an exemption (in addition to providing the associated technical justification) in conjunction with an OL application to omit the shift technical advisor (STA) from the Natrium staffing complement; this is the subject of a limitation and condition included with the staff’s evaluation of NAT-2965-A. The staff determined that review of this technical justification for omission of the STA, as well as consideration of any associated exemption request, can occur during the review of the OL application because the systems approach to training requirements in 10 CFR 50.120, “Training and qualification of nuclear power plant personnel,” apply to applicants for and holders of OLs.

11.1.1.11 Role and Function of Vendors During Design and Construction

DANU-ISG-2022-05 criterion 11.2(k) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant’s organization and human-system interfaces are met if the applicant has described the role and function of the architect-engineer and the nuclear steam supply system vendors during design and construction in a manner consistent with NUREG-0800, section 13.1.1, as well as the organizational controls over the project-related activities of the architect-engineer and nuclear reactor vendors, including preservation of documentation. This criterion also notes that this information is to address 10 CFR 50.34(f)(3)(vii). The Natrium Project Organization is identified as being the design authority; this organization is noted in section 11.1.2 as being led by the USO executive responsible for project execution. PSAR section 11.1 states that TerraPower partners with and oversees BPC to perform the construction of KU1. PSAR section 11.1.3 specifically describes that BPC implements management systems for quality, health, safety and environmental, cost and program metrics, and monitoring across the project. Section 11.1.3 further states that “BPC is responsible for the control and preservation of documentation within the construction organization. BPC has processes for document turnover with TerraPower.” Based on the staff’s review as described above, the staff determined that the applicant has described the roles of the architect-engineer and the nuclear steam supply system vendors, organizational controls over project-related activities, and provisions for the preservation of documentation, consistent with criterion 11.2(k) in DANU-ISG-2022-05.

11.1.1.12 Reporting Responsibilities and Authorities

Under 10 CFR 50.34(a)(6), a PSAR must include a preliminary plan for the applicant's organization. DANU-ISG-2022-05 criterion 11.2(l) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the applicant has identified and describe the reporting responsibilities and authorities in the functional areas of radiation protection/health physics, QA, and training. Additionally, this criterion states that the associated reporting responsibilities and authorities should ensure independence from normal operating pressures. Section 11.1.4 of the PSAR defines the responsibilities and authorities of the USO Station Sciences Manager, Director of Training, Director of Nuclear Oversight, and QA Manager. The Station Sciences Manager is described as overseeing management of both the chemistry and radiation protection programs. PSAR section 11.1.4.1 states that Director of Training is responsible for the qualification and continuing training of personnel within the areas of operations, maintenance, station sciences, and engineering, in addition to ensuring the delivery of general nuclear plant training to plant personnel. The Director of Nuclear Oversight is identified as having responsibility for providing independent nuclear oversight, which is described as including audits, quality control, and employee concerns. Additionally, the QA Manager is described as being responsible for supervising employees who perform functions that include quality control inspections, non-destructive examination inspections, and QA audits.

PSAR figure 11.1-4 indicates that the Director of Training and Station Sciences Manager (to whom the Radiation Protection Supervisor reports) each report directly to the PGM, in a manner that is independent from the Director of Operations. The PGM, in turn, reports directly to the CNO. Furthermore, this figure also depicts that the Director of Nuclear Oversight (to whom QA Manager reports) reports directly to the CNO in a manner that is separate and independent from the PGM. Based on the discussion above, the staff determined that applicant has adequately described both reporting responsibilities and authorities for the areas of radiation protection, QA, and training and, furthermore, has presented an organizational reporting structure that should reasonably provide independence from normal operating pressures, consistent with criterion 11.2(l) presented in DANU-ISG-2022-05.

11.1.1.13 Responsibilities of the Operating Organization

Under 10 CFR 50.34(a)(6), a PSAR must include a preliminary plan for the applicant's organization. DANU-ISG-2022-05 criterion 11.2(m) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the applicant has defined the responsibilities of the operating organization related to activities important to the safe operation and maintenance of the facility. It also states that functional areas should be separately supervised and managed. Section 11.1.4 of the PSAR states that USO will establish an operating organization which includes the managers, functional managers, supervisors, and staff necessary to support, among other things, the engineering, QA, radiation protection, maintenance, and operations functions before pre-operational testing begins. The structure of this organization is depicted by PSAR figure 11.1-4 which shows that each of these functional areas is under the cognizance of a separate manager or director. PSAR section 11.1.4 discusses the scope of responsibilities that are assigned across these various functional areas of the operating organization, including for on-site operations, plant maintenance, work control, chemistry and radiation protection programs, quality control inspections, non-destructive examination inspections, QA audit functions, and engineering activities related to operations and maintenance. The staff determined that the applicant has defined the responsibilities of the operating organization as they relate to safe operation and maintenance of the facility, with key organizational functions

being separately supervised and managed, consistent with criterion 11.2(m) presented in DANU-ISG-2022-05.

11.1.1.14 Managerial Depth

DANU-ISG-2022-05 criterion 11.2(n) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if sufficient managerial depth is available to provide qualified backup, consistent with the QA program and ANS 3.2, for overall station operation in the event of temporary unexpected contingencies. This criterion specifically says this is to address 10 CFR 50.34(b)(6)(v), which does not apply to CP applications. As such, the staff determined that it was not necessary to evaluate whether the application was consistent with this criterion at this time. Should the applicant apply for an OL, the staff will evaluate whether the OL application is consistent with this criterion and, therefore, complies with 10 CFR 50.34(b)(6)(v), in the OL review. To the extent that the CP application included information to address this criterion, the staff takes no position on the acceptability of that information for the purpose of satisfying 10 CFR 50.34(b)(6)(v) at this time.

11.1.1.15 Plans for Preventing Personnel Impairment from Fatigue

DANU-ISG-2022-05 criterion 11.2(o) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the numbers of licensed and non-licensed personnel subject to 10 CFR 26.205 will be sufficient to allow shift schedules that prevent personnel impairment from fatigue due to the duration, frequency, or sequencing of successive shifts. The criterion says this is to address 10 CFR 26.205(c). Under 10 CFR 26.3(c), CP holders must comply with the requirements of 10 CFR Part 26, "Fitness for Duty Programs," except for Part 26 Subpart I, "Managing Fatigue," and must comply with Subpart I by no later than the receipt of special nuclear material in the form of fuel assemblies. 10 CFR 26.205 is located in Subpart I. Section 11.6.1 of the PSAR states, in part, the following:

Compliance with the requirements of 10 CFR Part 26 Subpart I will be achieved prior to the receipt of special nuclear material in the form of fuel assemblies. Staffing levels sufficient to meet the provisions of 10 CFR Part 26 Subpart I for individuals identified in 10 CFR 26.4(a)(1) through (a)(5) will be established prior to the receipt of special nuclear material in the form of fuel assemblies.

Because 10 CFR 26.205 does not apply at this time and the applicant states it will comply with 10 CFR 26.205 before receipt of special nuclear material, the staff determined that the applicant has, to an extent appropriate for a PSAR, described how the numbers of personnel subject to 10 CFR 26.205 will be sufficient to prevent personnel impairment from fatigue. Thus, as adjusted for the level of review appropriate for a CP application, the staff determined that this application is consistent with criterion 11.2(o) presented in DANU-ISG-2022-05.

11.1.1.16 Training Program for Licensed Operators

DANU-ISG-2022-05 criterion 11.2(p) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the training program for licensed operators meet the requirements of 10 CFR Part 55, "Operators' Licenses." Section 11.1.5.2 of the PSAR states that the operator training and requalification plan will be developed and implemented in accordance with 10 CFR

Part 55 and that a description of the operator requalification plan will be provided at the OL application stage. The PSAR further states that the operator qualification process will include both a comprehensive written exam and an operating test, consistent with 10 CFR Part 55 for the relevant type of application. Under 10 CFR 55.2, "Scope," the regulations in Part 55 apply to (1) any individual who manipulates the controls of any utilization facility licensed under Parts 50, 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," or 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants;" (2) any individual designated by a facility licensee to be responsible for directing any license activity of a licensed operator; and (3) and facility license. A CP does not authorize possession or use of radioactive material without receiving the relevant materials licenses and does not authorize operation of the facility. As such, KU1 will not have an OL and it will, therefore, have no individuals who manipulate the controls of a utilization facility or are designated by a facility license to be responsible for directed any licensed activity of a licensed operator.

Consequently, the staff determined that the applicant has described the licensed operator training program to the extent warranted for a PSAR and that further review of the initial and continuing licensed operator programs can be completed in conjunction with the review of a future OL application. Thus, the staff determined that the application is consistent with criterion 11.2(p) presented in DANU-ISG-2022-05.

11.1.1.17 Training Program for Non-licensed Personnel

DANU-ISG-2022-05 criterion 11.2(q) states that staff should be able to reach a safety finding and conclude that the requirements governing the applicant's organization and human-system interfaces are met if the training program for non-licensed nuclear plant personnel is consistent with the requirements of 10 CFR 50.120(b)(2) and (3).

Section 11.1.5.2 of the PSAR states that USO will comply with the requirements of 10 CFR 50.120. Section 11.1.5.2 of the PSAR states that "[t]raining programs will be in place that meet the requirements of 10 CFR 50.120 within 18 months prior to fuel load." During the audit (ML25302A443), USO further clarified that it plans to seek an exemption as part of the OL application to omit the STA. Additionally, USO indicated that should plans change such that training programs that meet the requirements of 50.120 will not be in place 18 months prior to fuel load, an exemption from the relevant regulation will be requested.

As discussed in DANU-ISG-2022-05, the staff can adjust the review depth to the level appropriate for the relevant type of application. Under 10 CFR 50.120(a), the requirements of 10 CFR 50.120 apply to applicants for, and holders of an OL issued under Part 50 for a nuclear power plant of the type specified in 10 CFR 50.21(b) or 50.22, "Class 103 licenses; for commercial and industrial facilities." As the requirements of 10 CFR 50.120 do not apply to CP applicants or holders, the staff determined that the applicant's plans to develop the relevant training program in the future were sufficient. Should USO apply for an OL, the staff will evaluate the training program at that stage. Therefore, the staff determined that the application is consistent with criterion 11.2(q) presented in DANU-ISG-2022-05.

11.1.4 Conclusion

Based on the evaluations described above, the staff conclude that the information in PSAR section 11.1 is consistent with applicable guidance, and therefore meets relevant portions of the regulatory requirements in 10 CFR 50.34(a)(6), 50.34(a)(9), 50.35, "Issuance of construction permits," and 50.40 for the issuance of a CP.

11.2 Human Factors Engineering

Section 11.2 of the KU1 PSAR provides the HFE program and associated technical information for the facility. The HFE input to the CP application describes the program of HFE activities and analyses informing the design of SSCs based on the definition of the set of users and defined scope of the application to the Nuclear Island design, operational modes, and design stages, with focus on important human actions.

The applicable regulatory requirements for the evaluation of HFE are as follows:

- 10 CFR 50.34(a);
- 10 CFR 50.34(a)(3); and
- 10 CFR 50.34(a)(9).

The applicable guidance for the evaluation of HFE are as follows:

- DANU-ISG-2022-05, "Advanced Reactor Content of Application Project Chapter 11, 'Organization and Human-System Considerations' Interim Staff Guidance," (ML23277A143)
- NUREG-0711, "Human Factors Engineering Program Review Model," Rev. 3, (ML12324A013)
- RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Rev. 0, (ML17325A611)

In accordance with this guidance, the following requirements are applicable to the review of the KU1 CP application:

- Principal Design Criteria (PDC) 19, "Control Room" (required by 10 CFR 50.34(a)(3))

Definitions of the PDCs are included in PSAR section 5.3. The staff identified PDC 19 as having relevance to HFE and states, in part, the following:

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions... Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during shutdown, and (2) with a potential capability for subsequent safe shutdown of the reactor through the use of suitable procedures.

11.2.1 Technical Evaluation

The staff evaluated section 11.2 of the PSAR using the guidance and criteria of DANU-ISG-2022-05. The objective of this review was to ensure that the application includes sufficient information, of a scope and level of depth appropriate for a CP application/PSAR, to

understand human-system interface considerations. As the same DANU-ISG-2022-05 criteria are intended to support reviews of 10 CFR Part 50 CP applications and, the staff applied judgement to adapt and apply the DANU-ISG-2022-05 criteria within the context of a PSAR review.

11.2.1.1 Inclusion of Appropriate PDC

Section 11.1.3 of DANU-ISG-2022-05 describes that an application should include proposed PDC for the design, fabrication, construction, testing, and quality of those SSCs whose satisfactory performance depends on HFE. Section 5.3 of the PSAR describes PDC 19, which addresses the plant locations that are of relevance within the context of this guidance, specifically the Natrium control room and remote shutdown locations. PDC 19 was previously evaluated by the staff in the TR NATD-LIC-RPRT-0002, "Principal Design Criteria for The Natrium Advanced Reactor," Rev. 1 (ML24283A066). The staff determined that the PSAR includes a PDC that addresses the design, fabrication, construction, testing, and quality of both the control room and remote shutdown locations, consistent with criterion presented in section 11.1.3 of DANU-ISG-2022-05.

11.2.1.2 Application of HFE in Meeting PDC

DANU-ISG-2022-05 criterion 11.2(b) of the staff's review guidance states, in part, that an application should describe how the HFE and planned human-system interactions satisfy the design-specific PDC related to HFE and human-system interactions. Section 11.1.3 of DANU-ISG-2022-05 describes that an HFE program description should cover the HFE design process, the HFE final design, its implementation, and ongoing performance monitoring, to the extent relevant to the application. PDC 19 addresses the Natrium control room and remote shutdown locations, which are relevant from an HFE standpoint.

Section 11.2 of the PSAR states that NAT-2965-A is incorporated by reference into the PSAR. The staff evaluation of NAT-2965-A determined that, subject to the conditions and limitations detailed within the staff's evaluation, the Human Factors Engineering Program Plan (HFEPP) (ML25211A232) contained in NAT-2965-A presents an overall methodology that is generally consistent with a state-of-the-art approach to HFE, as described by NUREG-0711. In particular, the staff concluded that the HFEPP presents HFE implementation plans which are adequate to address the relevant HFE elements and should reasonably lead to the development of a design that incorporates appropriate HFE principles. The staff notes that the information provided in section 11.2 of the PSAR is consistent with the HFEPP, as previously evaluated by the staff. Additionally, as further noted in the staff's evaluation of the HFEPP, making a finding on the overall HFE acceptability of a design produced via implementation of the HFEPP will require future staff review of the information contained in the Results Summary Reports that will be developed via implementation of the HFEPP, as well as that of the associated OL application.

The staff determined that the application, including NAT-2965-A, that is incorporated by reference in the application describes an HFE program to an extent appropriate for a CP application, including how HFE will be applied to satisfy PDC 19. However, the staff also notes that a complete evaluation of the adequacy of HFE within the KU1 design, including any associated finding related to 10 CFR 50.34(f)(2)(iii), must be left until the future OL application, relevant HFE RSRs, and other items specified within the limitations and conditions in the staff's SE of the HFEPP have been reviewed by the staff.

11.2.1.3 Technologically Relevant HFE-related Requirements of 50.34(f)

Section 11.1.3 of DANU-ISG-2022-05 states that applications should describe how the design will satisfy any requirements of 10 CFR 50.34(f) that are relevant to the technology, with all Part 50 applicants being expected to both address 50.34(f)(2)(iii) and describe how the design will satisfy any other requirements of 50.34(f) that are relevant to the technology. PSAR table 1.4-4 identifies, in part, the following regulations of 10 CFR 50.34(f) as having applicability to Natrium: (f)(2)(ii) for procedures, (f)(2)(iii) for HFE in the control room, (f)(2)(iv) for safety parameters display system, (f)(2)(v) for status of safety systems, (f)(2)(xi) for relief and safety valve indication, (f)(2)(xvii) for containment related indications, (f)(2)(xviii) for core cooling indications, (f)(2)(xix) for instrumentation to monitor post-accident plant conditions, (f)(2)(xxvi) for leakage control and detection, (f)(2)(xxvii) for radiation monitoring, and (f)(3)(i) for operating experience. Section 11.2 of the PSAR incorporates by reference, in its entirety, NAT-2965-A. In the staff's evaluation of NAT-2965-A, the staff determined that, subject to the limitations and conditions described within that evaluation, TerraPower's HFEPP addresses the incorporation of technologically relevant, post-TMI requirements. The staff determined that the applicant has described how the design would satisfy the technically relevant requirements of 10 CFR 50.34(f) to an extent that is reasonable for a PSAR and that further evaluation of the completed implementation of these items can occur as part of the OL application.

11.2.2 Conclusion

Based on the evaluations described above, the staff conclude that the information in PSAR section 11.2.3 is sufficient and meets the applicable guidance and regulatory requirements identified in this chapter for issuance of a CP in accordance with 10 CFR 50.34(a)(3), 10 CFR 50.34(a)(9), 10 CFR 50.35, and 10 CFR 50.40.

11.3 Emergency Planning

Section 11.3 of the PSAR provides a description of the preliminary plans for coping with emergencies and states that the KU1 emergency plan will be provided when the OL is submitted to the NRC. The PSAR further states that a performance-based emergency preparedness program will be developed to meet the requirements of 10 CFR 50.160, "Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities," and the associated RG 1.242, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," Rev. 0 (ML23226A036). The PSAR does not state that the applicant plans to deviate from the 10 CFR 50.160 requirements or the guidance of RG 1.242.

The applicable regulatory requirements for the evaluation of emergency planning and preparedness are as follows:

- 10 CFR 50.33(g)(2);
- 10 CFR 50.34(a)(10)(ii);
- 10 CFR 50.35;
- 10 CFR 50.40; and

- 10 CFR 50.160.

The applicable guidance documents for the evaluation of emergency planning and preparedness are as follows:

- RG 1.242;
- DANU-ISG-2022-01; and
- NUREG-0800, section 13.3, “Emergency Planning.” Rev. 3 (ML063410307)

The staff notably considered the following specific guidance to determine and make a safety finding that the requisite level of preliminary emergency planning design information was provided in the PSAR for the CP application to meet the minimum Emergency Protection (EP) information criterion in 10 CFR 50.34(a)(10):

- RG 1.242, section C, “Staff Regulatory Guidance,” guidance item 2.c, which states that the analysis of the emergency planning zone (EPZ) size required in 10 CFR 50.33(g)(2) may be preliminary for CPs consistent with the preliminary design of the facility as described in the PSAR. It further states that an applicant should describe its EPZ sizing analysis methodology and the preliminary analysis results, including the radius of the plume exposure pathway EPZ.
- NUREG-0800, section 13.3, section III, “Review Procedures,” subsection “General Review Procedures,” guidance item 3, provides guidance to the staff on the extent of a CP review, and section IV, “Evaluation Findings” guidance item 1, “Construction Permit,” which provides staff evaluation guidance on a CP preliminary plan’s acceptability and guidance on sufficient information for a PSAR for staff findings and conclusions.
- DANU-ISG-2022-01, Appendix C, “Construction Permit Guidance,” Section titled “Detailed Non-Light Water Reactor Construction Permit Guidance,” Staff Review Guidance, provides staff guidance that addresses the minimum information for a CP application for the staff to make the findings for a CP under 10 CFR 50.35(a), for CP applicants that have not supplied all of the technical information required to support the issuance of a CP.

Applicable guidance for the review of this section of the emergency plan can be found in the base document of DANU-ISG-2022-01 and in Appendix C, “Construction Permit Guidance,” of this document. As noted above the applicable regulation is 10 CFR 50.160, and the applicable guidance in DANU-ISG-2022-01 Appendix C notes the following:

“For CP applicants that choose to comply with the new alternative emergency planning requirements in 10 CFR 50.160, which allows a scalable approach for determining the size of the plume exposure pathway EPZ, the application must also contain the analysis used to determine whether the criteria in 10 CFR 50.33(g)(2)(i)(A) and (B) are met and, if they are met, the size of the plume exposure pathway EPZ. Specifically, § 50.34(a)(10) requires applicants for CPs to describe within the PSAR their preliminary plans for coping with emergencies. As stated in RG 1.242, the analysis of the EPZ size required in 10 CFR 50.33(g)(2) may be preliminary for construction permits consistent with the

preliminary design of the facility as described in the preliminary safety analysis report. As such the CP applicant should describe its EPZ sizing analysis methodology and the preliminary analysis results, including the radius of the plume exposure pathway EPZ.”

11.3.1 Technical Evaluation

The staff reviewed the KU1 facility PSAR emergency preparedness preliminary plan design descriptions and information for coping with emergencies and documented, for each applicable criterion, the PSAR emergency preparedness preliminary plan design information that meets the requisite PSAR minimum information for a CP application requirement in 10 CFR 50.34(a)(10). The staff’s review and evaluation focus on the sufficiency and acceptability of the preliminary emergency preparedness plan design descriptions and information included in the KU1 CP application PSAR to meet 10 CFR 50.160.

The following sections of this chapter of the staff SE document the staff’s findings and determinations of the requisite minimum emergency preparedness design information to be included in a CP application PSAR, for the KU1 Natrium reactor technology and plant design. The staff notes that other, or non-similar reactor designs, may produce different PSAR emergency preparedness minimum design information determinations and findings.

11.3.1.1 Facility Site and Plume Exposure Pathway (PEP) EPZ Boundary and Physical Characteristics Descriptions, and EPZ Size Determination Analysis Evaluation

The applicant is required to submit an analysis under 10 CFR 50.33(g)(2) that provides the technical basis for determining whether the criteria in 10 CFR 50.33(g)(2)(i)(A) and (B) are met and, if they are met, the size of the proposed PEP EPZ. The analysis of the EPZ size required in 10 CFR 50.33(g)(2) may be preliminary for CPs consistent with the preliminary design of the facility as described in the PSAR. The applicant should describe its EPZ sizing analysis methodology and the preliminary analysis results, including the radius of the PEP EPZ

The following SE sections document the staff’s review, evaluation, and regulatory findings of requisite KU1 emergency preparedness preliminary plan CP application PSAR design information.

11.3.1.1.1 KU1 Facility Site Layout and PEP EPZ, Boundary and Physical Characteristics General Description Evaluation (10 CFR 50.160(b)(3))

The staff used the guidance of RG 1.242, section C, guidance evaluation items 2.c, 6.a(1), and Appendix A, guidance item A-1.a, to review and evaluate the KU1 emergency preparedness preliminary plan design descriptions and information to determine and make regulatory findings of the requisite minimum design information to be included in a CP application PSAR.

The staff found that a general description of the KU1 facility is provided in PSAR section 1.2, and section 2.2.1. PSAR figure 2.2-20 displays the location of facility structures and site layout, and PSAR figure 2.2-21 provides a site area map. These PSAR figures also identify the site and property boundaries, adjacent site transportation routes, and the KU1 facility exclusion area boundary (EAB) and low population zone.

PSAR section 2.2.2.1, and PSAR section 2.2.2.3, state that the applicant has authority to control all activities occurring within the EAB including exclusion and removal of personnel and property

from the area, that there are no public roads within the EAB, and that there are no federal, State or County roads that traverse the exclusion area. The EAB is defined by a 400-meter (approximately 0.25 mile) radius from the KU1 Reactor Building (RXB) center point.

PSAR section 11.3.3.1, states that the preliminary PEP EPZ sizing analysis technical report, TP-LIC-RPT-0012, "Preliminary Emergency Planning Zone Determination Analysis, Technical Report," (ML24088A087), is referenced in PSAR table 1.4-3, and is incorporated by reference into the PSAR. TP-LIC-RPT-0012 was reviewed by the staff as part of the KU1 CP application as discussed below in SE section 11.3.1.1.2. TP-LIC-RPT-0012, section 2, and section 9, state that the proposed boundary of the PEP EPZ can be set at the EAB (a 400-meter radius from the RXB center point). In addition, TP-LIC-RPT-0012, section 1.1, states that the preliminary PEP EPZ sizing methodology addresses the consideration of the area in which predetermined, prompt protective measures are necessary. The physical layout and boundary of the KU1 site proposed PEP EPZ (i.e., EAB) is shown in PSAR figure 2.2-21.

Based on the staff's review and evaluation of the KU1 proposed site and PEP EPZ boundary and physical characteristic design descriptions documented above, the staff determined that the PSAR preliminary plans describing the KU1 facility site and PEP EPZ boundary and physical characteristics contain sufficient information to address the minimum information necessary in a CP application described in DANU-ISG-2022-01 and for a performance-based emergency plan consistent with the guidance in RG 1.242. Therefore, the staff finds that the KU1 site and preliminary PEP EPZ boundary and physical characteristics PSAR design descriptions meet the regulatory requirement of 10 CFR 50.34(a)(10)(ii) with respect to the PSAR content and 10 CFR 50.160(b) with respect to the content of the emergency plan.

11.3.1.1.2 KU1 Facility PEP EPZ Size Determination Analysis Evaluation (10 CFR 50.33(g)(2)(i))

10 CFR 50.160(b)(3) requires applicants to submit as part of the application, the analysis used to determine whether the criteria in 10 CFR 50.33(g)(2)(i)(A)-(B) are met and provide the technical basis justifying the proposed PEP EPZ size. The CP applicant should describe its EPZ sizing analysis methodology and the preliminary analysis results, including the radius of the PEP EPZ. This section of the SER will document the staff's evaluation of the KU1 preliminary PEP EPZ size determination analysis compliance to the PEP EPZ size determination requirements of 10 CFR 50.33(g)(2)(i) and 10 CFR 50.160(b)(3).

PSAR section 11.3.3.1, states that the guidance in RG 1.242, Appendix A, was used to determine the KU1 PEP EPZ size. TP-LIC-RPT-0012 states that the boundary of the KU1 PEP EPZ can be set at the EAB for the KU1 site. TP-LIC-RPT-0012, section 1.1, "Purpose," states that the TerraPower TR, NAT-3056, "Plume Exposure Pathway Emergency Planning Zone Sizing Methodology," (ML25104A001) provides an approach for determining the KU1 PEP EPZ size based on the area within which public dose, as defined in 10 CFR 20.1003, "Definitions", is projected to exceed 10 millisievert (mSv) (1 rem) total effective dose equivalent (TEDE) over 4 days (96 hours) from the release of radioactive materials from the facility, considering accident likelihood and source term, timing of the release sequence, and meteorology. In addition, NAT-3056 section 1.1 states that the methodology addresses the regulatory criterion that the PEP EPZ is the area in which predetermined, prompt protective measures are necessary. TP-LIC-RPT-0012 applied the NAT-3056 EPZ size determination methodology to determine the KU1 site-specific PEP EPZ size. NAT-3056 is listed in PSAR table 1.4-2 as a TR that is

incorporated by reference into the KU1 CP application, which is maintained as a separate document.

NAT-3056, Rev. 1, was submitted for NRC staff review and approval on November 16, 2023, with the review of the TR methodology coincident with the initial evaluation of the KU1 CP application review. The staff approved NAT-3056, Rev. 3 (NAT-3056-A), on March 28, 2025 (ML25076A653). NAT-3056-A is a risk-informed approach consistent with the guidance in RG 1.242, Appendix A, for determining PEP EPZ size to meet the requirements in 10 CFR 50.33(g)(2)(i). Both the NAT-3056-A methodology TR and the TP-LIC-RPT-0012 PEP EPZ determination analysis technical report follow the guidance of RG 1.242 to provide results to compare to the PEP EPZ sizing criteria in 10 CFR 50.33(g)(2)(i).

The staff evaluated whether the 6 limitations and conditions on the use of NAT-3056-A were either met or evaluated as part of the PSAR and the KU1 preliminary PEP EPZ sizing analysis technical report TP-LIC-RPT-0012. The staff's SER for NAT-3056-A (included in NAT-3056-A), imposed the following limitations and conditions:

1. The PRAs used to implement the TR methodology will be design- and site-specific and developed for all applicable hazards, all modes, and all sources of radioactive material, using the guidance in RG 1.247 "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities" and appendix A to RG 1.253 "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" (ML23269A222). Prior to the initial fuel loading, any exceptions to meeting capability categories referred to in RG 1.247 should be justified and documented.
2. An applicant that references this TR must justify the technical acceptability of the PRAs performed for the selected hazards and modes (e.g., site-specific scoping level seismic PRA). Prior to the initial fuel loading, PRAs supporting this methodology must include all applicable hazards and modes.
3. An applicant that references this TR must provide discussions of (1) how PRA key assumptions and key sources of uncertainty for each analyzed hazard, mode, and radioactive source were identified; (2) how the key assumptions and key sources of uncertainty identified as having the potential to significantly impact the PRA results have been characterized in a manner consistent with the current state of knowledge; and (3) how the impact of each identified key assumption and source of uncertainty was assessed and dispositioned.
4. An applicant that references this TR must justify that the scoping level seismic PRA is of sufficient technical acceptability. This means that the model will be design- and site-specific and developed based on acceptable methods and data. The engineering analyses, assumptions, and approximations used in developing the scoping level seismic PRA should be appropriate and should demonstrate the robustness of the conclusions with respect to the uncertainties in the assessment. Prior to the initial fuel loading, an applicant that references this TR must reassess the EPZ size using a seismic PRA that meets the requirements of non-LWR PRA standard, as endorsed in RG 1.247, to the extent necessary to support plume exposure pathway EPZ sizing calculation.

5. An applicant that references this TR will provide site-specific justification for the use of the upper bound peak ground acceleration (PGA) of 1.0 g when exercising the methodology in TR section 3.7, "Selection of Seismic Release Sequences," to choose the seismic scenario for the EPZ sizing analysis.
6. A periodic evaluation of the plume exposure pathway EPZ sizing analysis must be performed following an update or upgrade to the user's PRAs based on a review of changes to the plant structures, systems, and components, operational practices, and applicable plant and industry operational experience. Any changes to the emergency preparedness plan as a result of the evaluation should be conducted under 10 CFR 50.54(q).

The staff reviewed and evaluated the applicant's use of NAT-3056-A in TP-LIC-RPT-0012 and determined that the preliminary PEP EPZ sizing analysis includes discussion of the use of the KU1 PRA in the PEP EPZ sizing analysis. This same PRA was used in the Licensing Modernization Project-based PSAR analysis. The KU1 PRA was found to be acceptable for this CP application as discussed in section 3.1 of this SE. Therefore, based on its finding of PRA acceptability for the CP application, the staff also determined that the PEP EPZ sizing analysis supporting the PSAR addresses limitations and conditions 1 through 4 of NAT-3056-A.

The staff notes that because the preliminary PEP EPZ sizing analysis did not apply the NAT-3056-A methodology's upper bound PGA in the selection of a seismic scenario, the staff determined that limitation and condition 5 did not apply to this review. In addition, staff notes that limitation and condition 6 is not applicable to this CP application PSAR review of a preliminary PEP EPZ size calculation because limitation and condition 6 is applicable for maintenance of the final PEP EPZ approved size determination.

The staff reviewed the basis for the preliminary size of the KU1 PEP EPZ at the EAB in the preliminary EPZ sizing analysis technical report TP-LIC-RPT-0012. Because the PEP EPZ sizing methodology in NAT-3056-A was approved after the development of the preliminary EPZ sizing analysis technical report TP-LIC-RPT-0012, the staff's review focused on the differences of these two reports to assess whether the methods used in TP-LIC-RPT-0012 were acceptable.

Event Selection

Both the NAT-3056-A TR methodology and TP-LIC-RPT-0012 technical report PEP EPZ sizing analysis use the KU1 facility-specific PRA to evaluate the consequences of a range of events. TP-LIC-RPT-0012, section 7, states that external events and events involving the transfer of the spent fuel to dry cask storage were not developed in the PRA for use in the PSAR. The TP-LIC-RPT-0012 analysis did assess the consequences to bound a seismic design basis accident (DBA) consistent with the TerraPower EPZ sizing methodology in NAT-3056-A. The staff determined that the scope of the PRA (i.e., internal events) was acceptable for the CP application stage as discussed in section 3.1 of this SE, and therefore also determined it acceptable for use in the preliminary determination of the PEP EPZ size. Based on the preceding discussion, the staff determined that the use of the PRA to develop the range of events and the use of a seismic DBA is consistent with NAT-3056-A methodology and therefore determined that the PEP EPZ sizing analysis has appropriately considered a range of potential events. The subsequent OL application should include the technical justification for the final PEP EPZ size in accordance with the requirements of 10 CFR 50.33(g)(2)(i). In addition, if the

final PEP EPZ sizing analysis uses the approved methodology in NAT-3056-A, the applicant should address the limitations and conditions in the topical report.

TP-LIC-RPT-0012, section 8, describes the preliminary radiological consequence analysis results. The staff determined through review of the PEP EPZ size analysis descriptions in the TP-LIC-RPT-0012 technical report in comparison to the licensing basis events analysis in PSAR chapter 3, that the preliminary PEP EPZ size analysis covers a full spectrum of likely accident scenarios for the proposed Natrium reactor as described in the PSAR. The preliminary PEP EPZ sizing analysis also evaluated a conservative surrogate seismic event to bound the dose consequences from potential seismic events, which was termed the seismic other quantified event (OQE). As stated in TP-LIC-RPT-0012, the seismic OQE is representative of the likely damage resulting from a seismic scenario. The staff's review found that the preliminary analysis did not follow the NAT-3056-A methodology to determine a seismic DBA based on a bounding peak ground acceleration. Instead, the seismic OQE used a bounding source term and consequence analyses. The staff verified through regulatory audit of the supporting calculation reports (ML25302A443) that the seismic OQE is sufficient to represent seismic scenarios for the preliminary PEP EPZ sizing analysis as determined with relation to the preliminary Natrium design information for the CP application. Specifically, the staff determined the seismic OQE source term is sufficiently bounding of likely damage from seismic scenarios for the CP application because it is represented by a source term based on the assumption of failure of the fuel cladding in [

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The subsequent OL application must include the technical justification for the final PEP EPZ size in accordance with the requirements of 10 CFR 50.33(g)(2)(i). In addition, if the final PEP EPZ sizing analysis uses the approved methodology in NAT-3056-A, the applicant should address the limitations and conditions provided in NAT-3056-A.

As discussed in PSAR section 11.3.3.1, the preliminary PEP EPZ sizing analysis does not specifically identify security events, which is consistent with the CP PSAR licensing phase of design. During the audit, the staff verified the preliminary EPZ sizing consequence analysis considered security events consistent with the NAT-3056-A methodology. As described in PSAR section 11.3.3.1, the final PEP EPZ sizing analysis will assess security events either with a qualitative or quantitative approach once more detail about security events is developed to determine those that may not be bound by the list of internal and external licensing basis events. The staff will review the final PEP EPZ size and justification in the review of the OL application.

Based on its review as discussed above, the staff determined that the event selection for the preliminary PEP EPZ sizing analysis is acceptable because it is consistent with the NAT-3056-A methodology or justifies differences from the methodology, and conforms with the guidance in RG 1.242, Appendix A.

Preliminary PEP EPZ Sizing Consequence Analysis

As stated in TP-LIC-RPT-0012, section 8, the preliminary PEP EPZ sizing analysis uses the NAT-3056-A methodology to determine the consequences of the evaluated events and assess the analysis against the PEP EPZ sizing criteria. The staff's review of the information in the technical report analysis confirmed that the description of the events that are licensing basis events (LBEs) and the source term methodology are the same as described in PSAR chapter 3. The staff's evaluation of the LBE source terms is discussed in this SE, in section 3.2. The staff

also confirmed that the determination of source terms for the OQEs followed the same methodology used for LBEs, based on the TerraPower TR, “Radiological Source Term Methodology,” which was approved as TR NAT-9392-A (ML25211A271). Consequences for each event were quantified using analysis methods consistent with the approved TerraPower TR NAT-9391-A, “Radiological Release Consequences Methodology,” (ML25211A267), radiological release consequences analysis methodology to develop the TEDE to an individual for exposure from the release plume. The preliminary PEP EPZ sizing analysis used the NAT-3056-A approved PEP EPZ sizing methodology dose-related criteria to determine the size of the PEP EPZ in accordance with the requirements in 10 CFR 50.33(g)(2)(i).

As stated in the technical report TP-LIC-RPT-0012 section 8, the analysis included evaluation of potential cliff edge effects for events with frequency as low as 1×10^{-8} per reactor-year. The applicant’s evaluation assesses whether the LBEs had adequate margin to the Nuclear Energy Institute (NEI) 18-04, “Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Rev.1 (ML19241A472), F-C target, whether OQEs with 95th percentile frequencies in the beyond design basis event range had adequate margin to the NEI 18-04 F-C target and included evaluation of the highest consequence event quantified in the PRA model. The analysis states that based on the review of the LBE categorization and risk-significance determination for both mean and 95th percentile values, no events required cliff-edge consideration. Through the regulatory audit, the staff verified that the PEP EPZ sizing analysis consideration of uncertainty and cliff edge effects are consistent with the treatment of uncertainty and cliff edge effects in the PRA used in the PSAR and is also consistent with the approved NAT-3056-A methodology.

The preliminary PEP EPZ sizing analysis results provided in TP-LIC-RPT-0012 showed that for all evaluated events, no doses exceeded the PEP EPZ sizing dose criteria at the EAB. Therefore, consistent with the approved methodology, there was no need for the applicant to reassess the necessity of prompt protective measures to meet all the PEP EPZ sizing criteria and support the determination of the PEP EPZ set at the EAB for the KU1 CP. Based on its review of the information provided in TP-LIC-RPT-0012, the staff determined that the consequence analysis is consistent with the approved PEP EPZ sizing methodology in NAT-3056-A and supports the preliminary establishment of a PEP EPZ set at the KU1 site EAB because it conforms with the guidance in RG 1.242, Appendix A.

In accordance with the requirements in 10 CFR 50.33(g)(2)(i), the subsequent KU1 OL application will be required to justify the final KU1 site PEP EPZ size.

11.3.1.1.3 Staff’s Findings and Conclusions of the KU1 Facility PEP EPZ Size Determination Analysis Evaluation (10 CFR 50.160(b)(3))

Based on the above review, staff determined that the information in the CP application concerning the preliminary KU1 PEP EPZ size determination analysis, supports a preliminary finding that the KU1 PEP EPZ can be set at the KU1 site EAB, addresses the minimum information necessary in a CP application described in DANU-ISG-2022-01, and conforms with the guidance in RG 1.242, Appendix A. Therefore, the staff determined that the preliminary KU1 PEP EPZ size determination analysis information is acceptable and meets the applicable PSAR preliminary plan discussion requirement in 10 CFR 50.34(a)(10)(ii), the PEP EPZ size analysis criteria in 10 CFR 50.33(g)(2)(i), and the PEP EPZ boundary and physical characteristics description requirements of 10 CFR 50.160(b)(3). The final KU1 PEP EPZ emergency plan design information will be evaluated by staff at the OL stage, should the applicant apply for an

OL, such application would include the KU1 final safety analysis report (FSAR) and emergency plan.

11.3.1.2 KU1 Performance-based Emergency Preparedness Framework
(10 CFR 50.160(b)(1))

Applicants and licensees complying with the alternative emergency preparedness performance-based requirements of 10 CFR 50.160, are required under 10 CFR 50.160(b)(1), to demonstrate effective response in drills and exercises and describe in their emergency plans how they will maintain preparedness. To comply with the performance-based emergency preparedness framework requirements of 10 CFR 50.160(b)(1), both applicants and licensees, with an established site boundary PEP EPZ, will need to have an emergency plan that meets the requirements of 10 CFR 50.160(b)(1)(i) through (iv)(A).

The following SE subsections document the staff's review, evaluation and regulatory findings of requisite KU1 performance-based emergency preparedness framework PSAR design information.

11.3.1.2.1 KU1 Emergency Response Performance Evaluation (10 CFR 50.160(b)(1)(iii))

The emergency response performance criteria of 10 CFR 50.160(b)(1)(iii) requires that the licensee's emergency response team (i.e., emergency response organization) have sufficient capability to demonstrate the enumerated emergency response functions listed in 10 CFR 50.160(b)(1)(iii)(A) through (H) using drills or exercises. The emergency plan must include descriptions of how the emergency response organization (ERO) has been provided with sufficient capability to demonstrate the emergency response functions using drills or exercises.

The following SE sections document the staff's review, evaluation, and regulatory findings of requisite KU1 emergency preparedness preliminary plan design information included in the KU1 PSAR.

11.3.1.2.1.1 Event Classification and Mitigation (10 CFR 50.160(b)(1)(iii)(A))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(iii)(A) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate event classification and mitigation. Specifically, 10 CFR 50.160(b)(1)(iii)(A) states, "[a]ssess, classify, monitor, and repair facility malfunctions in accordance with the emergency plan to return the facility to safe conditions." The staff's review addresses these elements below:

Capability to Classify

To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the staff used the guidance of RG 1.242, section C.6.c(3), which states that the emergency plan should describe the emergency classification scheme, the associated emergency action levels (EALs) applicable for the design of the facility, and the technical basis and methodology for determining the thresholds corresponding to each of the EALs,

The staff's review identified that PSAR sections 11.3.1 and 11.3.6.2 state that a design and site-specific emergency classification scheme and associated EALs, and the documentation of the methodology for determining the EALs, will be developed and submitted in the emergency plan at the OL application licensing phase. The event classification scheme will include the KU1 facility emergency classification levels and EALs that will be based on a predetermined set of facility conditions. Emergency plan implementing procedures (EPIPs) will be developed to guide the ERO mitigation efforts.

Capability to Assess and Monitor

RG 1.242, section C.6.c(1), states, in part, that the emergency plan should describe the capabilities to perform event classification and mitigation.

The staff's review identified that PSAR sections 11.3.6.3 and 11.3.9.2 state that capabilities and procedural guidance will be provided to assess, monitor, or report:

- Radiological conditions and consequences of emergency situations in and around the facility.
- Dose projections using real-time meteorological information.
- Core or reactor vessel damage.
- The extent of any radiological release, including the release of hazardous chemicals produced from licensed material.
- On-site conditions to support recommendations for protective actions to offsite agencies.

PSAR section 11.3.9.3 states that ERO positions will be provided with computers and voice communication equipment needed to support their ERO functions and that a system to collect, process, display, share, and communicate plant parameters to aid in emergency response decision making will be designed and available for use. PSAR section 7.6.7.3 states that the capability to provide indication of plant status to the designated KU1 primary and backup emergency response facilities (ERFs) is provided by plant data received from the Nuclear Island Control System to provide indication of plant status.

The ERFs are designed to allow the ERO to perform the emergency response functions. During an emergency, the designated ERF will be the primary communications center for the plant ERO. The ERFs provide the capability to access technical data information and displays to access and assist in the detailed analysis and diagnosis of abnormal plant conditions. In addition, capabilities and EPIPs will be provided for assessing and identifying the seriousness of the radiological consequences of emergency situations will be available.

PSAR section 2.4.5.1.1 states that the instrumentation on the KU1 site meteorological tower, a 60-meter (197-foot) meteorological tower, consists of wind speed, wind direction, temperature sensors, and a relative humidity sensor, among other instruments. PSAR figure 2.4-63 shows the location of the KU1 site meteorological tower.

Capability to Repair Facility Malfunctions and Return the Facility to Safe Conditions

RG 1.242, section C.6.c(2), states, in part, that the ERO would use EIPs to assess malfunctions and repair facility malfunctions in a timely manner and return the facility to safe conditions.

The staff's review identified that PSAR section 11.3.1 states that the capabilities and guidance will be developed to access the facility after a radiological emergency or a hostile action-based emergency and to address and implement reentry plans. In addition, a process will be developed for maintaining and making changes to the KU1 EIPs, including methods to account for facility changes. PSAR section 7.6.5.3.2, provides an example of facility equipment that will be designed and available for use to repair facility malfunctions such as the Reactor Instrumentation System instrument channels that support the post-accident monitoring that have testing capability to facilitate repair of malfunctioning equipment.

11.3.1.2.1.2 Staffs Findings and Conclusions of Event Classification and Mitigation Emergency Response Function Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(A))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies be included in a PSAR for a CP application.

Based on the staff's review and evaluation described above in section 11.3.1.2.1.1, the staff concludes that the KU1 CP application PSAR preliminary plan proposed event classification and mitigation emergency response function capabilities are acceptable.

Thus, staff finds that the above KU1 PSAR preliminary event classification and mitigation emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan ERO emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(A), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. At that time, the staff will perform a more detailed evaluation of the event classification and mitigation emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.3 Protective Actions Emergency Response Function Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(B))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(iii)(B) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate protective actions. To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the NRC staff used the guidance of RG 1.242, section C.6.d, "Protective Actions," which states, that the emergency plan should describe capabilities to determine, implement, and recommend appropriate protective actions for a variety of hazards, to include the methods, processes, equipment,

facilities, and personnel, to implement protective action emergency response function capabilities.

The staff's review identified that PSAR sections 11.3.6.1 and 11.3.6.3 state that KU1 on-site protective action capabilities will include:

- Providing recommendations for and use of personnel protective gear and equipment.
- The use of on-site communication systems and action specific alarms (e.g., site evacuation) to communicate appropriate on-site protective actions to on-site occupants.
- The use of contamination control measures such as moving personnel away from contaminated areas and establishing restricted areas.
- Sheltering on-site.
- Site evacuation to include site evacuation of non-essential personnel.
- Recommend protective actions to offsite authorities (i.e., offsite emergency response organizations) as conditions warrant.

PSAR section 11.3.6.3 states that protective equipment will be stored on-site and provided for use to continue and expand mitigation and protective actions. PSAR section 11.3.9.3 states that protective actions capabilities will be provided to notify the KU1 ERO personnel in a timely manner and to alert on-site personnel regarding protective actions that are implemented on-site. As stated in PSAR section 11.3.1, guidance will be developed to determine, implement, and recommend appropriate protective actions.

The capability to assess on-site conditions to support recommendations for protective actions to offsite emergency response organizations (OROs) will be provided in the emergency plan. PSAR section 2.2.2.1, states that for the areas within the site EAB (i.e., PEP EPZ), but outside of licensee ownership, the licensee will maintain agreements with the appropriate OROs as part of the KU1 emergency plan and that these agreements will ensure appropriate plans are in place that include protective actions within the areas outside of KU1 licensee ownership.

11.3.1.2.1.3.1 Staffs Findings and Conclusions of the Protective Actions Emergency Response Function Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(B))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies be included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plan for proposed protective actions emergency response function capabilities. Thus, staff find that the above PSAR preliminary protective action emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan protective actions emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(B), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the

KU1 OL application. At that time, the staff will perform a more detailed evaluation of the protective action emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.4 Communications Emergency Response Function Evaluation
(10 CFR 50.160(b)(1)(iii)(C))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(iii)(C) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate communications. To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for the minimum design information necessary for a PSAR, the staff used the guidance of RG 1.242, section C.6.e, which states, in part, that the emergency plan should describe capabilities to establish and maintain communications among the response facilities, ERO and organizations that may have on-site responsibilities or agreements (e.g., local law enforcement, medical and hospital services, fire response services), and the procedures, methods, processes, equipment, facilities, and personnel that will implement the ERO communication capabilities emergency response function.

The staff's review identified that PSAR section 11.3.9.3 states that, to support the emergency preparedness plan, communication capabilities will be established and maintained and that information sharing between ERO personnel positions, ERFs (e.g., main control room, primary ERF, and backup ERF), and other on-site and offsite organizations that may have emergency response functions including:

- Capabilities for 24-hour notification will be provided for initial and ongoing communication for activation of the ERO in a timely manner, to alert on-site personnel regarding the status of the emergency, and to on-site and offsite organizations that may have facility emergency response functions.
- On-site communication systems and action specific alarms (e.g., site evacuation) will be provided to communicate appropriate on-site protective actions.
- ERO personnel positions will be provided with computers and voice communication equipment needed to support their ERO functions.
- Capabilities will be developed to provide emergency response status and other necessary information to the NRC. The methods used to notify the NRC and the information provided will be established in coordination with the NRC and will be addressed in EIPs.
- Coordination with the NRC, state, and local governmental agencies to determine the voice and data communication systems compatible with their respective systems will be performed.
- Notification requirements, protocols, and agreements will be developed and established in coordination with the Wyoming Office of Homeland Security and local governmental agencies within the immediate vicinity of the KU1 site.

PSAR section 1.1.4.3.12, states that the KU1 Nuclear Island Plant Communication System (PCO) provides internal and external communications during normal and emergency plant operations, as well as an independent means of effective on-site and offsite communication during all modes of plant operation, including abnormal and emergency operating conditions.

11.3.1.2.1.4.1 Staffs Findings and Conclusions of the Communications Emergency Response Function Capabilities Evaluation (10 CFR 50.160(b)(1)(iii)(C))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies be included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plans for proposed communications emergency response function capabilities. Thus, staff finds that the above KU1 PSAR preliminary communications emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and is sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan communications emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(C), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application, should the applicant apply for an OL. At that time, the staff will perform a more detailed evaluation of the communications emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.5 Command and Control Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(D))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(iii)(D) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate command and control. To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the NRC staff used the guidance of RG 1.242, section C.6.f, which states that the emergency plan should describe the ERO command and control emergency response function capabilities to perform adequate command and control, the supporting ERO organizational structure with defined roles, responsibilities, and authorities for directing and performing emergency response functions, and the qualitative criteria to assess command and control.

The staff's review identified that PSAR section 11.3.4.2 states that upon declaration of an emergency, designated qualified members of the KU1 facility on-shift operations organization will fulfill the corresponding ERO roles and activities in response to a declared emergency. As described in PSAR section 11.3.4.2, once an emergency declaration is made, the on-shift Shift Manager will function as the ERO Emergency Director (ED), the on-shift health physics personnel will support the ED with the radiological health physics aspects of the emergency, the on-shift security personnel will provide security to address the pending emergency, and the on-shift engineering personnel will begin to focus on plant assessment and technical support for operations in response to the emergency.

The ED will assume the initial ERO command and control emergency response function following the emergency declaration. The ED is responsible for ensuring that the KU1 ERO's enumerated emergency response functions in 10 CFR 50.160(b)(1)(iii) are fulfilled. The Shift Manager will function as the ED and will fulfill this role until the ED duties have been transferred to the activated ERF ED. Once the ERF ERO has been activated and staffed, the initial on-shift ERO emergency response functions and responsibilities will be transferred to the activated ERF ERO personnel. The activated ERF ERO will relieve the on-shift reactor operators of peripheral emergency response duties and communications not directly related to reactor control during an emergency, which will allow the main control room (MCR) on-shift personnel to focus on responding to the operational aspects of the emergency. The ERF ERO will assist in the management and control of plant emergency response capabilities to support the on-shift MCR personnel, mitigate the consequences of an accident, and respond quickly to abnormal plant conditions.

The Director of Organizational Effectiveness is responsible for emergency planning and is shown on the KU1 operating organization block diagram PSAR figure 11.1-4. The Director of Organizational Effectiveness reports to the PGM who oversees the daily operations of KU1. The PGM is responsible for occupational and public radiation safety.

PSAR section 11.3.4, states that the below listed ERO command and control emergency response function design details will be identified in the emergency plan and submitted with the KU1 OL application:

- The KU1 minimum ERO personnel required to conduct immediate emergency response (e.g., on-shift designated ERO).
- The final ERO specific roles and responsibilities for ERO personnel.
- The ERO activation and ERO emergency function response times.
- The ERO EIPs will be for available audit with the KU1 OL application.

11.3.1.2.1.5.1 Staff Findings and Conclusions for the Command and Control Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(D))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plans for proposed command and control emergency response function capabilities. Thus, staff find that the above KU1 PSAR preliminary command and control emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan command and control emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(D), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application, should the applicant apply for an OL. At that time, the staff will perform a

more detailed evaluation of the command and control emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.6 Staffing and Operations Emergency Response Function Evaluation
(10 CFR 50.160(b)(1)(iii)(E))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to Part 50. 10 CFR 50.160(b)(1)(iii)(E) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate staffing and operations. To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the staff used the guidance of RG 1.242, section C.6.g, which states that the emergency plan should describe the capabilities to adequately staff the ERO emergency response functions within an appropriate timeframe and the process used to complete training before assigning roles and responsibilities to the ERO personnel.

The staff's review identified that PSAR section 11.3 states that postulated emergencies will be determined for the KU1 facility and used to inform the emergency response functions, ERO staffing and ERFs. ERO positions and staffing level will be determined based on the roles and responsibilities needed to implement the emergency response functions. PSAR section 11.3.4, states that the KU1 minimum ERO required to conduct immediate emergency response, the ERF ERO personnel, the ERO emergency function response times, and the specific roles and responsibilities for ERO personnel, will be identified and provided in the KU1 emergency plan.

PSAR section 11.3.8 states that initial training will be provided to individuals before being assigned ERO responsibilities. Additional capabilities that will be used to complete the KU1 ERO training:

- Personnel will receive generic emergency preparedness plan training and training specific to their assigned emergency response functions.
- The content of the training program will include the overall emergency plan and the relevant implementing procedures.
- Off-site personnel and agencies whose assistance will be needed to respond to an emergency will be provided site familiarization training.

PSAR section 11.3.8 states that details of the ERO training program will be provided in the emergency plan.

PSAR section 11.3.10, states that the ERO drills and exercises will be used to demonstrate the ERO capabilities to perform and maintain the emergency response functions enumerated in 10 CFR 50.160(b)(1)(iii)(A) through (H), and that drills and exercises, as well as responses to actual declared emergency events, will be critiqued to identify weaknesses or deficiencies in ERO performance and the KU1 emergency preparedness program. A performance monitoring program will be developed to demonstrate the effectiveness of the implementation of the emergency plan to carry out an effective response to emergency and accident conditions and

will use drills and exercises to demonstrate the capabilities to perform and maintain the emergency response functions.

11.3.1.2.1.6.1 Staff Findings and Conclusions for the Staffing and Operations Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(E))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plans for proposed staffing and operations emergency response function. Thus, staff find that the above KU1 PSAR preliminary staffing and operations emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan staffing and operations emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(E), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. At that time, the staff will perform a more detailed evaluation of the communications emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.7 Radiological Assessment Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(F))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(iii)(F) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate radiological assessment. To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the staff used the guidance of RG 1.242, section C.6.h, which states that the emergency plan should describe the capabilities to assess, monitor, and report to the applicable ERO, the radiological conditions in and around the facility and on-site locations, the capabilities to issue and use protective equipment, the capabilities to assess, monitor, and report to the applicable ERO the extent of any core (or other vessel containing irradiated special nuclear material) damage, the capabilities to assess, monitor, and report to the applicable ERO personnel the extent of any radiological release and the releases of hazardous chemicals produced from licensed material, and how the ERO team will use EIPs to perform these radiological emergency response functions.

The staff's review identified PSAR section 11.3.9.2 states that capabilities, provisions, equipment, and procedural guidance (i.e., EIPs) will be provided to assess, monitor, and report radiological conditions and consequences of emergency situations in and around the KU1 facility, which includes the capability to:

- Provide dose projections using real-time meteorological information.
- Report the extent of any core or reactor vessel damage, or other vessel containing irradiated special nuclear material.

- Assess, monitor, and report the extent of any radiological release, including the releases of hazardous chemicals produced from licensed material.

The EIPs for assessing and identifying the seriousness of the radiological consequences of emergency situations and the capability to notify the ERO in a timely manner regarding the status of the emergency, will be available for audit with the KU1 OL application in the final emergency plan.

PSAR section 11.3.6.3, states that protective equipment will be stored and provided on-site for use to continue and expand mitigation and protective actions.

11.3.1.2.1.7.1 Staff Findings and Conclusions for the Radiological Assessment Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(F))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plans for proposed radiological assessment emergency response function capabilities. Thus, staff find that the above KU1 PSAR preliminary radiological assessment emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan radiological assessment emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(F), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application, should the applicant apply for an OL. At that time, staff will perform a more detailed evaluation of the radiological assessment emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.8 Reentry Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(G))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(iii)(G) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate reentry. To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the staff used the guidance of RG 1.242, section C.6.i, which states that the emergency plan should describe the reentry emergency response function capabilities to develop and implement reentry plans for access to the facility after radiological emergencies and the capabilities to develop and implement reentry plans for access to the facility following hostile action-based emergencies, and how the ERO will use EIPs to demonstrate the development and implementation of reentry plans.

The staff's review identified that the KU1 PSAR emergency preparedness preliminary plan states that reentry emergency response function capabilities and guidance will be developed to address and implement reentry plans to access the facility after a radiological or hostile action-based emergency. The staff evaluated the PSAR for reentry function capabilities that would be a part of a reentry plan for determining when facility conditions would be acceptable to justify

reentry following a radiological emergency. Several reentry emergency response function capabilities described in the PSAR are as follows:

- PSAR section 7.6.6, states that the Radiation Monitoring System's (RMS) process radiation monitoring subsystem monitors airborne locations for radiation levels during all operating conditions. The RMS provides local indication of plant radiological conditions in all areas that are required to be monitored for area radiation levels, airborne activity, and radioactive material in various gaseous and liquid process and effluent streams. The RMS provides continuous radiation monitoring and sampling and is provided for radioactive processes and for release points where radioactive effluents leave the site boundary.
- Provisions, equipment, and procedural guidance will be provided in the KU1 emergency plan to assess, monitor, and report radiological conditions, the extent of any radiological releases, and consequences of emergency situations in and around the facility.
- Capabilities will be provided to assess and develop protective actions for on-site personnel.
- Capabilities will be provided to notify the ERO in a timely manner and to alert on-site personnel regarding the status of the emergency (e.g., releases, location of the emergency, protective actions that are implemented on-site).
- On-site means for decontamination and first aid for personnel, such as decontamination showers and first aid kits, will be provided.
- On-site communication systems and action specific alarms will be used to communicate appropriate on-site protective actions.
- The ERO ED has the authority to call in additional personnel to provide further emergency response support and assign these individuals ERO duties based on the emergency response function(s) to be performed.
- EIPs for assessing and identifying the seriousness of the radiological consequences of emergency situations and to guide mitigation efforts will be available to the ERO.

11.3.1.2.1.8.1 Staff Findings and Conclusions for the Reentry Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(G))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plans for proposed reentry emergency response function capabilities. Thus, staff find that the above PSAR preliminary reentry emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan reentry emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(G), will be provided by the applicant

through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. At that time, the staff will perform a more detailed evaluation of the reentry emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.9 Critique and Corrective Actions Emergency Response Function Evaluation
(10 CFR 50.160(b)(1)(iii)(H))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(iii)(H) requires the emergency plan contain information needed to demonstrate that the emergency response team has sufficient capability to demonstrate critique and corrective actions. As described in PSAR section 11.3.10, the critique and corrective actions emergency response function is the licensee's emergency preparedness program that will evaluate (e.g., critique) the performance of the ERO's emergency response functions in drills, exercises, or responses to actual emergencies, and identify weaknesses or deficiencies in the ERO's emergency response performance. The licensee would then use a corrective action program to evaluate, track, and correct the identified ERO emergency response performance weaknesses and deficiencies identified in drills, exercises, or responses to actual emergencies.

To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the staff used the guidance of RG 1.242, section C.6.j, which states that the emergency plan should describe the emergency preparedness program's capabilities to critique emergency response functions and implement effective corrective actions and describe how the ERO will use EIPs to critique emergency response functions and implement timely corrective actions.

The staff's review identified that PSAR section 11.3.10 states that the drills and exercises that will be used to demonstrate the capabilities to perform and maintain the emergency response functions and responses to actual declared emergency events will be critiqued to identify weaknesses or deficiencies in the ERO performance and the KU1 emergency preparedness program. A methodology will be developed to critique the effectiveness of the emergency response functions after drills, exercises, and actual plant events. The KU1 corrective action program will be used to evaluate, track, and correct identified emergency program weaknesses and deficiencies. The corresponding corrective actions will be identified, implemented, and addressed by the corrective action program.

11.3.1.2.1.9.1 Staff Findings and Conclusions for the Critique and Corrective Actions Emergency Response Function Evaluation (10 CFR 50.160(b)(1)(iii)(H))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plans for proposed critique and corrective actions emergency response function capabilities. Thus, the staff finds that the above KU1 PSAR preliminary critique and corrective actions emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan critique and corrective actions emergency response function capability design details, necessary for staff to perform a SE for compliance with the applicable emergency response function capabilities criteria in 10 CFR 50.160(b)(1)(iii)(H), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. At that time, the staff will perform a more detailed evaluation of the critique and corrective actions emergency response function final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.1.10 Staff Findings and Evaluation Conclusions of the ERO Emergency Response Performance Evaluation (10 CFR 50.160(b)(1)(iii))

Based on the staff's review and evaluation of the PSAR emergency preparedness preliminary plan principal design criteria and descriptions of the KU1 ERO emergency response function capabilities, the staff finds that the KU1 PSAR preliminary plan descriptions of the emergency response function capabilities which are required in 10 CFR 50.160(b)(1)(iii)(A) through (H) address the minimum information necessary in a CP application described in DANU-ISG-2022-01. Thus, the staff finds that the above KU1 PSAR ERO emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement in 10 CFR 50.34(a)(10)(ii) and sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

11.3.1.2.2 KU1 Emergency Response Performance Objectives Evaluation (10 CFR 50.160(b)(1)(ii))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. As discussed in RG 1.242, the performance objectives criteria under 10 CFR 50.160(b)(1)(ii) require applicants and licensees to develop a performance monitoring program that includes performance metrics and objectives for each emergency response function required in 50.160(b)(1)(iii)(A) through (H) and describe it in their emergency plans. In addition, RG 1.242 discusses that the applicant or licensee must describe the implementation of a performance objective scheme that reflects the enumerated emergency response functions.

To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the staff used the guidance of RG 1.242, section C.6.b, which provides guidance for the performance objective topics that should be included in the emergency plan's description of the performance monitoring program. The staff's review identified that PSAR section 11.3.10 describes the emergency response performance objectives as part of the performance monitoring. PSAR section 11.3.1 states that the emergency plan, EIPs, and other programmatic documents will be considered quality documents and will be addressed by the USO QA program that will be submitted as part of the OL application. The quality assurance program is described in PSAR section 8.1 and in detail in TP-QA-PD-0001 Rev. 14-A, which is incorporated by reference in the PSAR; the staff found the quality assurance program implementation and the incorporation by reference of TP-QA-PD-0001 Rev. 14-A to be acceptable in section 8.2 of this SE. PSAR section 11.3.10 states that under the KU1 performance monitoring program, drills and exercises will be used to demonstrate the capabilities to perform and maintain the emergency response functions in 10 CFR 50.160(b)(1)(iii)(A) through (H).

11.3.1.2.2.1 Staff Findings and Evaluation Conclusions of the KU1 Emergency Response Performance Objectives Evaluation (10 CFR 50.160(b)(1)(ii))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains preliminary plans for proposed performance objectives. Thus, staff find that the above KU1 PSAR preliminary performance objectives emergency response function capability descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and are sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan performance objectives and performance monitoring program design details, necessary for staff to perform a SE for compliance with the applicable performance objectives criteria in 10 CFR 50.160(b)(1)(ii), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. At that time, the staff will perform a more detailed evaluation of the final performance objectives design details under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.3 KU1 Maintenance of Performance Evaluation (10 CFR 50.160(b)(1)(i))

Under 10 CFR 50.34(a)(10), the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(1)(i) requires the emergency plan contain information needed to demonstrate maintenance of performance. To review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information, the staff used the guidance of RG 1.242, section C.6.a(2), which states that the emergency plan should describe the process for maintaining and making changes to the emergency plan and associated procedures, including methods to account for facility changes and the methods used to conduct independent reviews of the emergency preparedness program.

The staff's review identified that PSAR section 11.3.1 states that the methods for maintaining the emergency plan, EIPs, and other programmatic documents, including methods for conducting independent reviews and periodic reviews by the offsite organizations, will be provided in the KU1 emergency plan. A process will be developed for making changes to the emergency plan and EIPs, and to account for facility changes. Changes to the emergency plan would follow the requirements in 10 CFR 50.54(q), "Emergency plans." In addition, emergency plan maintenance and reviews will be conducted in accordance with the requirements of 10 CFR 50.54(t).

11.3.1.2.3.1 Staff Findings and Evaluation Conclusions of the KU1 Maintenance of Performance Evaluation (10 CFR 50.160(b)(1)(i))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains a description of preliminary plans for proposed maintenance of performance. Thus, staff find that the above

KU1 PSAR preliminary maintenance of performance descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and are sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan maintenance of performance design details, necessary for staff to perform a SE for compliance with the applicable maintenance of performance criteria in 10 CFR 50.160(b)(1)(i), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. At that time, the staff will perform a more detailed evaluation of the maintenance of performance final design details under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.4 KU1 Onsite Planning Activities (10 CFR 50.160(b)(1)(iv)(A))

The staff used the on-site planning activities guidance of RG 1.242, section C.7, to perform the review and evaluation of the KU1 PSAR preliminary plan on-site planning activities design descriptions and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information.

Prepare and Issue Public Information During Emergencies (10 CFR 50.160(b)(1)(iv)(A)(1))

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. Under 10 CFR 50.160(b)(1)(iv)(A)(1) the emergency plan shall maintain the capability to prepare and issue public information during emergencies. As described in RG 1.242, section C.7, the emergency plan should describe the capabilities and processes to manage and coordinate the licensee's media information during an emergency and support the public information functions of the federal, State, and local authorities.

The staff's review identified that PSAR section 11.3.1 states that capabilities and guidance will be developed to manage and coordinate the dissemination of information to the media and to support the public information functions of the federal, state, and local authorities. Capabilities will also be provided to assess on-site conditions to support recommendations for protective actions for offsite agencies. State and local government notification requirements, protocols, and agreements will be developed and established in coordination with the Wyoming Office of Homeland Security and local governmental agencies within the immediate vicinity of the site. The letters of agreement obtained will be submitted with the KU1 emergency plan. The applicable EPIPs for taking on-site protective measures include coordination with the fire protection and security plans.

Implement the NRC-approved Emergency Response Plan in Conjunction with the Licensee's Safeguards Contingency Plan (10 CFR 50.160(b)(1)(iv)(A)(2))

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. Under 10 CFR 50.160(b)(1)(iv)(A)(2) the emergency plan must maintain the capability to implement the NRC-approved emergency response plan in conjunction with the licensee's safeguards contingency plan. As described in RG 1.242, section C.7, the emergency plan should describe the capabilities and processes or procedures to support implementation of the emergency plan in conjunction with the licensee safeguards contingency plan.

The staff's review identified that PSAR section 11.3.1 states that guidance will be developed to coordinate the implementation of the KU1 emergency plan with the safeguards contingency plan, including coordination, support, and communication with the incident command post and incident commander.

*Establish Voice and Data Communications with the NRC for Emergencies
(10 CFR 50.160(b)(1)(iv)(A)(3))*

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. Under 10 CFR 50.160(b)(1)(iv)(A)(3) the emergency plan must maintain the capability to establish voice and data communications with the NRC for emergencies. As described in RG 1.242, section C.7, the emergency plan should describe the planning activities, capabilities, and processes or procedures to support actions including, but not limited to, those described in RG 1.242, section C.7.

The staff's review identified that PSAR section 11.3.1 states that capabilities and guidance will be developed to provide emergency response status and other necessary information to the NRC. The applicant plans to work with the NRC to determine the compatibility of voice and data communication systems with their respective systems.

Establish an Emergency Facility or Facilities from Which Effective Direction Can Be Given and Effective Control Can Be Exercised During an Emergency. With Capabilities to Support the Enumerated Emergency Response Functions (10 CFR 50.160(b)(1)(iv)(A)(4))

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. Under 10 CFR 50.160(b)(1)(iv)(A)(4) the emergency plan must maintain the capability to establish an emergency facility or facilities from which effective direction can be given and effective control can be exercised during an emergency, with capabilities to support the emergency response functions as described in 10 CFR 50.160(b)(1)(iii). As described in RG 1.242, section C.7, the emergency plan should describe each ERFs, including, as applicable, descriptions of location, capabilities, size, equipment, and backup locations to transfer the functions if the facility is not habitable or accessible. The emergency plan should also describe the processes, systems, and equipment for collecting and processing data for decision making in an emergency.

The staff's review identified that PSAR section 11.3.9.1 states that a primary ERF (ERFP), located on-site, and a backup ERF (ERFB), located offsite, will be constructed for the KU1 facility. There are no differences in the functions performed by ERFP and ERFB. Both ERFs will be designed to allow the ERO to perform the enumerated emergency response functions listed in 10 CFR 50.160(b)(1)(iii). Both ERFs will be designed to support functions like those required for a Technical Support Center and Operational Support Center as described in NUREG-0696, "Functional Criteria for Emergency Response Facilities," (ML051390358), and NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," (ML19275B682). The ERFP and ERFB will both include access to plant technical data information and displays to assist in the detailed analysis and diagnosis of abnormal plant conditions.

During an emergency or event, once the ERO has been activated, ERO personnel will assemble and staff the ERF. Once the ERF is fully staffed, the emergency response functions

that were initially being performed by the on-shift MCR personnel and designated facility on-shift ERO personnel, will be transitioned to the fully staffed, activated ERF ERO. Once the ERF has fully transitioned from the on-shift staff and is fully activated and operational, the ERF will be the primary communications center for the KU1 facility, during the emergency, directing and performing communications capabilities on-site, with the NRC and the emergency plan required applicable OROs (e.g., other state and local governmental agencies and emergency response support entities) as required by the emergency plan.

Provide Site Familiarization Training for Any Offsite Organization That May Respond to the Site in the Event of an Emergency (10 CFR 50.160(b)(1)(iv)(A)(5))

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. Under 10 CFR 50.160(b)(1)(iv)(A)(5) the emergency plan must maintain the capability to provide site familiarization training for any offsite organization that may respond to the site in the event of an emergency. As described in RG 1.242, section C.7, the emergency plan should describe: (1) site familiarization training to be provided, the expected participants, and the frequency of training; and (2) any coordination to ensure that the local law enforcement, medical, and fire services are familiar with the site environs and hazards associated with the site. Service-specific information concerning the site's capabilities should be shared with the responding service.

The staff's review identified that PSAR section 11.3.10 states that ORO supporting organizations that have an emergency response role, responsibility, or authority, as described in the KU1 emergency plan, will be requested to participate in scheduled drills and exercises. The periodicity of drills and exercises will be frequent enough to ensure ERO members and ORO support organizations maintain the key skills necessary to ensure effective emergency response.

Establish Methods for Maintaining the Emergency Plan, Contacts and Arrangements, Procedures, and Evacuation Time Estimate Up to Date, Including Periodic Reviews by the Onsite and Offsite Organizations (10 CFR 50.160(b)(1)(iv)(A)(6))

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. Under 10 CFR 50.160(b)(1)(iv)(A)(6) the emergency plan must establish methods for maintaining the emergency plan, contacts and arrangements, procedures, and evacuation time estimate up to date, including periodic reviews by the on-site and offsite organizations. As described in RG 1.242, section C.7, the emergency plan should: (1) describe the process by which the emergency plan, implementing procedures, forms, and other programmatic documents are maintained at a high quality; (2) identify the individual(s) and organizations responsible for ensuring the documents are up to date; and (3) describe the frequency and process for the review of emergency plan-related documentation and for the coordination of reviews with any offsite organization that may have emergency response responsibilities.

The staff's review identified that PSAR 11.3.1 states that a process will be developed for maintaining and making changes to the KU1 emergency plan and EIPs, including methods to account for facility changes and the methods used to conduct independent reviews of the KU1 facility emergency preparedness program. These documents are considered quality documents and will be addressed by the KU1 QA program. Emergency plan maintenance and reviews will be conducted in accordance with 10 CFR 50.54(t). The methods for maintaining the emergency

plan, EIPs, and other programmatic documents, up to date, including periodic reviews by the on-site and offsite organizations, will be provided in the KU1 emergency plan.

11.3.1.2.4.1 Staff Findings and Evaluation Conclusions of the KU1 Onsite Planning Activities (10 CFR 50.160(b)(1)(iv)(A))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies is included in a PSAR for a CP application. Based on the staff review and evaluation above, the staff concludes that the KU1 CP application PSAR contains a description of preliminary plans for proposed on-site planning activities. Thus, staff find that the above KU1 PSAR preliminary on-site planning activities descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and are sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan on-site planning activities design details, necessary for staff to perform a SE for compliance with the applicable on-site planning activities criteria in 10 CFR 50.160(b)(1)(iv)(A), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OLA, should the applicant apply for an OL. Staff will perform a more detailed evaluation of the KU1 on-site planning activities final design descriptions under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B).

11.3.1.2.5 Staff Findings and Evaluation Conclusions of the KU1 Performance-based Framework (10 CFR 50.160(b)(1))

Based on the above review and evaluation of the KU1 performance-based framework PSAR preliminary plan principal design descriptions for (1) maintenance of performance, (2) emergency response performance objectives, (3) ERO sufficient emergency response function capabilities, and (4) the licensee's required on-site emergency response planning activities design, the staff determined that the preliminary performance-based framework design descriptions meet the preliminary discussion requirement of 10 CFR 50.34(a)(10)(ii) and are sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

11.3.1.3 KU1 Hazard Analysis and Significant Impediments to the Development of Emergency Plans Evaluation (10 CFR 50.160(b)(2); 10 CFR 100.21(g))

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. Under 10 CFR 50.34(b)(2) the licensee must conduct a hazard analysis of any contiguous facility, such as industrial, military, and transportation facilities, and include any credible hazard into the licensee's emergency preparedness program that would adversely impact the implementation of emergency plans.

The staff used the guidance of RG 1.242, section C.9, to review and evaluate the KU1 emergency preparedness preliminary plan design descriptions for requisite PSAR minimum design information. This guidance states that the emergency plan should describe the results of a hazard analysis of any contiguous or nearby facility in which a credible hazard has been identified that would adversely impact the implementation of the emergency plans. It also says that the emergency plan should describe planning activities or emergency response functions that will address any credible hazard that would adversely impact the implementation of emergency plans.

The staff's review identified that PSAR section 2.3 provides a description of potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports located near the site. This information was used to determine whether the effects of potential hazards on-site or in the vicinity of the site should be considered as design basis hazards. For each identified hazard, the applicant assessed the nature and extent of nearby activities. This assessment considered location, distance, frequency, and potential hazards that could result in a design basis hazard. The applicant also evaluated hazards associated with nearby industrial, transportation, and military facilities. The identified potential external hazardous facilities that are displayed in PSAR Figure 2.3-1 are:

- Kemmerer Mine (located within 5 miles of the KU1 site)
- Naughton Power Plant (located within 5 miles of the KU1 site)
- Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant (located within 10 miles of the KU1 site)

The chemicals stored at the Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant were verified against the Environmental Protection Agency's Risk Management Program Guidance. This verification established that the wastewater plant did not have toxic endpoints that require any further analysis. The other nearby industrial facilities are Kemmerer Mine and Naughton Power Plant, which contain chemicals and substances that require analysis. This analysis is detailed in PSAR table 2.3-1 and includes the identification of hazardous chemicals and the maximum stored quantity reported. The disposition of the hazards associated with these chemicals is summarized in PSAR table 2.3-2.

NUREG-0800, Standard Review Plan section 13.3, section II, guidance item 17, states that 10 CFR 100.21, "Non-seismic siting criteria," paragraph (g), requires the identification of physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans. PSAR section 11.3.2.1 states that no unique physical site characteristics were identified that would significantly impede development of the KU1 emergency plan or protective action strategies proposed in the KU1 preliminary emergency plan. PSAR section 11.3.2.2 identifies, as discussed above, that there are nearby facilities that may pose hazards to the site, and also states that an analysis of the potential adverse impacts from any credible hazard(s) from these facilities on the implementation of the KU1 emergency response plan will be addressed in the emergency preparedness plan, which will be provided at the OL stage as discussed in PSAR section 11.3.

11.3.1.3.1 Staff Findings and Evaluation Conclusions of the KU1 Hazard Analysis and Evaluation and Significant Impediments to the Development of Emergency Plans (10 CFR 50.160(b)(2); 10 CFR 100.21(g))

10 CFR 50.34(a)(10) requires that a discussion of an applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50 is included in a PSAR for a CP application. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains a description of preliminary plans for proposed hazard analysis and identification of impediments to emergency plan development. Thus, staff find that the above KU1 PSAR preliminary hazard analysis and emergency plan development impediments identification descriptions meet the emergency preparedness

preliminary plan discussion requirement in 10 CFR 50.34(a)(10)(ii) and are sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

PSAR section 11.3.2.1 states that no unique physical site characteristics were identified that would significantly impede implementation of the KU1 emergency plan. Therefore, the staff determined that 10 CFR 100.21(g) is met. The final KU1 emergency plan hazard analysis and confirmation of identification of impediments to emergency plan development final design details, necessary for staff to perform a SE for compliance with the applicable regulatory criteria of 10 CFR 50.160(b)(2) will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. The staff will perform a more detailed evaluation of the KU1 facility site hazard analysis and identification of impediments to emergency plan development final design results under the FSAR requirements of 10 CFR 50.34(b)(6)(v)(B) at that time.

11.3.1.4 Ingestion Response Planning Evaluation (10 CFR 50.160(b)(4))

Under 10 CFR 50.34(a)(10) the PSAR must include a discussion of the applicant's preliminary plans for coping with emergencies based on the requirements in 10 CFR 50.160 or appendix E to Part 50. 10 CFR 50.160(b)(4) requires applicants and licensees to describe or reference in the emergency plan the capabilities that provide actions to prevent contaminated food and water from entering into the ingestion pathway. The staff used the guidance of RG 1.242, section C.3, to review the applicant's approach to meeting these requirements. RG 1.242 states that an applicant must describe in the emergency plan, the federal, State, and local resources and capabilities for protection of the ingestion pathway, in the event of a radiological emergency.

The staff's review found that PSAR section 11.3.5.1, states that federal resources that may be called to support events that may impact the ingestion pathway emergency responses include the federal Radiological Monitoring and Assessment Center, and the Advisory Team for Environment, Food and Health. The capabilities and descriptions of appropriate state and local government agencies to address the major exposure pathways associated with the ingestion of contaminated food and water to avoid exceeding ingestion protective action guideline doses will be provided in the KU1 emergency plan submittal, which would be included as part of an OL application.

11.3.1.4.1 Staff Findings and Conclusions of the KU1 Ingestion Response Planning Evaluation (10 CFR 50.160(b)(4))

10 CFR 50.34(a)(10) requires that the PSAR include a discussion of an applicant's preliminary plans for coping with emergencies. Based on the staff's review and evaluation above, the staff concludes that the KU1 CP application PSAR contains a description of preliminary plans for proposed ingestion response planning. Thus, staff determined that the above KU1 PSAR preliminary ingestion response planning descriptions meet the emergency preparedness preliminary plan discussion requirement of 10 CFR 50.34(a)(10)(ii) and are sufficient for the issuance of a CP under 10 CFR 50.35, as applicable.

The final KU1 emergency plan ingestion response planning design details, necessary for staff to perform a SE for compliance with the applicable ingestion response planning criteria in 10 CFR 50.160(b)(4), will be provided by the applicant through the submittal of the final KU1 emergency plan and FSAR as part of the KU1 OL application. The staff will perform a more detailed evaluation of the KU1 facility ingestion response planning final design details under the FSAR requirements of 10 CFR 50.34(b) at that time.

11.3.2 Conclusion

Based on the evaluations described above, the staff conclude that the information in PSAR section 11.3 is sufficient and, where applicable, consistent with applicable guidance. It, therefore, meets the regulatory requirements identified in this chapter for issuance of a CP in accordance with 10 CFR 50.34(a)(10)(ii), 50.35, and 50.40.

11.4 Aircraft Impact Assessment

Section 11.4 of the KU1 PSAR provides an overview of the aircraft impact assessment (AIA) that was performed according to 10 CFR 50.150, "Aircraft impact assessment." The KU1 AIA identifies the design features and functional capabilities intended to ensure that, with limited operator actions, functional containment remains intact, and the integrity of the spent fuel pool (SFP) is maintained following a postulated aircraft strike. The assessment for KU1 also includes an evaluation of the fuel stored in the ex-vessel storage tank.

The applicable regulatory requirements for the evaluation of the AIA are as follows:

- 10 CFR 50.150;
- 10 CFR 50.150(a)(1);
- 10 CFR 50.150(a)(2); and
- 10 CFR 50.150(b).

The applicable guidance documents for the evaluation of the AIA are as follows:

- RG 1.217, Rev. 0, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts" (ML092900004), which endorses the guidance in NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Rev. 8P (ML111440006), which provides an acceptable method for use in satisfying the NRC requirements in 10 CFR 50.150(a) regarding the assessment of aircraft impacts for new nuclear power reactors;
- RG 5.79, "Protection of Safeguards Information" (ML103270219);
- NUREG-0800, section 19.5, "Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts," (ML12276A112), which pertains to the staff review of an applicant's AIA that could provide additional inherent protection to withstand the effects of a large commercial aircraft impact; and
- DANU-ISG-2022-01.

11.4.1 Technical Evaluation

Under 10 CFR 50.150, applicants for new nuclear power plants are required to assess the potential effects of a commercial aircraft impact on the facility's design. Applicants are required to submit a description of the design features and functional capabilities in their application that

demonstrate compliance with the criteria in 10 CFR 50.150(a)(1). Applicants subject to 10 CFR 50.150 must make the complete AIA available for NRC inspection upon request, in accordance with 10 CFR 50.70, "Inspections," 10 CFR 50.71, "Maintenance of records, making of reports," and section 161(c) of the Atomic Energy Act of 1954, as amended.

The staff's evaluation of the KU1 AIA focused on the design features and functional capabilities credited by USO to demonstrate the plant's ability to withstand the effects of a commercial aircraft impact. USO AIA information is designated as safeguards information (SGI) and, as a result, most of the assessment details were withheld from the PSAR. Therefore, to inform its review of the KU1 AIA, the staff conducted a regulatory audit of the supporting documentation (ML25302A443). During the audit, USO (1) provided documentation to support the AIA, (2) justified the selection of the design basis aircraft impact event, and (3) described how the assessment addressed fire protection, structural barriers, confinement integrity, and the protection of both the MCR and the remote shutdown complex.

Based on its review and confirmation through an audit of the supporting documentation, the staff determined that the KU1 AIA was performed using a methodology consistent with the guidance in NEI 07-13, which is endorsed by RG 1.217. The methodology demonstrates that, with minimal reliance on operator actions, the plant can withstand the effects of a large commercial aircraft impact, thereby meeting the requirements of 10 CFR 50.150(a).

Conforming to the guidance, USO identified and described in its supporting documentation the specific design features demonstrating that, even with reduced reliance on operator actions, the containment remains intact, and the integrity of the SFP and the ex-vessel storage tank are preserved following an aircraft impact.

Specifically, the KU1 AIA assessed the effects of a large commercial aircraft impact on functional containment barriers and the SFP, including:

- Physical damage from the fuselage and engines,
- Shock effects from impact-induced vibration on structures, systems, and components (SSCs), and
- Fire effects resulting from aviation fuel combustion.

KU1 AIA specifies that operator actions are to be reduced to only two key actions, both of which can be performed from either the MCR or the remote shutdown facility. However, the staff notes that the operator action assessment will be completed later to support the OL application. Specifically, the USO's AIA does not yet address:

1. Identification of cues for initiating plant shutdown from the control room, and
2. The process for control room evacuation and activation of the remote shutdown facility.

The staff determined that it is reasonable to evaluate the detailed evaluation of operator actions until plant-specific procedures and emergency operating guidelines are developed at the OL stage.

Based on its review of the PSAR section 11.4, as confirmed through the audit, the staff determined that, the KU1 containment structure assessment was reasonably performed by the applicant to ensure that the containment structure would not be perforated by the impact of a large commercial aircraft representative of those used for long-distance flights and, therefore, would not compromise the reactor vessel or fuel. The assessment addressed both in-vessel and ex-vessel containment integrity, as well as the ability to establish long-term cooling through mitigating actions.

The KU1 AIA also evaluated the impact of a large commercial aircraft on the SFP wall and support structures. The assessment confirmed that such an impact would not lead to wall perforation or leakage through the SFP liner.

The staff also notes that, while the KU1 design includes features intended to limit the spread of jet fuel fires resulting from a large aircraft crash, a complete fire isolation assessment has not yet been performed as part of the AIA for the CP application. NAT-8646, "SGI Aircraft Impact Assessment," identifies the locations of credited fire barriers used to confine fire propagation; however, it does not fully document whether these barriers meet the 3-hour fire rating or provide the associated design details discussed in regulatory guidance.

Nonetheless, the staff determined this is acceptable at the CP stage due to the early phase of design development and because PDC 3, "Fire Protection," as presented in KU1 PSAR chapter 7, addresses both the 3-hour fire rating and separation requirements. The staff expects that at the OL stage, the fire protection measures associated with aircraft impact will be fully evaluated and documented in accordance with applicable AIA regulatory guidance.

In accordance with NUREG-0800, section 19.5, the staff verified that the KU1 AIA was performed by qualified USO personnel with over 20 years of relevant experience and detailed knowledge of the KU1 plant design and layout.

The staff determined that USO performed a reasonable design-specific AIA of the effects on the facility of the impact of a large, commercial aircraft in complying to 10 CFR 50.150(a)(1) as documented in NAT-8646 and NAT-8645, "Loss of Large Area." Through the audit, the staff confirmed that the assessment was performed using a conservative approach.

The staff also finds that USO satisfied the requirements of 10 CFR 50.150(a)(2) by using a large commercial aircraft as the basis for the impact scenario. This aircraft type is representative of long-distance flights within the United States and includes a typical aviation fuel load.

In addition, the staff confirmed that USO has complied with 10 CFR 50.150(b) by identifying in PSAR section 11.4, specifically in table 11.4-1, the design features and functional capabilities relied upon, along with descriptions of how they satisfy the assessment criteria.

11.4.2 Conclusion

Based on the evaluations described above, the staff finds that the information in PSAR section 11.4 is consistent with the applicable guidance and meets the regulatory requirements of 10 CFR 50.150(a) and 50.150(b).

11.5 Technical Specifications

Section 11.5 of the PSAR provides an identification and justification for the selection of variables, conditions, or other items which are determined as the result of the preliminary safety analysis and evaluation to be the probable subjects of technical specifications (TSs) for the facility with special attention given to those items which may significantly influence the final design.

The applicable regulatory requirements for the evaluation of the TSs are as follows:

- 10 CFR 50.34(a)(5)

The applicable guidance for the evaluation of the TSs are as follows:

- DANU-ISG-2022-08, "Risk-Informed Technical Specifications" (ML23277A146).

11.5.1 Technical Evaluation

The regulations in 10 CFR 50.34(a)(5), require a CP applicant to provide an identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of TSs for the facility, with special attention given to those items which may significantly influence the final design. Acceptance criterion 1 from DANU-ISG-2022-08 specifies that the reviewer should be able to reach a safety finding and address the topic[s] in the NRC's SE report if the application includes the information required by 10 CFR 50.34(a)(5).

PSAR section 11.5.1 states that the format and content of the KU1 TSs, which would be submitted with a KU1 OL application:

1. would address the requirements in 10 CFR 50.36, "Technical specifications," in accordance with 10 CFR 50.34(b)(6)(vi) requirement that TSs be included in an OL application,
2. would be consistent with guidance provided in NUREG-1431, "Standard Technical Specifications- Westinghouse Plants," Volume 1, "Specifications," Rev. 5 (ML21259A155).

PSAR table 11.5-1 provides variables and conditions that the applicant expects to be subject to TSs, based on its preliminary safety analysis described in other PSAR chapters. Table 11.5-1 also summarizes USO's basis for each listed TS variable or condition.

PSAR table 11.5-1 indicates that safety limits (SLs) will be placed on certain process variables, specifically fuel cladding temperature and peak fuel temperature, to ensure fuel and fuel cladding integrity are maintained. In lieu of specifying explicit values for limiting safety system settings (LSSs), USO states that an administrative control technical specification will be proposed to maintain a setpoint control program (SCP) consistent with Option B of TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions," Rev. 0 (ML092150990), to satisfy 10 CFR 50.36(c)(1)(ii)(A).

PSAR table 11.5-1 indicates that Limiting Condition for Operation (LCOs), which are the lowest functional capability or performance level required for safe operation of KU1 will be provided in the TSs. Proposed subjects of LCOs and their supporting bases derived from the preliminary safety analysis listed in PSAR table 11.5-1 include:

- Reactivity control systems
- Power distribution limits
- Instrumentation
- Primary heat transport system
- Reactor air cooling system
- Functional containment
- Refueling operations

PSAR table 11.5-1 provides a list of probable design features, which, if altered or modified, could have a significant effect on safety. These probable design features include a description of the site location, a description of the fuel assemblies, and a discussion of fuel storage requirements. PSAR table 11.5-1 also lists probable administrative controls relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. These probable administrative controls include delineation of responsibility, description of organization, unit staff qualifications, requirements for procedures, programs and manuals, and reporting requirements. The staff will evaluate KU1's design features and administrative controls included in its finalized TSs during the OL application review.

PSAR section 11.5.2 and PSAR table 11.5-2 describe four operational modes for the KU1 TSs, which are defined in terms of combinations of sodium temperature, control rod drive status, and reactor cooling mechanism. The staff will evaluate KU1's operational modes during the OL application review.

The staff evaluated the sufficiency of the preliminary information on KU1's TSs, as described in PSAR section 11.5, against the requirements of 10 CFR 50.34(a)(5). In its evaluation, the staff also considered the preliminary safety analysis information and discussions of the probable subjects of TSs in other PSAR chapters. The staff finds that USO's probable subjects of TSs are consistent with important parameters determined as a result of the preliminary safety analyses in the PSAR, with special attention given to items (e.g., SLs, and LCOs) that may significantly influence the final design, and that the probable subjects of TSs are supported by appropriate bases. Therefore, based on the information in PSAR section 11.5 and other PSAR chapters, the staff finds that USO's identification and justification of the preliminary subjects of TSs meets the requirements of 10 CFR 50.34(a)(5). The staff will perform a detailed evaluation of the complete and finalized TSs for KU1, including SLs, SCP, LCOs, safety-related (SR), design features, administrative controls, and operational modes during its review of a KU1 OL application. The staff will evaluate USO's proposed SCP at the OL stage to ensure that the implementation of the program was performed in accordance with Option B of TSTF-493.

11.5.2 Conclusion

Based on the evaluations described above, the staff conclude that the information in PSAR section 11.5 is sufficient and is consistent with the applicable guidance and meets the regulatory requirements (10 CFR 50.34(a)(5)) identified in this chapter for issuance of a CP in accordance with 10 CFR 50.35 and 10 CFR 50.40.

11.6 Fitness-for-Duty and Security

The CP application for the KU1, submitted by TerraPower on behalf of USO a wholly owned subsidiary of TerraPower, LLC, describes the site characteristics applicable to security and provides information to demonstrate that security plans and measures can be developed in accordance with the applicable requirements of 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," and 10 CFR 100.21(f). The applicant describes the characteristics of the proposed KU1 site before applying for an OL for operation of the unit in the CP application's PSAR within chapter 1, chapter 2, chapter 11, and section 11.6.

The KU1 site is located in the Elkol, Wyoming, 7.5-minute quadrangle in Zone 12T in Lincoln County. USO owns a 334-acre property within which the approximately 290-acre site is located. Structures and facilities, parking lots, and roads related to the site are identified in figure 1.2-1 of the PSAR. Orientation of KU1 is such that "plant north" faces true north. Plant elevation 500 feet-0 inches equals vertical elevation 6,758 feet-0 inches NAVD88.

Sections 2.1 through 2.8 of the PSAR (Rev. 1) provide information on the demographic, meteorological, hydrological, geological, and seismological characteristics of the site and the surrounding area. A description of potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports located near the KU1 site is provided in PSAR section 11.6 (with references to PSAR chapter 2) to establish whether the effects of potential hazards on-site or in the vicinity of the site should be considered as design basis hazards in accordance with 10 CFR 100.20, "Factors to be considered when evaluating sites." PSAR section 11.6 describes site characteristics to address the applicable regulatory requirements for the KU1 site to be such that adequate security plans and measures can be developed.

PSAR figure 2.2-21 identifies the EAB and Low Population Zone (LPZ) for the KU1. The EAB and LPZ are both defined by a 0.25-mile (0.4-kilometer, 400 meter) radius from the KU1 RXB center point. USO owns the KU1 property and will have the authority to determine all activities therein including exclusion and removal of personnel and property from the area. In certain places, the EAB extends beyond the KU1 property boundary (PSAR figure 2.2-21). For the areas within the EAB but outside of USO ownership, USO will maintain agreements with the appropriate affected entities as part of the KU1 Emergency Plan, as indicated in section 11.3. The agreements will ensure appropriate plans are in place that include protective actions within the areas outside of USO ownership, as warranted.

Section 11.6 of the PSAR provides an overview of the FFD program and an overview of the security plan, training and qualification plan, and cyber security.

The provisions of 10 CFR 50.34(a)(1)(i)(D) state, in part, that, "Site characteristics must comply with part 100 of this chapter."

The provisions in 10 CFR 100.21, set forth the requirements regarding non-seismic siting criteria for proposed commercial power reactor sites.

- 10 CFR 100.21(f), as it relates to the requirement that site characteristics must be such that adequate security plans and measures can be developed.

The provisions in 10 CFR 73.55 set forth the requirements for power reactor licensees and applicants to, in part establish and maintain a physical protection program, including a security organization, which will have as its objective to provide reasonable assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to public health and safety.

Acceptance criteria adequate to meet the above requirements include those set forth in:

- RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Rev. 3, (ML23348A082), as it relates to the suitability criteria for a proposed site.
- SRP Chapter 13, section 13.6.3, "Physical Security – Early Site Permit," Rev. 2 (ML15061A471).

The applicable regulatory requirements for the evaluation of the FFD and Security programs are as follows:

- 10 CFR Part 26;
- 10 CFR 50.34(a)(1)(i);
- 10 CFR 73.54, "Protection of digital computer and communication systems and networks;" and
- 10 CFR 100.21(f).

The general performance objective of 10 CFR 73.55(b)(1) is to provide "high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety." In SRM-SECY-16-0073, (ML16279A345), the Commission stated that "the concept of 'high assurance' of adequate protection found in our security regulations is equivalent to 'reasonable assurance' when it comes to determining what level of regulation is appropriate... The staff should operate under this paradigm and eliminate ambiguity on this point..." Accordingly, this document uses the term "reasonable assurance" to provide clarity on the applicable standard.

The applicable guidance for the evaluation of the FFD and Security programs are as follows:

- RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Rev. 3 (ML12188A053),
- RG 5.66, "Access Authorization Program for Nuclear Power Plants," Rev. 2 (ML112060028),
- RG 5.71, "Cyber Security Programs for Nuclear Facilities" Rev. 1 (ML22258A204),
- RG 5.79, "Protection of Safeguards Information" (ML103270219),
- RG 5.84, "Fitness-for-Duty Programs at New Reactor Construction Sites" Rev. 0 (ML15083A412),

- DANU-ISG-2022-01 contains sections titled, “Fitness for Duty,” “Security Plans,” and “Cyber Security Plan,” which provides the application and staff review guidance,
- NUREG–0800, section 13.7.2, “Fitness For Duty – Construction,” Rev. 0 (ML15111A034),
- NUREG–0800 section 13.6.3, “Physical Security – Early Site Permit and Reactor Siting Criteria,” Rev. 2 (ML14140A210),
- NUREG–0800 section 13.6.4, “Access Authorization – Operational Program,” Rev. 0 (ML15226A009).

In addition, DANU-ISG-2022-01 Appendix C, “Construction Permit Guidance,” notes that information demonstrating that site characteristics are such that adequate security plans and measures can be developed consistent with the guidance in section 2.1, “Site Characteristics and Site Parameters (Overview),” of DANU-ISG-2022-02, “Site Information.” DANU-ISG-2022-02 section 2.1 states that under 10 CFR 100.20(a) and 10 CFR 100.21(f)–(g), the CP application must confirm that the site poses no significant impediments to the development of emergency plans, that adequate security measures can be developed, and that the radiological risk to the public from potential accidents is low.

For the individuals specified in 10 CFR 26.4(e) and (g), the CP applicant must implement an FFD program during construction that meets all 10 CFR Part 26 requirements except Subparts I, “Managing Fatigue,” and K, “FFD Programs for Construction.” As discussed in DANU-ISG-2022-01 section C.17, a CP applicant that elects not to implement a Subpart K FFD program for the individuals specified in 10 CFR 26.4(f) must include those individuals in the FFD program for the individuals specified in 10 CFR 26.4(e) and (g).

Applicable guidance for the security plan can be found in DANU-ISG-2022-01, section 8 and 9.

To the extent that security plans, training and qualification plans, safeguards contingency plans and access authorization plans are provided at the CP stage applicable regulatory guidance for the review of these plans can be found in the following documents:

- RG 5.71
- NUREG-0800, section 13.6.6, “Cyber Security Plan” Rev. 0 (ML102630477).

11.6.1 Technical Evaluation

In conducting the technical evaluation of the information contained in PSAR section 11.6 the staff also reviewed the pertinent information and figures contained in the PSAR sections 2.1-2.8. In addition, the staff reviewed pertinent information contained in the PSAR and supplemental information provided by the applicant. The staff review focused on: (1) whether the information in the application meets the requirements stated in 10 CFR 100.21(f) to demonstrate that the site characteristics and potential hazards do not present impediments to preclude the development of adequate security plans and measures ; and (2) that the applicant has considered the applicable physical protection requirements stated in 10 CFR 73.55 in the selection of the site and its proposed layout;. These characteristics were analyzed as to meet the guidance in NUREG–0800.

11.6.1.1 Site Location

In PSAR section 11.6, the applicant states the development of the security plan will consider the site characteristics described in chapter 2 and potential vulnerabilities will be mitigated.

The staff discussed the need for supplemental information during the pre-application readiness review. In letter TP-LIC-LET-0133 (ML24123A243), the applicant states, that:

“In accordance with 10 CFR 100.21(f), the Kemmerer Unit 1 site characteristics are such that adequate security plans and measures can be developed pursuant to 10 CFR 73[, “Physical Protection of Plants and Materials.”] The Kemmerer Power Station Unit 1 site and layout have been assessed and are sufficiently large to implement the requirements of 10 CFR 73.55, including adequate distances between safety-related structures and the required security boundaries and consideration of land-based vehicle bombs. There are no bodies of water or waterways that would require protection from waterborne vehicle bombs...Natural topography of the site and planned structures for the site do not limit spatial separation, and no highways, railroads, or waterways traverse the exclusion area, owner-controlled area, or protected area. The location of the site would not impact the availability of material, equipment, or services in support of maintaining security systems and operations...Because the detailed site layout design has not been finalized, all possible planned culverts or unattended openings that could extend from the outside to the inside of the designated protected area have not been identified. The site-specific security assessment for operations will identify and describe the specific security design features to be implemented to address planned culverts or unattended openings.”

Based on the above response, the staff determined that the information contained in the application is consistent with the requirements of 100.21(f) and provides a basis to conclude that the site characteristics described in chapter 2 of the PSAR are sufficient to reasonably determine that adequate security plans and measures can be developed.

11.6.1.2 Hazardous Material in Vicinity, Onsite, and Nearby Industrial, Military, and Transportation Facilities

In PSAR section 2.3 the applicant describes and depicts the site characteristics and potential nearby hazards to include a description of potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports located near the KU1 site. The applicant utilized the following regulatory guidance to evaluate the potential hazards, RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” Rev. 2 (ML21253A071) and RG 1.91, “Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants,” Rev. 3 (ML21260A242). The applicant evaluated whether the effects of potential hazards on-site or in the vicinity of the site should be considered as design basis hazards in accordance with 10 CFR 100.20. The assessment considered location, distance, frequency, and potential hazards that could result in a design basis hazard.

Based on the information in PSAR sections 2.3, the staff determined that the information contained in the application is consistent with the requirements of 10 CFR 100.21(f) and provides a basis to conclude that the site characteristics described in Chapter 2 of the PSAR are sufficient to reasonably determine that adequate security plans and measures can be developed.

11.6.1.3 Regional Climatology and Local Site Meteorology

In PSAR section 2.4 the applicant describes conditions within a radius of approximately 50 mi (80 km) of the site. The applicant describes the process used for the evaluation as, “[a] climatological summary of normal and extreme values of relevant meteorological parameters is presented for the first-order National Weather Service station or Automated Surface Observing System with no human augmentation.”

Based on the information in PSAR sections 2.4, the staff determined that the information contained in the application is consistent with the requirements of 10 CFR 100.21(f) and provides a basis to conclude that the site characteristics described in chapter 2 of the PSAR are sufficient to reasonably determine that adequate security plans and measures can be developed.

11.6.1.4 Floods and Low Water Conditions

In PSAR section 2.5, the applicant describes conditions that examine historical flooding in the vicinity of KU1 and summarizes the different types, and combinations of flood-producing phenomena considered in establishing the flood design basis. The impacts of local intense precipitation are also described in this section. PSAR section 2.5.1.1 states that Lincoln County, Wyoming historical flood records do not contain reports of major floods near the KU1 site.

The staff discussed the need for supplemental information during the readiness review. In letter TP-LIC-LET-0133, the applicant states, that:

“The effects of severe weather and resulting environmental conditions are not expected to challenge operations of the facility. Engineered and administrative controls are not challenged in the event of flooding or low water at the site, and security standoff distances are not impacted by flooding or low water.”

Based on the above response, the staff determined that the information contained in the application is consistent with the requirements of 10 CFR 100.21(f) and provides a basis to conclude that the site characteristics described in Chapter 2 of the PSAR are sufficient to reasonably determine that adequate security plans and measures can be developed.

11.6.1.5 Geological and Seismology

In PSAR section 2.6 the applicant describes conditions applicable to SR and non-safety-related with special treatment (NSRST) SSCs. Seismic design and analysis methodology and parameters presented in this section are evaluated in accordance with PDC 2.

Based on the information in PSAR sections 2.6, the staff determined that the information contained in the application is consistent with the requirements of 10 CFR 100.21(f) and provides a basis to conclude that the site characteristics described in chapter 2 of the PSAR are sufficient to reasonably determine that adequate security plans and measures can be developed.

11.6.1.6 Fitness for Duty

In PSAR section 11.6.1 the applicant provided a description of the FFD program that would be implemented at the KU1 reactor construction site. The staff's technical evaluation consisted of a review of the applicant's FFD program description for consistency with the requirements in 10 CFR Part 26 that apply during reactor construction and guidance contained in NUREG-0800 section 13.7.2.

During reactor construction, any individual performing an activity described in 10 CFR 26.4(f) is subject to an FFD program (i.e., constructing or directing the construction of safety- or security-related SSCs). These individuals are subject to a FFD program that meets the requirements in 10 CFR Part 26, Subpart K, "FFD Programs for Construction," unless the applicant subjects these individuals to a full FFD program (i.e., meets all requirements in 10 CFR Part 26, except those in Subparts I, and K).

Section 11.6.1 in the PSAR states that an FFD program consistent with NEI 06-06, "Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites," Rev. 6 (ML13093A340), as endorsed, with an exception, by RG 5.84, "Fitness-for-Duty Programs at New Reactor Construction Sites" Rev. 0 (ML15083A412), would be implemented prior to commencing construction of SR and security-related SSCs. RG 5.84 endorsed the methodology discussed in NEI 06-06 with an exception, as acceptable for implementing an FFD program at a new reactor construction site that meets the requirements of Subpart K of 10 CFR Part 26 for the worker population identified in 10 CFR 26.4(f). The staff therefore determined that the FFD program description contained in PSAR section 11.6.1 defines adequate FFD program measures for the worker population identified in 10 CFR 26.4(f).

During reactor construction, individuals performing any applicable activity specified in 10 CFR 26.4(e) (e.g., serving as security personnel required by the NRC, until receipt of special nuclear material in the form of fuel assemblies; performing QA, quality control, or quality verification activities related to safety- or security-related construction activities; supervising or managing the construction of safety- or security-related SSCs) and 10 CFR 26.4(g) (e.g., FFD program personnel involved in the day-to day operations of the FFD program, such as persons who make determinations of fitness, who notify individuals to appear for testing, or who collect specimens for drug and alcohol testing). For any individual performing an applicable activity in 10 CFR 26.4(e) or (g), the CP applicant must implement an FFD program during construction that meets all 10 CFR Part 26 requirements except Subparts I and K."

Section 11.6.1 in the PSAR states that individuals identified in 10 CFR 26.4(e) and 10 CFR 26.4(g) will be subject to an FFD program that meets the requirement of 10 CFR Part 26, Subparts A, "Administrative Provisions," through H, "Determining Fitness-for-Duty Policy Violations and Determining Fitness," N, "Recordkeeping and Reporting Requirements," and O, "Inspections, Violations, and Penalties," (i.e., a full FFD program, except for Subparts I and K). The staff therefore determined that the FFD program description in section 11.6.1 in the PSAR defines adequate FFD program measures for the worker populations identified in 10 CFR 26.4(e) and (g).

11.6.1.7 Access Authorization

In PSAR section 11.6.2 the USO provided a description of the security measures, including personnel access authorization measures, that would be implemented prior to commencing construction of SR and security-related SSCs. The technical evaluation for this area consisted

of a review of the applicant's measures for consistency with guidance in NUREG-0800 section 13.6.4. Specifically, NUREG-0800 section 13.6.4 states that an access authorization program describes the requirements for deterring unauthorized activity during construction of a new reactor. Applicable program measures (as reflected in table 1 of NUREG-0800 section 13.6.4) include the verification of identity using official photo identification, conducting demographic data checks to ensure that individuals are not suspected terrorists, and providing a copy of the construction area work policy (describing prohibited activities, behaviors, and items) to construction personnel.

Section 11.6.2 in the PSAR states that construction phase security measures, including access authorization measures, would be consistent with the guidance contained in NEI 09-01, "Security Measures During New Reactor Construction," Rev. 0 (not publicly available). As discussed in SRM-SECY-10-0137, "Staff Requirements – SECY-10-0137 – Proposed Rule: Requirements for Access Authorization and Physical Protection During Nuclear Power Plant Construction" (ML110890821), the NRC supports implementation of the personnel access authorization measures defined in NEI 09-01. The staff have determined that these measures are consistent with those described in NUREG-0800 section 13.6.4, and that PSAR section 11.6.2 therefore describes adequate access authorization measures.

11.6.1.8 Cybersecurity

The regulations in 10 CFR 73.54 require that licensees or applicants for an OL submit a cybersecurity plan. Guidance for developing a cybersecurity plan is provided in RG 5.71, and NEI 08-09, "Cyber Security Plan for Nuclear Power Reactors," Rev. 6 (ML101180437). Both guidance documents provide a cybersecurity plan template and acceptable methods and approaches for developing and establishing a cybersecurity program to satisfy the requirements of 10 CFR 73.54. The staff evaluates the applicant's/licensee's plan to provide reasonable assurance that the digital computer and communication systems and networks associated with safety, security, and emergency preparedness functions, as well as support systems and equipment, which if compromised, would adversely impact safety, security, or emergency preparedness functions, are adequately protected against cyber-attacks.

The staff reviewed pertinent information contained in the PSAR provided by the applicant and confirmed in audit. PSAR section 11.6.2 states the applicant committed to a "cyber security plan that meets the requirements of 10 CFR 73.54 that will be provided at the operating license stage as required by 10 CFR 50.34(c)(1) and (2)."

During the audit, staff engaged with USO staff regarding the cybersecurity plan requirements, including those elements that, while not formally required until the operating phase, are most effectively addressed during the design and construction phases. The applicant provided responses to audit questions focused on supply chain, identification of critical digital assets, and creation of a cybersecurity assessment team. The staff acknowledges that USO demonstrated an understanding of these requirements in their audit responses; USO referenced NEI 08-09 as the guidance being used to develop the cybersecurity program. However, USO elected not to commit to implementation activities during the construction phase as documented in the audit summary report (ML25302A443). The staff did not conduct cybersecurity-specific reviews or audits during the CP phase of the review. The staff notes that all regulatory oversight and evaluation will occur during the OL phase, including prior to the introduction of fuel on-site, consistent with 10 CFR 73.55(a)(4).

As noted above, the staff will review the KU1 cybersecurity program at the OL stage consistent with the requirements of 10 CFR 50.34(c)(1), 10 CFR 50.34(c)(2), and 10 CFR 73.54.

11.6.2 Conclusion

Based on the evaluations described above, the staff finds that the information in PSAR section 11.6 is sufficient and is consistent with the applicable guidance and meets the regulatory requirements (10 CFR 50.34(c) and 100.21(f)) identified in this chapter for issuance of a CP in accordance with 10 CFR 50.35, and 10 CFR 50.40.

11.7 References

American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," ANSI/ANS 2014 edition.

----. ANSI/ANS-3.2, "Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants," ANSI/ANS 2012 edition.

12 POST-CONSTRUCTION INSPECTION, TESTING, AND ANALYSIS PROGRAM

This chapter of the safety evaluation (SE) describes the staff's review and evaluation of the Kemmerer Power Station Unit 1 (KU1) Preliminary Safety Analysis Report (PSAR) chapter 12, post-construction inspection, testing and analysis program (PITAP) description. PSAR section 12.1 states that the primary objective of the PITAP is to demonstrate that the plant has been constructed as designed, that the safety-significant structures, systems, and components (SSC)s perform consistent with the plant design, and that activities culminating in operation at full licensed power, including initial fuel load, initial criticality, and power ascension are performed in a controlled and safe manner.

The applicable regulatory requirements for the evaluation of the PITAP are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(7);
- 10 CFR 50.35, "Issuance of construction permits;"
- 10 CFR 50.40, "Common standards;" and
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The staff has issued the following guidance relevant for the development and review of the PITAP:

- DANU-ISG-2022-06, "Advanced Reactor Content of application Project," chapter 12 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23277A144).

The applicant did not indicate that it followed DANU-ISG-2022-06 when putting together its application. Guidance documents do not present regulatory requirements; rather, they present one acceptable way of meeting the regulations. As the applicant did not use DANU-ISG-2022-06 in conducting its review, the staff will follow DANU-ISG-2022-06's guidance on reviewing a PITAP to the extent possible but will deviate from the guidance where necessary. As discussed in DANU-ISG-2022-06, "[i]f the application is for a construction permit (CP), the PITAP description can be limited to descriptions of the phase 1 inspection, testing, and verification program elements required by the quality assurance program [QAP] under 10 CFR 50.34(a)(7)." This should include a description of the scope, objectives, and programmatic controls associated with the phase 1 program. DANU-ISG-2022-06 defines phase 1 of PITAP as those testing activities conducted before fuel load.

PSAR chapter 12 states that the PITAP is composed of the following four stages of testing:

- Stage 1, construction testing performed by the construction organization
- Stage 2, component testing performed by the construction start-up group
- Stage 3, system testing performed by the construction start-up group following successful completion of the associated component testing
- Stage 4, initial start-up testing performed by licensed operators

USO provided a general description of each PITAP stage in the PSAR, but only requested review of the aspects of the PITAP required for the issuance of a CP. The relevant regulatory requirement for the PITAP description in a CP application is 10 CFR 50.34(a)(7). The staff

determined that descriptions of only stages 1, 2, and 3 of KU1 PITAP in chapter 12 occur before fuel load and, therefore constitute phase 1 as described in DANU-ISG-2022-06. The staff determined that a description of stages 1, 2, and 3 of the KU1 PITAP is all that is necessary to meet the requirements of 10 CFR 50.34(a)(7) because a CP, if granted, only authorizes activities prior to fuel load. Therefore, descriptions of the PITAP for fuel load and post-fuel load activities are not necessary in a CP application. Further, the staff determined that only the description of the PITAP's scope, objectives, and programmatic controls is necessary to satisfy 10 CFR 50.34(a)(7) because the scope, objective, and programmatic controls are the aspects of the PITAP that will feasibly be finalized at this stage of the design and application process. Therefore, the staff reviewed the applicant's descriptions of the objective, scope, and programmatic controls of stages 1, 2, and 3 of the KU1 PITAP.

12.1 Technical Evaluation

The staff evaluated the SSCs included in the scope of KU1 PITAP phase 1. PSAR section 12.1 states that the safety-related SSCs and the NSRST SSCs will be included in the scope of the PITAP. DANU-ISG-2022-06 states that PITAP scope is acceptable if it includes safety-significant SSCs, which as discussed in chapter 1 of this SE includes safety-related and NSRST SSCs. Therefore, the staff determined that the PITAP scope is acceptable.

The staff reviewed the PITAP phase 1 objectives. chapter 12 of the PSAR states that "the primary objective of the PITAP is to demonstrate that the plant has been constructed as designed, that the safety-significant SSCs perform consistent with the plant design, and that activities culminating in operation at full licensed power, including initial fuel load, initial criticality, and power ascension are performed in a controlled and safe manner." In addition, chapter 12 of the PSAR describes that the PITAP provides assurance that the operating and emergency procedures have been evaluated and demonstrated, and that the plant operations personnel are knowledgeable about the plant operating procedures and are fully prepared to operate the plant in a safe manner. DANU-ISG-2022-06 describes an acceptable objective for the PITAP. The staff determined that the PSAR description captures those portions of the objective that are relevant at the CP stage and, therefore, are acceptable.

The staff reviewed the PITAP phase 1 programmatic controls described in the PSAR. Chapter 12 of the PSAR states that the PITAP is executed under the QAP described in section 8.1 of the PSAR. Section 8.1 states that all construction-related activities will be conducted under the TerraPower Quality Assurance Program Description (QAPD) as incorporated by reference by USO. The programmatic controls relevant to the PITAP are found in the QAPD referenced in section 8.1 of the PSAR. In chapter 8 of this SE, the staff determined that the QAPD meets the requirements of 10 CFR Part 50, Appendix B, subject to CP condition 3, or uses a graded approach to quality for safety-significant SSCs. 10 CFR Part 50, Appendix B establishes required quality assurance programmatic controls for design, fabrication, construction, and testing of SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Thus, the staff determined that the QAPD contains adequate programmatic controls for conduct of the PITAP. Therefore, the staff finds that the programmatic controls described for phase 1 of the KU1 PITAP meet regulatory requirements.

12.2 Conclusion

The staff finds that PSAR chapter 12 description of the PITAP adequately describes the scope, objectives, and programmatic controls. Therefore, the PITAP description of the phase 1

inspection, tests, and analyses meets the regulatory requirements identified in this section for the issuance of a CP in accordance with 10 CFR 50.35 and 10 CFR 50.40, and, as such, the KU1 PITAP is acceptable for implementation during the design and construction of KU1.

13 RESEARCH AND DEVELOPMENT

13.1 Research and Development

Chapter 13 of the Kemmerer Power Station Unit 1 (KU1) Preliminary Safety Analysis Report (PSAR) identifies those structures, systems, and components (SSCs) that require further research and development (R&D) to confirm their design requirements.

The applicable regulatory requirements for the evaluation of R&D are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(8),
- 10 CFR 50.35(a)(3) and (b), and
- 10 CFR 50.43(e).

10 CFR 50.34(a)(8) requires an applicant for a construction permit (CP) to identify those SSCs of the facility requiring R&D to confirm the adequacy of their design. The applicant must identify and describe the R&D program which will be conducted to resolve safety questions associated with any such SSCs and provide a schedule of the R&D program showing that the safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.

10 CFR 50.35(a)(3) allows the U.S. Nuclear Regulatory Commission (NRC) to issue a CP if the Commission finds, in part, that safety features or components requiring R&D have been described and the applicant has identified, and there will be conducted, an R&D program reasonably designed to resolve any safety questions associated with such features or components. 10 CFR 50.35(b) specifies that the Commission may incorporate provisions into the CP requiring the applicant to furnish periodic reports of the progress and results of R&D programs designed to resolve safety questions.

In 10 CFR 50.43(e), the NRC lists performance demonstration requirements specific to certain applications under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," that propose nuclear reactor designs that differ significantly from light water reactor designs licensed before 1997 or use simplified, inherent, passive, or other innovative means to accomplish their safety functions. The staff notes that while 10 CFR 50.43(e) is not applicable to a CP application, completion of the R&D activities discussed in the PSAR may be, in part, necessary for the required performance demonstrations at the operating license stage. As such, the staff considered the R&D activities in contemplation of ensuring a clear path to demonstrating compliance with 10 CFR 50.43(e) at the operating license phase. The regulation in 10 CFR 50.43(e)(1) states that the NRC will approve applications for such a reactor design only if:

- (i) the performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;
- (ii) interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and

- (iii) sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences (including equilibrium core conditions).

Additionally, 10 CFR 50.43(e)(2) allows the testing of a prototype plant over a sufficient range of conditions to meet the testing requirements.

Applicable guidance on the staff's review of R&D activities can be found in DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," Appendix C (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23277A139). The guidance specifies that the staff should review identified R&D program plans necessary to resolve safety questions associated with safety features and components. In consideration of the requirements of 10 CFR 50.43(e), the guidance states that the staff's review should consider new safety or security features that differ from existing designs for operating reactors or that use simplified, inherent, or passive means to accomplish their safety functions. The staff should verify that the testing ensures the new features will perform as predicted, provide for the collection of sufficient data to validate computer codes, and show the effects of system interactions are acceptable.

Consistent with the requirements of 10 CFR 50.34, "Contents of applications; technical information," and 10 CFR 50.35, "Issuance of construction permits," the staff's review should verify that the R&D activities are completed on a schedule to resolve safety questions at or before the latest date stated in the application for completion of construction of the facility. Also, the guidance further states that staff should ensure that the applicant has provided a summary description of preoperational and startup testing that is planned for each unique or first-of-a-kind principal design feature that may be included in the facility design. The staff may accept information, as applicable, that is sufficient to credit previously performed testing for identical unique or first-of-a-kind design features at other NRC-licensed production facilities. In addition, the staff should determine whether the R&D plans will permit the staff to make the findings required by 10 CFR 50.43(e), as discussed above.

13.1.1 Technical Evaluation

13.1.1.1 R&D Program and Scope

PSAR section 13.1 states that the R&D program is implemented consistent with the Sodium reactor testing program, which was submitted by TerraPower to the NRC on September 30, 2022, as a white paper "TerraPower, LLC, Submittal of Testing Programs," Rev. 0 (ML22273A073). The staff's feedback on the testing program was provided on March 17, 2023 (ML23074A347).

R&D activities included in the PSAR were identified using a process that considered the maturity of different SSCs and safety functions using techniques developed by the Department of Energy and Government Accountability Office, resulting in technology readiness levels (TRLs). The staff is familiar with the TRL concept and notes it is an established methodology that can be a reasonable, if somewhat subjective, indicator of the maturity of a given SSC. However, not all safety questions that must be answered through testing, analysis, etc., are reasonably addressed by the TRL concept, particularly those with cross-cutting applicability like material performance. Consistent with the limitations of the TRL concept, PSAR section 13.1 indicates that while TRL played a significant role in the determination of R&D items for inclusion

in the PSAR, it was not the sole factor considered. The applicant uses technology maturation plans (TMPs) to guide R&D activities. The PSAR indicates these TMPs are designed to prove adequate performance of critical technology elements (which can include design features, performance characteristics, materials of construction, or software) through testing, analysis, modeling, calculation, verification and validation, etc.

The staff reviewed the scope of R&D activities identified in the PSAR and determined, based on a holistic review of the SSCs and their role in plant safety documented in chapters 7 and 3 of this SE, that the applicant appropriately identified those SSCs requiring R&D to confirm the adequacy of their design pursuant to the requirements of 10 CFR 50.34(a)(8). The staff audited (ML25302A443) several TMPs as part of the review to confirm that they appropriately identified activities necessary to resolve safety questions. All R&D activities identified in the PSAR are evaluated in detail in the sections that follow.

13.1.1.2 Reactor Air Cooling Heat Transfer Performance

PSAR section 13.2.1 describes R&D activities associated with the performance of the reactor air cooling system. As discussed in section 7.2.1 of this SE, the reactor air cooling system (RAC) provides a fully passive means of heat removal. As such, the staff notes that RAC performance is one of the features that must be adequately demonstrated pursuant to 10 CFR 50.43(e) at the operating license stage.

RAC heat transfer performance depends on a number of critical characteristics, including but not limited to the reactor vessel (RV) surface temperature and emissivity, the emissivity on the inside and outside of the guard vessel (GV), and the design of the collector cylinder. The R&D activities identified in the PSAR related to this topic include design features to improve heat transfer performance (such as coatings or surface treatments, which enhance emissivity) and refinement of the analyses involving RAC, such as crediting RV liner overflow from the hot pool into the cold pool annulus during accident conditions. Based on PSAR sections 7.1.2, 7.2.1, and 13.2.1 and the letter dated October 1, 2025 (ML25274A124) (R&D supplement), the applicant states that surface roughening of the RV outer surface, GV inner and outer surface, and RAC collector cylinder will be used to increase surface emissivity to ensure the necessary heat transfer performance in a RAC-only cooling event (section 7.1.2.1.10 of this SE discusses the impact of roughening on fatigue performance of the RV and GV). The R&D activities include [[

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The R&D supplement also identifies future work involving sensitivity analysis, methods validation, and establishing a final emissivity value. The staff reviewed the information provided on the docket and noted that the plan covers important parameters associated with [[
]] to address related safety questions tied to RAC heat transfer performance.

The staff also identified several separate effects tests related to RAC performance discussed in NAT-9390-A, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," Rev. 2 (ML25211A127), which are essential to ensuring codes used to analyze RAC performance are adequately validated. Beyond these activities, the staff identified that aspects of the RAC design, including the inlet and outlet ducting (particularly in the vicinity of the RV) and the collector cylinder, must be completed before RAC performance can be demonstrated; the staff interprets the action in PSAR section 13.2.1 of "selection of final design solution" to include the completion of these aspects of the design. The PSAR states that the R&D activities

associated with RAC system performance are expected to be completed prior to completion of construction activities.

Based on the above, the staff determined that the R&D activities associated with RAC performance are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3) in that they are reasonably designed to address safety questions associated with the adequacy of heat removal using RAC and are scheduled to be completed prior to completion of construction. The R&D activities discussed in PSAR section 13.2.1 are also necessary to demonstrate compliance with 10 CFR 50.43(e), which will be reviewed at the OL stage.

13.1.1.3 In-Vessel Fuel Handling System In-Vessel Transfer Machine Grapple Finger Operation

PSAR section 13.2.2 describes R&D activities associated with the in-vessel transfer machine (IVTM) grapple finger. The IVTM is part of the in-vessel fuel handling system (FHI), which is evaluated in section 7.3.3 of this SE. The specific R&D items relate to the performance of the IVTM grapple finger, which is the part of the IVTM that interfaces with the core assembly handling socket to lift core assemblies and move them around the core.

Because the gripper forms a critical portion of the load path used to transport assemblies around the core, it is necessary to demonstrate that it is sufficiently reliable in a prototypical environment. PSAR section 13.2.2 states that the R&D testing will test the integrated performance and reliability of the grapple fingers in a representative sodium environment. A variety of tests will be conducted, including full-scale prototype testing in a sodium environment, prior to completion of construction activities. The staff concluded that the testing described will reasonably determine the integrated performance and reliability of the IVTM grapple finger because it includes full-scale testing under prototypic conditions.

Based on the above, the staff determined that the R&D activities associated with the IVTM grapple finger are reasonably designed to address safety questions associated with the performance and reliability of the IVTM grapple finger. The PSAR states that R&D activities are scheduled to be completed prior to completion of construction. As such, the staff determined the R&D activities are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3).

13.1.1.4 Intermediate Heat Transport System Sodium-Salt Heat Exchanger Interaction

PSAR section 13.2.3 describes R&D activities associated with potential interactions between sodium and salt in the Intermediate Heat Transport System Sodium-Salt Heat Exchangers (SHXs). This R&D item includes improving the understanding of sodium-salt reactions and how the SHX design should be adjusted to appropriately prevent and mitigate such reactions. Partly informed by lab testing completed in 2024, PSAR section 13.2.3 identifies that an etched diffusion bonded heat exchanger design has been selected for the SHXs. The PSAR states that this heat exchanger design and the associated R&D activities are expected to demonstrate that there are no credible failures that could result in a sodium-salt reaction or that any credible failures will result in a negligible amount of sodium and salt interacting with no credible safety impacts. The R&D activities include characterizing the sodium-salt reaction, improving understanding of the design leak prevention measure impact on heat transfer, and developing and qualifying methods of leak detection. Sodium-salt reaction characterization covers assessing the reaction rate and associated heat and gas generation. It also will cover identifying

the expected reaction byproducts, quantifying the amount of oxide generated from the reaction and quantifying the impacts of the reaction byproducts on material corrosion.

The R&D supplement describes the applicant's R&D activities and plans for the SHX design to prevent sodium-salt interactions in greater detail. The primary leak prevention measure is the

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The R&D supplement also provides additional details on the materials of construction and plans to demonstrate adequate materials performance of the SHXs. The SHXs are identified as non-safety related with special treatment (NSRST) SSCs and will be fabricated to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section VIII. The SHXs will be fabricated by diffusion bonding of chemically etched plates made from 316L stainless steel

]] Further details on fabrication and design and construction codes and standards have not been provided given the current status of the SHXs in the conceptual design phase.]]

]] During audit discussion, the applicant confirmed that the reliability and integrity management (RIM) program will address the final SHX design, materials of construction, and consideration of the exposure of portions of the SHX to the salt environment, including a degradation mechanism assessment (DMA).]]

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The staff reviewed the information provided on the docket in the PSAR and the R&D supplement as well as supporting additional information under audit. The staff notes significant uncertainty with the SHX design still being at a conceptual design level of maturity and many details needing to be identified and resolved to reach a final design. The staff also notes the identified R&D activities cover a range of important aspects to ensure a safe SHX design for managing the risk of sodium-salt reaction. The R&D plan includes characterizing the sodium-salt reaction and its impacts, which is important to determine what level of sodium-salt interaction (if any) may be able to be justified as causing negligible impacts from a safety perspective. Given the level of uncertainty associated with sodium-salt reactions, at the OL stage the staff will closely evaluate the results of the R&D activities and how they may be used to justify impacts as being negligible in light of potential cliff-edge effects given the severity of the sodium-salt reaction observed in the literature (Csejka, et al., 1989)

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Another important aspect of the plan is addressing materials performance through [[
]] as well as a DMA under the RIM program. In addition to monitoring for leaks, an important aspect to ensuring SHX integrity and reliability and preventing sodium-salt reaction is demonstrating environmental compatibility with the sodium and salt environments for diffusion bonded 316L to reduce the frequency and degree of degradation in service. The staff notes there are many details associated with the physical design of the SHXs and the design and construction codes and standards and special treatments that will need to be developed and justified as appropriate. Plans for [[
]] and a DMA on the SHX are an important activity to satisfy PDC 4, which requires safety-significant SSCs to be designed to accommodate environmental effects. The staff also notes that the SHXs will be fabricated using 316L SS, which has poorer high temperature creep properties than 316H and 304H SS specified for use in many other high temperature SSCs in the design. The staff will review the approach to high temperature long-term service of 316L (diffusion bonded or plate), including final design, construction, and performance monitoring, to ensure reasonable confidence of SHX integrity over its design life at the OL stage.

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10 CFR 50.35(b) states, in part: “[t]he Commission may, in its discretion, incorporate in any CP provisions requiring the applicant to furnish periodic reports of the progress and results of R&D programs designed to resolve safety questions.” Given the significant ongoing and planned R&D work, and the current level of design maturity of the SHXs, the staff recommends the

inclusion of a condition associated with this R&D item in the KU1 CP. This condition is a general update on activities associated with this R&D item:

Prior to the completion of construction, USO shall submit periodic reports to the NRC, with the first submitted by December 31, 2026, and continuing annually, covering the latest results and future plans for research and development activities associated with the sodium-salt heat exchanger design and sodium-salt reactions. These reports shall include activities to characterize the sodium-salt reaction, mature and develop the SHX design, and develop appropriate design features and controls needed to prevent and mitigate sodium-salt reactions. The reports shall also include research activities, such as materials testing, to improve understanding of the effects of high temperature and exposure to the sodium and salt environment on SHX materials, including weld metals and diffusion bonded material.

Based on the above, the staff determined that the R&D activities associated with potential interactions between sodium and salt and the design of the SHXs are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3) in that they are reasonably designed to address safety questions associated with the design of the SHXs to appropriately prevent and mitigate sodium-salt reactions and are scheduled to be completed prior to completion of construction as stated in the PSAR.

13.1.1.5 Core Restraint System Compressive Assembly

Section 13.2.4 of the PSAR identifies R&D activities associated with the core restraint system compressive assemblies, which support the core restraint function and enhance the core lock-up behavior as described in PSAR section 7.1.1. During audit discussions, the applicant described how the compressive assemblies enable the core to be designed such that core assemblies can be removed without excessive force during shut down, but ensure core lock-up (i.e., contact between core assemblies), as the core is brought up in power. This is necessary for the core restraint system to function as designed over the range of power operations as described in the PSAR.

The safety questions associated with the compressive assemblies relate to ensuring they can achieve the capability and reliability needed in a sodium environment. The R&D activities defined in PSAR section 13.2.4 for the compressive assemblies are to develop specific requirements and perform testing in a relevant environment. The PSAR states these R&D activities will be completed prior to the completion of construction.

Because further developing the design of the compressive assemblies and testing them in a prototypic environment would reasonably resolve the safety questions, and because the work will be completed prior to the completion of construction, the staff determined that the R&D activities are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3).

13.1.1.6 Code Validation Testing

PSAR section 13.2.5 summarizes that the topical and technical reports incorporated by reference, as discussed in PSAR chapter 1, include descriptions of testing and analyses necessary to validate the analytic methodologies. In the case of the topical reports (TRs), these plans to conduct testing and analyses in support of methodology validation were found to be acceptable in the staff's review as documented in the safety evaluations enclosed in the approved versions of the TRs. The technical report, TP-LIC-RPT-0011, "Core Design and

Thermal Hydraulic Technical Report,” Rev. 1, (ML25276A289), which is evaluated in sections 3.11 and 3.12 of this SE, also discusses validation activities expected prior to the OL. Validation of the analytic methodologies is key to ensuring that performance of the plant safety systems is adequate, so the staff considers it appropriate to identify these activities as R&D items pursuant to 10 CFR 50.34(a)(8). The activities identified in the topical and technical reports also serve to provide confidence that the applicant would be able to demonstrate compliance with 10 CFR 50.43(e) at the OL stage.

Between the evaluation of the testing in the approved TRs and the validation activities discussed in sections 3.11 and 3.12 of this SE, the staff determined that the R&D activities associated with code validation are reasonably designed to address safety questions associated with the various analytic methodologies. The PSAR states these R&D activities are scheduled to be completed prior to completion of construction. As such, the staff finds the R&D activities are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3). These activities also provide confidence that the applicant would be able to demonstrate compliance with 10 CFR 50.43(e), which will be reviewed at the OL stage.

13.1.1.7 In-Service Structural Materials Performance

PSAR section 13.3.1 identifies R&D activities associated with assuring adequate in-service structural materials performance for safety-significant SSCs included in the RIM program. This R&D item includes research activities to improve the understanding of environmental effects on materials, determination of requirements for additional materials testing for environmental compatibility, and the development of performance monitoring methods, such as material surveillance programs, inspection methods and means of access, and monitoring approaches.

The staff evaluated various aspects of this R&D item in chapters 7 and 8 of this SE and relied on this R&D item as part of the basis for the findings made in those respective sections of the SE. Section 7.1.2.1.4 of this SE evaluates the approach to address environmental effects on components in contact with sodium or cover gas as well as the preliminary DMA. Sections 7.1.2.1.6 and 7.1.2.1.10 of this SE evaluate the potential monitoring and non-destructive examination practices being considered by the applicant to ensure passive component integrity. Finally, section 8.1.3 of this SE evaluates the RIM program scope, process, use of reliability targets, and preliminary RIM strategy development.

Given the significant ongoing R&D work to ensure in-service materials performance through the RIM program, the staff recommends the inclusion of two conditions associated with this R&D item in the KU1 CP. The first condition is a general update on activities associated with this R&D item:

Prior to the completion of construction, USO shall submit periodic reports to the NRC, with the first submitted by December 31, 2026, and continuing annually, covering the latest results and future plans for research and development activities necessary to ensure adequate materials performance for SSCs included in the RIM program. These reports shall include materials testing and research activities to improve understanding of the effects of high temperature, chemistry exposure, and irradiation on materials, including weld metals. The reports shall also include activities needed to support the RIM program to mature and develop appropriate performance monitoring methods, such as surveillance coupons, inspection methods and means of access, and monitoring approaches supported by validated technical bases.

The second proposed condition is tied specifically to the DMA for environmental compatibility of safety-significant SSCs. At the CP stage, the applicant provided a preliminary DMA that partially addressed environmental effects for safety-significant SSCs in contact with sodium and did not provide a DMA to address safety-significant SSCs in contact with molten salt (partly due to the design evolution of the sodium-salt heat exchangers as discussed in sections 7.1.4 and 13.1.1.4 of this SE). Therefore, the staff recommends including a condition that the applicant provide a complete DMA, which may be based on preliminary design information, by the end of calendar year 2026:

By December 31, 2026, USO shall submit a report with a complete degradation mechanism assessment (DMA) for the reliability and integrity management program (RIM) program. This is based on research and development activities related to materials qualification for environmental compatibility of safety-significant SSCs. This complete DMA may be based on preliminary design information but shall include RIM screening criteria and a technical basis for all degradation mechanisms that could affect safety-significant SSCs, including those exposed to a molten salt environment. The report shall also include a description of how new and ongoing testing as well as performance monitoring will inform RIM program development and assure component performance considering potential degradation mechanisms, the combined effects of potential degradation mechanisms, and inherent limitations on understanding how they will evolve over the KU1 design life.

Based on the evaluation of topics associated with this R&D item in chapters 7 and 8 of this SE, the staff determined that the R&D activities associated with assuring adequate in-service structural materials performance are reasonably designed to address associated safety questions. The PSAR states these R&D activities are scheduled to be completed prior to completion of construction. As such, the staff determined the R&D activities are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3). The staff will track the progression of these activities towards the final design as part of the updates provided in the associated permit conditions and evaluate them during the OL review.

13.1.1.8 Tribological Coatings

PSAR section 13.3.2 describes R&D activities associated with the development, application, and qualification of coatings and surface treatments for tribological purposes to reduce wear, friction, and self-welding. The coating R&D plan covers a range of activities, including optimizing coating deposition procedures from historical data, testing of coating performance subject to environmental conditions, functional testing, evaluation of the impact of coating failure or delamination, and ultimately the development of specifications for coating production and post-production inspection.

Section 11 of the supplemental materials report (ML25274A130) describes the applicant's R&D activities for coating development and qualification in greater detail. Tables 11.1 and 11.2 of the supplemental materials report describe the planned application of coatings on core components, while table 11.3 of the supplemental materials report covers the expected environmental conditions based on normal operation and design basis events. These applications to core components build off prior work and experience in support of applications for the Fast Flux Test Facility. The applicant stated that the core component coatings have been preliminarily classified as safety-related (QL-1) and subject to ASME NQA-1 requirements for all work to qualify and develop coating parameters. In addition to plans for coatings on core components, the applicant is also considering the use of coatings for other SSCs in contact with sodium, such

as bearings, sealing surfaces, and wear surfaces in the primary sodium pump, fuel handling equipment, in-vessel storage racks rings, the reactor head, and the intermediate heat exchanger.

The staff reviewed the information provided on the docket and noted that the plan covers important parameters associated with developing and qualifying coatings for use in the Sodium primary system environment. Additional information reviewed during the audit confirmed and provided greater detail than the docketed information. An important consideration for the staff's review is the coating safety function and consequences of coating failure. The supplemental materials report states that the coatings of core components are considered safety-related in the preliminary design, but the applicant indicated in audit discussions that this was a conservative approach that may change with the final design. The PSAR states that the R&D item will also characterize the impact of coating failure or delamination on other systems to determine the safety consequences. The level of assurance of coating performance necessary depends on the coating function and the potential consequence of coating failure, which may vary depending on the particular SSC being coated. This will be accounted for in the staff's review at the OL stage.

For environmental effects, the staff notes USO has a reasonable testing plan to characterize separate effects, but combined effects of exposure to irradiation, sodium, and thermal cycling create more uncertainty in how the coatings will perform. For potential ex-core applications on long-lived components, longer operating times in safety-significant functions make these combined environmental effects more important to consider when determining appropriate qualification activities and performance monitoring (e.g. in-service inspection) approaches. Therefore, the staff will appropriately consider the planned operating life and service conditions, along with coating function and consequence of failure, in its review of the qualification and performance monitoring of coatings at the OL stage.

Based on the above, the staff determined that the R&D activities associated with coating development and qualification are consistent with the requirements of 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3) in that they are reasonably designed to address safety questions associated with the use of these coatings in various applications and are scheduled to be completed prior to completion of construction.

13.1.2 Conclusion

The staff finds that the applicant meets the requirements of 10 CFR 50.34(a)(8) because the applicant appropriately identified those SSCs of the facility requiring R&D to confirm the adequacy of their design, identified R&D activities needed to resolve the safety questions, and identified that the activities support the resolution of the safety questions prior to completion of the construction of the facility. The staff notes that the R&D activities are described in PSAR chapter 13, the docketed supplements, and referenced TRs. Because the scope of several R&D items is extensive, the staff recommended incorporation of three permit conditions pursuant to 10 CFR 50.35(b). Based on these considerations, the staff determined that the R&D activities are sufficient to support issuance of a CP pursuant to the requirements of 10 CFR 50.35(a)(3). Finally, the staff notes that while compliance with 10 CFR 50.43(e) is not required for a CP applicant, compliance with 10 CFR 50.43(e) at the OL stage depends on some of the activities described in this section of the SE as described in greater detail above.

13.2 References

D.A. Csejka, et al. "The Interaction of Elemental Sodium with Molten NaNO₃-KNO₃ at 873 K," J. Materials Engineering, Vol. 11, No. 4, 1989.

14 FINANCIAL QUALIFICATIONS

This chapter of the SE describes the staff's review and evaluation of U.S. SFR Owner, LLC (USO's) financial qualifications (FQs) as presented in enclosure 1, "General and Financial Information," of the Kemmerer Power Station Unit 1 (KU1) construction permit (CP) application (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24088A061) as well as "Revision to Exemption Request from 10 CFR 50.33(f) and 10 CFR Part 50 Appendix C Financial Qualification Documentation Requirements and Supplemental to the General and Financial Information of the Kemmerer Power Station Unit 1 CP Application" submitted on September 9, 2025, (ML25251A127). With certain exceptions, a Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," application is required to provide sufficient information to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with regulations in 10 CFR Part 50, the activities for which the permit or license is sought. In the case of a CP application, the applicant must submit information that demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs. The applicant shall submit, among other things, estimates of the total construction costs of the facility and related fuel cycle costs, and shall indicate the sources of funds to cover these costs.

The applicable regulatory requirements for the evaluation of the USO's financial qualifications are as follows:

- 10 CFR 50.33, "Contents of applications; general information;"
- 10 CFR 50.38, "Ineligibility of certain applicants;"
- 10 CFR 50.54, "Conditions of licenses;"
- 10 CFR 70.23, "Requirements for the approval of applications;" and
- 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements."

The staff evaluated the exemption request in section 14.1 and the financial qualifications in section 14.2 of this section.

14.1 Request for Exemption Regarding Financial Qualification

By letter dated September 9, 2025, "Revision to Exemption Request from 10 CFR 50.33(f) and 10 CFR 50 Appendix C Financial Qualification Documentation Requirements and Revision to the General and Financial Information of the Kemmerer Power Station Unit 1 Construction Permit Application" (ML25251A127) USO requested an exemption from the current financial qualification requirements in 10 CFR 50.33(f) and Appendix C to 10 CFR Part 50, "A Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Construction Permits and Combined Licenses." This exemption request superseded an exemption USO had requested in enclosure 4 of the CP application, section 4, "Financial qualifications." Per the applicant's September 9 submittal, this exemption request would apply the financial qualification standard required for 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," applicants as described in 10 CFR 70.23(a)(5) in lieu of the financial

qualification standards in 10 CFR 50.33(f) and Appendix C to 10 CFR Part 50. Specifically, USO requests that it be permitted to demonstrate that it appears to be financially qualified for the issuance of a CP rather than meeting the various requirements of 10 CFR 50.33(f) and Appendix C to 10 CFR Part 50. USO cites the exemption granted to Nuclear Innovation North America for South Texas Project Units 3 and 4 combined license (COL), as precedent for granting this exemption. USO is relying on a financial capacity plan (FCP) combined with documentation showing it possesses greater than 50 percent of its funding at the time of application to meet the financial qualification requirements under Part 70.

Pursuant to 10 CFR 50.12, "Specific exemptions," upon application from any interested person or upon its own initiative, the Commission may grant exemptions from the requirements of 10 CFR Part 50 when (1) the exemptions are authorized by law; will not present an undue risk to public health and safety; and are consistent with common defense and security; and (2) when special circumstances are present.

14.1.1 Technical Evaluation

This action would exempt USO from the financial qualification requirements in 10 CFR 50.33(f) and Appendix C to 10 CFR Part 50. As stated above, 10 CFR 50.12(a) allows the Nuclear Regulatory Commission (NRC) to grant exemptions from regulations should certain criteria be met. This exemption evaluation focuses on USO's proposal to use the 10 CFR Part 70 standard of "appears to be financially qualified." The staff evaluated whether the application meets this standard in section 14.2 of this SE.

The purpose of the financial qualification regulations for CPs in 10 CFR 50.33(f) and Appendix C to 10 CFR Part 50 is to prevent safety lapses resulting from underfunded projects during construction. As discussed in SECY-79-299, the Commission has concluded that, "although technical reviews and inspection efforts are very effective direct methods of discovering deficiencies that could affect safety, the analysis of financial qualifications is an additional method, albeit indirect, of determining an applicant's ability to satisfy safety requirements. The financial qualifications review is one element in the Commission's system of multiple and redundant safety reviews and inspections. The purpose of the financial qualifications review in this system is analogous to the overlapping protective echelons of the 'defense-in-depth' approach used in designing nuclear power plants."

The following describes the staff's evaluation of USO's exemption request and how it satisfies the provisions in 10 CFR 50.12(a):

The Exemption is Authorized by Law

The Commission has the authority to issue USO's requested exemption. The exemption would not conflict with any provision of the Atomic Energy Act (AEA) of 1954, as amended, or any other law. In particular, the NRC has broad discretion to prescribe requirements for financial qualifications, as provided in Section 182(a) of the AEA. As such, the exemption requested, in this instance, is authorized by law.

The Exemption Presents no Undue Risk to Public Health and Safety

The applicant stated, the exemption does not pertain to any NRC safety requirements that directly govern design or construction activities for KU1 and the use of the 10 CFR 70 FQ standards for KU1 construction would not compromise public health and safety. Additionally, the

applicant referenced the NRC's regulatory basis document for the proposed amendment to the financial requirement in 10 CFR Part 50 (ML15322A185), which explains how the agency's oversight programs ensure public health and safety. The document also states that an applicant demonstrating commitments for at least 50% of construction funding provides sufficient evidence of financial capacity. This discussion was also referenced by the applicant. The staff notes that the proposed rulemaking was disapproved by the Commission not for substantive reasons, but rather the Commission directed the staff to address FQs during the development of 10 CFR Part 53, "Risk Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors."

The staff evaluated the exemption from the requirements of 10 CFR 50.33(f) and Appendix C to 10 CFR 50 to determine if it presents an undue risk to the public health and safety. As discussed above, the Commission has concluded that, "although technical reviews and inspection efforts are very effective direct methods of discovering deficiencies that could affect safety, the analysis of financial qualifications is an additional method, albeit indirect, of determining an applicant's ability to satisfy safety requirements. The financial qualifications review is one element in the Commission's system of multiple and redundant safety reviews and inspections. The purpose of the financial qualifications review in this system is analogous to the overlapping protective echelons of the 'defense-in-depth' approach used in designing nuclear power plants." Further, SECY-13-0124 states the following:

Commission history and precedent has consistently shown an ongoing concern for the potential of degraded safety in the face of degraded financial qualifications. However, this history also consistently indicates a Commission belief that any nexus between safety and the NRC's review of financial qualifications is indirect and of secondary importance to ensuring public health and safety. Furthermore, it is clear that Section 182a. of the AEA grants the Commission much flexibility in the arena of financial qualifications requirements.

The staff notes that the direct means to ensure safety today include a detailed technical licensing review, the construction reactor oversight process, the vendor inspection program, and the quality assurance inspection program. Further, the staff evaluated several programs for the CP application that, if implemented, support safe construction and operation of the facility. These programs include the quality assurance program and the reliability and integrity management program evaluated in section 8.2 and section 8.1 of the SE, respectively. Additionally, the licensing modernization project (LMP) methodology used by the applicant provides a structured approach to development of the licensing basis that enables comprehensive understanding of the risk and safety-significance of facility structures, systems, and components (SSCs). This understanding enables both the applicant and the NRC to evaluate the risk and safety-significance of SSCs during construction, ensuring that resources are strategically allocated to maintain facility safety and support informed regulatory oversight of construction-related changes.

Although the applicant is requesting an exemption from the financial requirements of 10 CFR 50.33(f) and Appendix C to 10 CFR 50, it will be applying the 10 CFR 70 requirements for FQ. The staff's evaluation of FQ against the 10 CFR Part 70 requirements in section 14.2 of this SE. While the FQ criteria in 10 CFR 70 is different than 10 CFR 50, a financial review is still completed and, in combination with the LMP approach, the staff determined, based on its technical judgement, a reasonable level of defense-in-depth is maintained.

Given the staff's established oversight programs, the evaluations of the applicant's programs supporting the safe construction and operation of the facility, and the financial review for the CP application, the staff concluded that issuance of the exemption does not constitute an undue risk to public health and safety.

The Exemption is Consistent with the Common Defense and Security

The applicant stated that the exemption pertains only to the financial qualification requirements for KU1 and does not affect any security requirements for KU1. The staff reviewed the information provided in the exemption request. The staff determined that these assertions are correct because the exemption only affects financial requirements for a CP application. Therefore, the staff determined the common defense and security will not be impacted and, thus, the discussed exemption request is consistent with the common defense and security.

Special Circumstances - 10 CFR 50.12(a)(2)(ii)

The applicant asserts that application of the financial qualification requirements in 10 CFR 50.33(f) and Appendix C are not necessary to achieve their underlying purpose in this case, which the applicant asserts is to protect public health and safety by preventing safety lapses during construction from underfunded projects.

The staff determined there are special circumstances that are present in that the application of the regulation is not necessary to achieve the underlying purpose of 10 CFR 50.33(f) and 10 CFR 50 Appendix C. As previously discussed in this section, the purpose of the financial qualification regulations for CPs in 10 CFR 50.33(f) and 10 CFR 50 Appendix C is to prevent safety lapses resulting from underfunded projects during construction. Further, as discussed in SECY-79-299, the Commission has concluded that, "although technical reviews and inspection efforts are very effective direct methods of discovering deficiencies that could affect safety, the analysis of financial qualifications is an additional method, albeit indirect, of determining an applicant's ability to satisfy safety requirements. The financial qualifications review is one element in the Commission's system of multiple and redundant safety reviews and inspections. The purpose of the financial qualifications review in this system is analogous to the overlapping protective echelons of the 'defense-in-depth' approach used in designing nuclear power plants." In lieu of the Part 50 requirements the applicant proposed an alternative approach to demonstrate financial qualification using the criteria in 10 CFR 70.23(a)(5).

The staff finds that, for this applicant, safety lapses would not occur as a result of underfunding for the following reasons. The applicant is participating in the U.S. Department of Energy's (DOE) Advanced Reactor Demonstration Program (ARDP). As described in "Funding Opportunity Announcement-Advanced Reactor Demonstration Program," dated May 14, 2020, referred to as the funding opportunity announcement (FOA) in this section, the ARDP is structured as a public-private partnership in which the DOE and industry jointly share the financial burden associated with the development and deployment of advanced nuclear reactors. The ARDP provides a unique and robust financial framework that supports project continuity.

The applicant's involvement in the ARDP is detailed in its submittal dated September 9, 2025, which states that the Natrium project has been awarded approximately \$1.98 billion in federal funding. As described in the FOA, the ARDP program requires awardees to provide at least 50 percent of the total project cost from non-federal sources. Additionally, ARDP funding is disbursed on a reimbursement basis, in incremental intervals, contingent upon the applicant's

expenditure of funds. As the ARDP operates on a reimbursement model, the applicant must possess sufficient upfront capital to incur project costs prior to receiving federal reimbursement. Based on the ARDP funding and the applicant's financial qualifications evaluated by the staff in section 14.2 of this SE, the staff finds that safety lapses would not occur as a result of underfunding during the construction of the facility. Thus, application of the financial qualification requirements in 10 CFR 50.33(f) and Appendix C is not necessary to achieve the underlying purpose of the regulation in this case, and that the special circumstance in accordance with 10 CFR 50.12(a)(2)(ii) is present.

14.1.2 Exemption Conclusion

The staff has determined that the exemption does not present an undue risk to the public health and safety and is consistent with the common defense and security. The staff further concludes that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii). Based on this discussion, staff finds that this exemption request should be granted.

14.2 Construction Costs

USO provided the projected costs for the construction of the proposed KU1 facility in a proprietary enclosure to the application. The costs included estimates for total nuclear plant cost, transmission, distribution, general plant costs, and nuclear fuel inventory cost for the first core. The USO estimates and bases are proprietary and therefore are not provided in this SE.

The staff evaluated the sufficiency of the information on the KU1 construction costs and related fuel cycle costs, as described in Enclosure 1 to the CP application to ensure USO's level of understanding of the size and scope of construction of KU1, including the level of capital necessary to undertake the project and support the overall evaluation of 10 CFR 70.23(a)(5). The staff audited (ML25302A443) USO's supporting documentation to confirm the appropriateness and the accuracy of the applicant's construction cost estimate. Additionally, the cost estimate was developed in partnership with the Bechtel Power Corporation (Bechtel), who has experience in nuclear plant engineering and construction since the 1950s, giving staff additional confidence that these values are reasonable for a novel design for which there exists no comparable data.

Based on the review of the cost estimates for KU1, as confirmed during the audit, and considering the partnership with Bechtel, the staff determined that USO used an appropriate approach to demonstrate its understanding of the size and scope of construction, supporting the conclusion that the applicant appears financially qualified and that the cost estimate is reasonable.

14.2.1 Financial Qualifications

Consistent with the staff's determination on the requested exemption discussed above, the staff evaluated USO's financial qualifications using the standard described in 10 CFR 70.23(a)(5). Under this standard the staff must determine that the applicant appears to be financially qualified to engage in the proposed activities.

Background

In a staff requirements memorandum (SRM) dated April 24, 2014, the Commission approved the staff's recommendation to conduct a rulemaking and directed the staff to amend the FQ

requirements in 10 CFR Part 50 to conform to the FQ review standard in 10 CFR Part 70. In response to the SRM, the staff developed and published for public comment a draft regulatory basis (80 FR 34559; June 17, 2015). In the draft regulatory basis, the staff included preliminary draft rule language that would require applicants for a CP, operating license (OL), or COL to submit a plan for financing the construction and operation of the facility. The plan would demonstrate that the applicant has both a well-articulated understanding of the size and scope of the project it is undertaking and the capacity to obtain the necessary financing prior to the start of licensed activities (i.e., construction or operations). In addition, the staff recommended rescinding Appendix C to 10 CFR Part 50 and using license conditions for applicants with 50 percent or less available funding to ensure that adequate funding is available prior to the start of licensed activities. In the regulatory basis for the Financial Qualifications for Reactor Licensing Rulemaking, the above-discussed disapproved rulemaking, the staff proposed that an applicant with commitments for greater than 50 percent funding for proposed licensed activities has made a reasonable and sufficient demonstration of financial capacity; such an applicant would not be subject to license conditions for future verification. The staff completed the final regulatory basis and informed the Commission via memorandum dated November 16, 2016. ("Memorandum from Victor M. McCree to the Commission; Regulatory Basis for Financial Qualifications for Reactor Licensing Rulemaking (3150-AJ43)," dated November 16, 2016 (ADAMS Accession No. ML16172A261). In SRM-SECY-18-00026 PROPOSED RULE: FINANCIAL QUALIFICATIONS REQUIREMENTS FOR REACTOR LICENSING (RIN 3150-AJ43) the Commission disapproved the draft proposed rule for non-substantive reasons. The Commission also told the staff to address financial qualifications during the development of Part 53 and to ask questions in the Part 53 draft proposed rule Federal Register Notice related to the appropriate requirements for Part 50.

Evaluation

To satisfy the financial qualifications standards, USO proposed to use the financial qualification standards required by 10 CFR 70.23(a)(5) for applicants under 10 CFR 70, specifically, "that the applicant appears to be financially qualified..." in lieu of the construction financial qualification requirements in 10 CFR 50.33(f) and 10 CFR 50 Appendix C; the exemption to use this approach was evaluated in section 14.1 of this SE. To determine if the applicant appears to be financially qualified, the staff was informed by the regulatory basis in the rulemakings discussed previously. Specifically, that the applicant has both a well-articulated understanding of the size and scope of the project it is undertaking and the capacity to obtain the necessary financing prior to the start of licensed activities. In evaluating the applicant's capacity to obtain the necessary financing prior to the commencement of licensed activities, the staff also relied on the regulatory basis outlined in the proposed rulemakings, which provides guidance on assessing whether the applicant can secure more than 50 percent of the projected costs. Based on the staff's technical judgement, the approach to demonstrate a well-defined understanding of the project's size and scope, and the ability to secure more than 50% of the projected costs prior to the commencement of licensed activities, provides reasonable assurance that the applicant appears to be financially qualified.

The applicant's supplement to the General and Financial Information dated September 9, 2025, contains an FCP, including a construction cost estimate that was identical to the estimate provided in the original application. The staff's evaluation and determination of the construction cost estimate is found in section 14.2.

The staff evaluated if the applicant has the capacity to obtain 50% of the total estimated project costs to meet the 70.23(a)(5) standard of appears to be reasonably qualified. To make this

determination, the applicant calculated 50 percent of the total estimated project cost, then subtracted both the actual expenditures through July 2025 and the remaining ARDP award funding to identify the amount needed for financial surety. The staff confirmed during the audit that the applicant's inclusion of actual spending through July 2025 in its financial surety calculation was appropriate because they were costs related to the CP. Additionally, the staff determined that including the ARDP funding was appropriate, as the FQ regulations under Part 70 provide sufficient flexibility on how "appears to be reasonably qualified" is demonstrated. Further, the staff reviewed the FOA section 3.3 "Cost Sharing," and confirmed that awardees receive funding for allowable costs incurred, up to the FOA limit of 50 percent. Reimbursement is for allowable costs incurred, ensuring the specified limit is met throughout the project. Thus, given the nature of the ARDP funding mechanisms, the fact that the funding will be provided by DOE, and that USO has secured sufficient funding to meet the specified limit allowing disbursement of ARDP funding, that staff determined it reasonably demonstrates the applicant's ability to secure the allotted funds through this program.

Based on the funding commitments described in the application, and enclosure 4, additional information evaluated during the audit, including capital raised through domestic and international investment opportunities and committed funding from the ARDP program, the staff determined USO demonstrates greater than 50 percent of committed construction funding.

Accordingly, based on its review of the FCP, the staff determined that the level of detail provided by USO related to its understanding of size and scope of constructing KU1 and the capacity to obtain more than 50 percent of the total estimated project costs is sufficient to meet the requirement in 10 CFR 70.23(a)(5). Therefore, based on the current level of committed funds, the staff finds that USO "appears to be financially qualified" to construct KU1.

14.2.2 Conclusion

In summary, based on the staff's conclusions above, the staff finds that the information regarding USO's financial ability to construct KU1 is sufficient and meets the 10 CFR Part 70 standard of appears to be financially qualified for the proposed activities. Additionally, the staff notes that in section 2.2 of the revised general and financial information (ML25251A127), the applicant states "[i]nformation demonstrating adequate financial qualification will be provided to the NRC Staff for review before the projected start of construction as defined by 10 CFR 50.10(a)(1). Construction will not begin until Staff review is complete." Because the staff is able to make the finding that the applicant appears to be financially qualified based on information in the application as supplemented, no additional information is necessary and the staff's review in this area is complete.

14.3 Foreign Ownership, Control, or Domination

10 CFR 50.38 states "Any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to believe is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, shall be ineligible to apply for and obtain a license." Section 2.4, "Foreign Ownership, or Control," of enclosure 1 to the KU1 CP application describes the ownership and control of USO. This section states that foreign entities do not possess the ability to vote or control decisions and cannot dominate USO.

The staff evaluated USO following the guidance provided in the "Final Standard Review Plan on Foreign Ownership, Control, or Domination" published in the *Federal Register* on

September 28, 1999 (64 FR 52355), to determine whether an applicant is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

The CP application states that, "TerraPower, LLC, and its 100% owned subsidiary USO (the Applicant), are not controlled, or dominated by any alien, foreign corporation, or foreign government as defined by [64 FR 52355]." Further, the application states that, "The total percentage of foreign non-governmental private entity ownership of TerraPower, LLC, is 19.1%, comprised of the following non-U.S. parties." USO provided in the financial supplement (ML25251A127) the countries of origin for the five non-U.S. parties and what rights, if any, foreign investors have with respect to safety and security decision making. The staff confirmed in audit (ML25302A443) that, the non-U.S. parties are from, in descending order of ownership interest, **[[** and that none of these non-U.S. parties have a voting interest and cannot dominate TerraPower or USO. **]]**

The staff conducted an independent analysis, including open-source research and verification of the information provided in the CP application, and found no evidence of foreign ownership, control, or domination (FOCD). Based on its review, the staff does not know or have reason to believe that USO is owned, controlled, or dominated by a foreign interest.

In summary, based on the staff's findings above, the staff concludes that the information on FOCD is sufficient and meets 10 CFR 50.38 the applicable statutory and regulatory requirements for the issuance of a CP.

14.4 Nuclear Insurance and Indemnity

The CP application states that USO will obtain insurance in accordance with 10 CFR 50.54(w) and the requirements in 10 CFR 140.11, "Amounts of financial protection for certain reactors," prior to loading fuel. The staff will review the sufficiency of USO's intent to obtain insurance and financial protection in accordance with the Price-Anderson Act at the OL stage.

APPENDIX A – CONSTRUCTION PERMIT CONDITIONS

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that a construction permit (CP) needs to be conditioned to require that the applicant perform analysis of excavations for safety related structures at the site, submit report(s) to verify that the applicant's Integrated Decision Process has been appropriately conducted, and submit reports on the latest results and plans for research and development activities associated with the sodium-salt heat exchanger design, sodium-salt reactions, and activities necessary to ensure adequate materials performance for structures, systems, and components (SSCs) included in the reliability and integrity management program. Therefore, the staff recommends that, should the permit be granted, the CP include the conditions set forth below. Additional details on the basis for each of these conditions appears in chapter 2, "Site Characteristics," chapter 4, "Integrated Evaluations," chapter 8, "Plant Programs," and chapter 13 "Research and Development," of the Kemmerer Power Station Unit 1 CP safety evaluation.

Proposed Permit Condition	SE Section	Description
1	2.6, "Geology, Seismology, and Geotechnical Engineering"	USO shall perform detailed geologic mapping of the excavations for the safety-related engineered structures; examine and evaluate the geologic features discovered in those excavations; and, once geologic mapping information from the excavations for safety-related structures is available for examination by the NRC, notify the Director of the Office of Nuclear Reactor Regulation, or the Director's designee, as specified in 10 CFR 50.4.
2	4.3, "Integrated Decision Process (IDP) and Integrated Decision Process Panel (IDPP)"	Prior to submittal of the operating licensing application, USO shall notify the NRC within 30 days after an integrated decision-making process panel is convened for internal approval of the safety analysis, risk assessment, or defense-in-depth adequacy of the design. The notification shall include the schedule for when the supporting documentation will be available for NRC examination.
3	8.2 "Quality Assurance"	Prior to the start of construction, as defined by 10 CFR 50.10, USO shall establish and document within the PSAR a dedicated position responsible for overseeing the execution of the Quality Assurance Program Description (QAPD). This position shall have full authority to make decisions related to QAPD implementation, independent of cost or schedule considerations, including responsibility for overseeing and ensuring the implementation of audit findings. This position shall not be assigned responsibilities for implementing the quality assurance programs of other organizations supporting the design and construction of KU1.
4	13.1, "Research and Development"	Prior to the completion of construction, USO shall submit periodic reports to the NRC, with the first submitted by December 31, 2026, and continuing annually, covering the latest results and future plans for research and development activities associated with the sodium-salt heat exchanger design and sodium-salt reactions. These reports shall include activities to characterize the sodium-salt reaction, mature and develop the SHX design, and develop appropriate design features and controls needed to

		prevent and mitigate sodium-salt reactions. The reports shall also include research activities, such as materials testing, to improve understanding of the effects of high temperature and exposure to the sodium and salt environment on SHX materials, including weld metals and diffusion bonded material.
5	13.1, “Research and Development”	Prior to the completion of construction, USO shall submit periodic reports to the NRC, with the first submitted by December 31, 2026, and continuing annually, covering the latest results and future plans for research and development activities necessary to ensure adequate materials performance for SSCs included in the RIM program. These reports shall include materials testing and research activities to improve understanding of the effects of high temperature, chemistry exposure, and irradiation on materials, including weld metals. The reports shall also include activities needed to support the RIM program to mature and develop appropriate performance monitoring methods, such as surveillance coupons, inspection methods and means of access, and monitoring approaches supported by validated technical bases.
6	13.1, “Research and Development”	By December 31, 2026, USO shall submit a report with a complete degradation mechanism assessment (DMA) for the reliability and integrity management program (RIM) program. This is based on research and development activities related to materials qualification for environmental compatibility of safety-significant SSCs. This complete DMA may be based on preliminary design information but shall include RIM screening criteria and a technical basis for all degradation mechanisms that could affect safety-significant SSCs, including those exposed to a molten salt environment. The report shall also include a description of how new and ongoing testing as well as performance monitoring will inform RIM program development and assure component performance considering potential degradation mechanisms, the combined effects of potential degradation mechanisms, and inherent limitations on understanding how they will evolve over the KU1 design life.

APPENDIX B – EXEMPTIONS

B.1 Introduction

The applicant submitted the following exemption requests as part of the Kemmerer Power Station Unit 1 (KU1) construction permit application (CPA) (ML24088A084).

- High-assay low-enriched uranium (HALEU) fuel (Title 10 of the *Code of Federal Regulations* (10 CFR) 50.68, “Criticality accident requirements”)
- Emergency core cooling system (ECCS) analysis (10 CFR 50.34(a)(4) and (b)(4))
- Maintenance rule (10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.”)
- Financial qualifications (10 CFR 50.33(f) and Appendix C to 10 CFR Part 50, “A Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Construction Permits and Combined Licenses”)

The staff evaluation of the requested exemptions is addressed in this appendix, except for the financial qualifications exemption, which can be found in chapter 14 of this SE.

Additionally, the staff identified the need for exemptions related to the definition of safety-related (SR) structures, systems, and components (SSCs) based on the applicant’s use of NEI 18-04, Rev. 1 (ML19241A336). These exemptions are being considered under a staff-initiated action, as described below.

B.2 HALEU Fuel

B.2.1 Background

Under 10 CFR 50.68(a), construction permit holders must comply with either 10 CFR 70.24, “Criticality accident requirements,” or 10 CFR 50.68(b). Those choosing to comply with 10 CFR 50.68(b) may do so in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24. Per 10 CFR 50.68(b)(7), fresh fuel assemblies are limited to a nominal maximum U-235 enrichment of five weight percent. The applicant stated it will comply with 10 CFR 50.68(b). The fuel proposed for KU1 uses a higher uranium enrichment than that specified by 10 CFR 50.68(b)(7), as discussed in preliminary safety analysis report (PSAR) section 7.1.1 and in NAT-2806-A (ML24354A192), which is incorporated by reference into the application. The staff previously approved NAT-2806-A subject to limitations and conditions, and found its incorporation by reference in the PSAR to be acceptable as documented in section 3.10 of this SE.

B.2.2 Request

The requested exemption would increase the nominal U-235 enrichment identified in 10 CFR 50.68(b)(7) from 5 weight percent to less than 20 weight percent to account for the use of HALEU fuel.

B.2.3 Discussion

Pursuant to 10 CFR 50.12, "Specific exemptions," the NRC may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50, "Domestic Licensing of Production And Utilization Facilities," when: (1) the exemptions are authorized by law, will not present an undue risk to the public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Under 10 CFR 50.12(a)(2), special circumstances include, among other things, when application of the specific regulation in the particular circumstances would not serve, or is not necessary to achieve, the underlying purpose of the rule.

B.2.3.1 Exemption is authorized by law

In accordance with 10 CFR 50.12, the NRC may grant an exemption from the requirements of 10 CFR Part 50 if the exemption is authorized by law. The exemption requested in this instance is authorized by law because no other legal prohibition exists to preclude the activities that would be authorized by the exemption.

As discussed above, 10 CFR 50.68 provides an alternative to the 10 CFR 70.24 requirement to maintain a monitoring system capable of detecting a criticality. Under 10 CFR 50.68(b)(7), however, the fresh fuel assembly maximum nominal U-235 enrichment is limited to 5 weight percent. This exemption, if granted, would raise that fresh fuel assembly maximum nominal U-235 enrichment limit to less than 20 weight percent. Nothing in the Atomic Energy Act of 1954, as amended (AEA), or in any other act specifically precludes the Commission from allowing a licensee to use this alternative in conjunction with fresh fuel assemblies with a maximum nominal U-235 enrichment above 5 weight percent or specifically mandates the imposition of a 5 weight percent limit, should the Commission determine that, in the case, the facility will still be safe.

As such, the staff has determined that granting an exemption from the 10 CFR 50.68(b)(7) requirement that the fresh fuel assembly maximum nominal U-235 enrichment to allow the applicant to use the alternate requirements in 10 CFR 50.68(b) with enrichments of less than 20 percent by weight will not result in a violation of the AEA or the Commission's regulations. Therefore, the exemption is authorized by law.

B.2.3.2 Exemption would not present an undue risk to public health and safety

Because the requested exemption is only from the enrichment limit of 10 CFR 50.68(b)(7), the other portions of 10 CFR 50.68 remain applicable. As such, the applicant must, for fuel up to the requested enrichment of less than 20 percent by weight: maintain appropriate fuel handling and storage procedures in accordance with 10 CFR 50.68(b)(1); demonstrate acceptable margin to criticality limits in accordance with 10 CFR 50.68(b)(2) through (4); maintain the quantity of special nuclear material (SNM) other than fuel stored on site to less than the quantity necessary for a critical mass; and provide radiation monitors in storage and handling areas. The staff notes that a change in enrichment does not affect the requirement for procedures under 10 CFR 50.68(b)(1), the requirement to maintain SNM other than fuel below a critical mass under 10 CFR 50.68(b)(5), and the requirement for radiation monitors under 10 CFR 50.68(b)(6). The staff's evaluation therefore focuses on the equipment used to store fresh and spent fuel and the analysis used to demonstrate it remains subcritical.

Details on fresh and spent fuel handling are discussed in PSAR sections 3.14 and 7.3. Fresh fuel assemblies are stored in the new assembly preconditioning station (NAPS), which provides a hot gas environment to precondition new assemblies prior to their insertion into the core. Individual fuel assemblies are transported from the NAPS to the core using the ex-vessel handling machine (EVHM). Once irradiated, individual spent fuel assemblies are transferred using the in-vessel transfer machine (IVTM) to the in-vessel storage (IVS), a set of racks in a sodium environment in the reactor vessel as described in PSAR section 7.1.2. From there, individual spent fuel assemblies are transported using the EVHM from the IVS to the ex-vessel storage tank (EVST), which is a standalone sodium tank described in detail in PSAR section 7.3.2.1.3. Single fuel assemblies are then transported using the bottom-loading transfer cask (BLTC) from the EVST to the pool immersion cell, before being transferred to the storage racks in the water-filled spent fuel pool (SFP), which is described in PSAR section 7.3.1. As noted in PSAR section 3.14, the SFP contains storage racks with borated stainless steel but does not use soluble boron to control reactivity.

The applicant stated in the exemption request that an increase in nominal U-235 enrichment to less than 20 percent by weight would not create the potential for inadvertent criticality, and that a criticality analysis of the fuel will demonstrate fuel handling and storage to be safely subcritical with margin as specified by the other requirements of 10 CFR 50.68(b). In PSAR section 3.14 the applicant evaluated the criticality safety of the portions of the facility that include multiple fuel assemblies, including the NAPS (a dry environment with inert gas), the IVS (a molten sodium environment), the EVST (a molten sodium environment), and the SFP (a water environment). As stated in PSAR section 3.14, the criticality safety analyses model fuel as fresh fuel at the maximum enrichment with no credit for burnup. The staff determined in section 3.13 of this SE that the applicant's criticality safety methodology provides a reasonable approach for the criticality safety analyses needed to ensure that criticality is prevented, and the preliminary criticality safety evaluations included in PSAR section 3.14 provide an adequate preliminary analysis and evaluation of the SSCs that prevent inadvertent criticality per 10 CFR 50.34, "Contents of applications; technical information," and 10 CFR 50.68.

The staff notes that the statements of consideration for 10 CFR 50.68 (63 FR 63127) addressed a comment that a limit on enrichment offered no "direct safety benefit" and should not have been included in the rule. The Commission disagreed, stating that the 5 percent limit was based on "possible criticality concerns even in a dry environment" as well as concerns related to the environmental review required by the National Environmental Policy Act (NEPA). Because the criticality analyses discussed above explicitly consider enrichment up to the proposed limit of less than 20 weight percent and model various configurations for the SSCs used to store fresh and spent fuel, including both dry and wet conditions, the staff determined that the Commission's concern on inadvertent criticality for higher enrichments has been addressed. Environmental considerations are discussed below in section B.2.3.5 of this appendix.

The staff determined that there is no undue risk to public health and safety resulting from the proposed exemption because (a) the applicant's criticality analysis is acceptable to meet the requirements of 10 CFR 50.68(b)(1) through (4) in consideration of the higher fuel enrichment proposed for KU1 and (b) concerns regarding criticality in both wet and dry conditions are addressed by the applicant's criticality safety analysis.

B.2.3.3 Exemption would be consistent with common defense and security

The applicant stated that the exemption is consistent with the common defense and security because it does not affect the design, function, or operation of structures or plant equipment

necessary to maintain the secure status of the plant, nor does it impact compliance with the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials." The staff determined that changing the enrichment limit in 10 CFR 50.68(b)(7) would not inherently affect these requirements but notes that the use of HALEU fuel must be accounted for by design and by the plant security and safeguards programs. The staff review of these aspects of the KU1 application is discussed in section 11.6 of this SE, where they were determined to be acceptable. Therefore, the staff concluded that this exemption would be consistent with common defense and security.

B.2.3.4 Special circumstances present in the exemption

Special circumstances exist pursuant to 10 CFR 50.12(a)(2)(ii), because application of the regulation in these particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of 10 CFR 50.68 is to provide reasonable assurance that an inadvertent criticality event will not occur during the storage or handling of special nuclear material at 10 CFR Part 50 licensee facilities. The preamble for the final rule promulgating 10 CFR 50.68 (63 FR 63127) indicated that increases to the enrichment limit of 5 weight percent U-235 could be addressed in the future. The Commission also indicated that it was imposing this limit because of possible criticality concerns even in a dry environment. As discussed in section B.2.3.2 of this appendix, the staff determined that the applicant's criticality safety analyses are adequate to support the requirements of 10 CFR 50.68, including consideration of fuel enriched to the limit proposed in the exemption, and appropriately address the Commission's concerns regarding inadvertent criticality in both wet and dry conditions. Therefore, the staff concludes that application of 10 CFR 50.68(b)(7) in this particular circumstance is not necessary for the licensee to achieve the underlying purpose of the rule, and special circumstance are present pursuant to 10 CFR 50.12(a)(2)(ii).

B.2.3.5 Environmental Considerations

The staff considered the environmental effects of this exemption in the environmental impact assessment it completed for the construction permit application.

B.2.4 Conclusions

Accordingly, the NRC has determined that, pursuant to 10 CFR 50.12, the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances, pursuant to 10 CFR 50.12(a)(2)(ii), are present. Therefore, the NRC hereby grants USO an exemption from the requirements of 10 CFR 50.68(b)(7) to allow the applicant to follow the requirements of 10 CFR 50.68(b) in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24 with fresh fuel assembly maximum nominal U-235 enrichment of less than 20 weight percent, as requested.

B.3 ECCS Analysis

B.3.1 Background

The requirement of 10 CFR 50.34(a)(4) require an applicant for a construction permit to perform an analysis and evaluation of ECCS cooling performance in accordance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

However, the requirements of 10 CFR 50.46 are only applicable to “boiling or pressurized light-water nuclear power reactor[s] fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding.” As described in the PSAR, the Natrium design used at KU1 is a sodium-cooled fast reactor fueled with uranium-zirconium alloy slugs in cylindrical HT9 cladding. Because of this, the requirements of 10 CFR 50.46 are not applicable to KU1. The applicant requested an exemption from 10 CFR 50.34(a)(4) and 10 CFR 50.34(b)(4) to remove the requirement to perform an analysis and evaluation of ECCS cooling performance in accordance with 10 CFR 50.46 because 10 CFR 50.46 is not applicable to the reactor and fuel design of KU1. The staff notes that the requirements of 10 CFR 50.34(b)(4) are not applicable to a CP applicant, and as such the staff did not consider granting an exemption from these requirements at this time. As appropriate, the staff would consider a future request for an exemption from 10 CFR 50.34(b)(4) for KU1 if requested.

B.3.2 Request

The requested exemption would remove the words “ECCS cooling performance and” and “§ 50.46 and” from 10 CFR 50.34(a)(4) and remove the final sentence of 10 CFR 50.34(b)(4). As noted above, the staff is only considering the exemption from 10 CFR 50.34(a)(4) at this time because 10 CFR 50.34(b)(4) is not applicable to a CP applicant. The result of the exemption under consideration by the staff would therefore be to modify 10 CFR 50.34(a)(4) to the following:

A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.

B.3.3 Discussion

Pursuant to 10 CFR 50.12, the NRC may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50 when: (1) the exemptions are authorized by law, will not present an undue risk to the public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Under 10 CFR 50.12(a)(2), special circumstances include, among other things, when application of the specific regulation in the particular circumstances would not serve, or is not necessary to achieve, the underlying purpose of the rule.

B.3.3.1 Exemption is authorized by law

As stated above, 10 CFR 50.12 allows the Commission to grant exemptions from the requirements of 10 CFR Part 50 if the Commission make certain findings. Nothing in the AEA or in any other act specifically mandates that the Commission require the specific analysis covered by this exemption, should the Commission determine that, in this case, the facility will still be safe. As such, the staff has determined that granting an exemption from 10 CFR 50.34 to remove a reference to a requirement that is not applicable to the applicant’s design will not

result in a violation of the AEA, or the Commission's regulations. Therefore, the exemption is authorized by law.

B.3.3.2 Exemption would not present an undue risk to public health and safety

The requirements of 10 CFR 50.34(a)(4) were amended along with adoption of 10 CFR 50.46 in January 1974 (39 FR 1002), where the rule change to 10 CFR 50.34 provided an implementation date for applicants for construction permits and operating licenses to comply with 10 CFR 50.46. The statements of consideration for the original issuance of 10 CFR 50.46 state that the purpose of the ECCS rule itself is to ensure an adequate margin of performance of the ECCS should a design basis loss of coolant accident (LOCA) ever occur, with margin provided by the conservative acceptance criteria and acceptable features of evaluation models specified in the rule. The LOCA definition, the ECCS acceptance criteria, and the evaluation model features specified in 10 CFR 50.46 are specific to light-water reactors, and as such are not applicable to KU1 as discussed above. However, as also noted in the statements of consideration, the overarching purpose of postulating serious, highly unlikely accidents (like loss-of-coolant accidents for LWRs) is to ensure that engineered safety features (like ECCSs for LWRs) can be designed and installed to mitigate their consequences, as part of the overall defense-in-depth approach to safety.

In determining whether the exemption would present an undue risk to public health and safety, the staff considered whether such serious, highly unlikely accidents would still be analyzed to ensure the adequacy of that appropriate engineered safety features and that defense-in-depth would be maintained. For its part, the applicant stated in its request that the exemption would not present an undue risk to public health and safety because an analysis and evaluation of core cooling systems is still performed in accordance with the first sentences of 10 CFR 50.34(a)(4).

As discussed in section 1.3.3.3.2 of this SE, various plant SSCs, including the reactor air cooling system (RAC) and intermediate air cooling system (IAC), are responsible for ensuring adequate heat removal from the reactor core. PSAR chapter 3 provides a preliminary analysis and evaluation of the design and performance of these SSCs, as required by 10 CFR 50.34(a)(4). As described in PSAR section 3.4, the selection of licensing basis events (LBEs) used in the evaluation is guided by NEI 18-04, as endorsed by RG 1.233. The staff evaluated the set of LBEs provided in the PSAR and found it to be acceptable in section 3.4 of this SE. The analyzed LBEs include all event sequences whose mean frequency of occurrence is above 5×10^{-7} per reactor year, though the applicant also retains sequences with lower frequencies as "other quantified events," for incorporation into the integrated risk evaluation and assessment of cliff-edge effects. As such, the staff concluded that the PSAR analyses include the highly unlikely accidents described in the 10 CFR 50.46 statements of consideration. The applicant explicitly evaluated anticipated operational occurrences, design-basis events (DBEs), and BDBEs using the plant's engineered safety features to ensure their consequences remain within the frequency-consequence target curve. Additional evaluations are performed for design basis accidents (DBAs), which are developed based on the DBEs, to ensure safety-related equipment adequately mitigates the dose to below the 25 rem criterion from 10 CFR 50.34(a)(1)(ii)(D). The staff evaluated these analyses in chapter 3 of the SE and found them to be acceptable. Furthermore, the applicant evaluated all LBEs to ensure defense-in-depth adequacy using an integrated approach, as discussed in chapter 4 of the KU1 PSAR. The staff found this integrated defense-in-depth adequacy evaluation to be acceptable in chapter 4 of this SE.

Because the applicant performed an analysis of highly unlikely accidents which were used to ensure that engineered safety features (in the case of KU1, passive cooling systems like the RAC and IAC) are appropriately designed to mitigate the consequences of these accidents, and explicitly evaluated KU1's defense-in-depth adequacy, and because these analyses and evaluations were found to be acceptable, the staff determined that the applicant evaluated the considerations that 10 CFR 50.46 was promulgated to address. As such, the staff determined that the exemption would not present an undue risk to public health or safety.

B.3.3.3 Exemption Would be Consistent with Common Defense and Security

The applicant stated that the requested exemption is consistent with the common defense and security because it does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant, nor does it have an impact on plant security or safeguards procedures. The exemption relates to a requirement to perform analyses and evaluations of certain SSCs. The staff noted that the exemption would not affect any requirements to provide physical protection of these SSCs and the staff did not identify any related security or safeguards requirements. Based on this, the staff determined that the exemption would not affect aspects of SSC design, function, or operation related to security or safeguards and security procedures. As such, the staff concludes that the proposed exemption is consistent with the common defense and security.

B.3.3.4 Special Circumstances Present in the Exemption

The applicant stated that special circumstances are present in that the application of the regulation is not necessary to achieve the underlying purpose of the rule, consistent with 10 CFR 50.12(a)(2)(ii). As noted above, the staff identified that the underlying purpose of the rule included in the statements of consideration was to ensure that serious, highly unlikely accidents are analyzed to ensure that engineered safety features can be designed and installed to mitigate their consequences, as part of the overall defense-in-depth approach to safety. As discussed in detail in section B.3.3.4 of this appendix, the staff determined that the applicant evaluated the considerations that 10 CFR 50.46 was promulgated to address. Therefore, application of the requirement in 10 CFR 50.34(a)(4) to perform an analysis and evaluation of ECCS cooling performance following postulated LOCA accidents in accordance with the requirements of 10 CFR 50.46 is not necessary to achieve the underlying purpose of the ECCS rule. Based on this, the staff determined that special circumstances are present pursuant to 10 CFR 50.12(a)(2)(ii).

B.3.3.5 Environmental Considerations

The staff considered the environmental effects of this exemption in the environmental impact assessment it completed for the construction permit application.

B.3.4 Conclusions

Accordingly, the staff has determined that, pursuant to 10 CFR 50.12, the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances, pursuant to 10 CFR 50.12(a)(2)(ii), are present. Therefore, the staff hereby grants USO an exemption from the requirement of 10 CFR 50.34(a)(4) to perform an analysis and evaluation of ECCS cooling performance in accordance with 10 CFR 50.46, as requested.

B.4 Maintenance Rule

B.4.1 Background

The regulations in 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” require holders of operating and combined licenses to monitor the performance or condition of SSCs in a manner to provide sufficient assurance that these SSCs are capable of fulfilling their intended functions. The scope of this monitoring program is required by 10 CFR 50.65(b) to include SR SSCs and certain non-SR SSCs, both of which have prescriptive definitions in the rule.

B.4.2 Request

The requested exemption would allow an alternate scope of SSCs included in the monitoring program required by 10 CFR 50.65 to those SSCs classified as SR or NSRST using the process identified in NEI 18-04, as endorsed by RG 1.233. The applicant stated in the exemption request that special circumstances exist pursuant to 10 CFR 50.12(a)(2)(ii), in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

B.4.3 Discussion and Conclusion

The staff is not granting the requested exemption because the requirements of 10 CFR 50.65 do not apply to an applicant for or holder of a construction permit. As appropriate, the staff would consider a future request for an exemption from 10 CFR 50.65 for KU1, if requested.

However, the staff observed that there are similarities between USO’s requested exemption from 10 CFR 50.65 and the NRC-initiated exemption to use an alternate definition for safety-related structures, systems, and components (SSCs) following the NEI 18-04 process in lieu of the definitions provided in 10 CFR Parts 26 and 50. The NRC-initiated exemption is discussed in section B.5 of this appendix. In that exemption, the staff identified that the NEI 18-04 process provides an acceptable approach to classifying SSCs based on their safety-significance, and its use as specified in the in the exemption does not present an undue risk to public health and safety. The staff also identified in the NRC-initiated exemption that special circumstances exist pursuant to 10 CFR 50.12(a)(2)(ii), in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule, similar to USO’s requested exemption for 10 CFR 50.65.

B.5 Use of NEI 18-04 Definition of Safety-Related Structures, Systems, and Components

B.5.1 Background

The KU1 PSAR uses the guidance in Nuclear Energy Institute (NEI) 18-04, Rev. 1, as endorsed by Regulatory Guide (RG) 1.233, Rev. 0 (ML20091L620), to categorize SSCs according to their safety-significance. Following this process, SSCs and associated functions may have SR classifications that are different from that expected to be SR based on the 10 CFR 50.2 definition for SR SSCs. Regulatory position C.2 of RG 1.233 states, in part, that applicants referencing this RG are expected to use the terminology in NEI 18-04, that use of the term “safety-related” in NEI 18-04 is not the same as the 10 CFR 50.2 definition, and that applicants “identify exceptions to and exemptions needed from NRC regulations.” The NEI 18-04 definition and safety classification process for non-LWRs provide a risk-informed, performance-based,

and technology-inclusive means of identifying the most safety-significant SSCs at the facility, and, as noted in regulatory position C.2 of RG 1.233, the SSCs included in the “safety-related” classification for non-LWRs may not be the same as those considered safety-related for LWRs. For those portions of the 10 CFR 50.2 definition that may not be technically relevant to the Sodium reactor design (i.e., reactor coolant pressure boundary, which is only defined in 10 CFR 50.2 for boiling and pressurized water reactors), the staff considers it appropriate for a non-LWR applicant to use the NEI 18-04 process to address whether it should be considered safety-significant, based on the acceptability of the NEI 18-04 process as discussed in RG 1.233. Therefore, the staff identified and concluded that granting an exemption from certain requirements related to the 10 CFR 50.2 definition of SR components was warranted.

In addition to the regulations listed below, the staff evaluated several regulations but determined not to include them in the exemption. Regulations in 10 CFR Part 21 and 10 CFR Part 50 that involve the use of the term “basic component” rely on a similar definition to that provided in 10 CFR 50.2 for SR SSCs. However, because this term is defined in Section 223³⁰ of the Atomic Energy Act of 1954, as amended, the staff may not issue an exemption that provides for an alternative to this definition. The staff also identified that 10 CFR 50.49, 10 CFR 50.65, 10 CFR 73.54, and 10 CFR 73.77 refer to SR SSCs under the 10 CFR 50.2 definition, but are not applicable to an applicant for a construction permit. Should USO apply for an OL, these regulations would be applicable and the applicant should consider requesting an exemption for consistency with this exemption.

B.5.2 Action

NEI 18-04, Rev. 1, defines SR SSCs as

- SSCs selected by the designer from the SSCs that are available to perform the required safety functions (RSFs) to mitigate the consequences of DBEs to within the LBE frequency-consequence target, and to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions.
- SSCs selected by the designer and relied on to perform RSFs to prevent the frequency of BDBEs with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the frequency-consequence target.

While the staff endorsed the use of this definition for SR SSCs in RG 1.233, the NEI 18-04 definition is different than the definition provided in 10 CFR 50.2, as discussed above. Though definitions themselves do not create requirements, the use of an alternate definition necessitates an exemption from the regulations where that definition is used. Accordingly, the staff identified where the definitions for “safety-related SSCs,” are used in other portions of the regulations that apply the same language as those definitions, and regulations that refer to SR SSCs in 10 CFR parts 26, 50, and 73 that are relevant to an applicant of a construction permit. The following sections describe the regulations affected by the exemption as well as specific modifications needed to implement the exemption.

³⁰ Section 223(b) of the AEA states, in part: “For the purposes of this subsection, the term ‘basic component’ means a facility structure, system, component or part thereof necessary to assure— (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut-down the facility and maintain it in a safe shut-down condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in an unplanned offsite release of quantities of fission products in excess of the limits established by the Commission.”

B.5.2.1 10 CFR Part 26, “Fitness for Duty Programs”

B.5.2.1.1 Affected Regulations

10 CFR 26.3, “Scope,” defines the scope of fitness for duty program requirements.

10 CFR 26.3(c)(4) specifies that the scope includes construction permit holder (under 10 CFR Part 50), and makes several references to SR SSCs.

10 CFR 26.4, “FFD program applicability to categories of individuals,” provides requirements on the applicability of fitness for duty (FFD) programs to categories of individuals.

10 CFR 26.4(e)(2) refers to quality assurance, quality control, quality verification activities related to SR SSCs, and 10 CFR 26.4(e)(5) refers to supervising or managing construction of SR SSCs. 10 CFR 26.4(f) refers to construction of SR SSCs.

10 CFR 26.5, “Definitions,” defines safety-related SSCs as:

...for the purposes of this part, those structures, systems, and components that are relied on to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1).

10 CFR 26.403, “Written policy and procedures,” provides requirements pertaining to FFD program policies and procedures. 10 CFR 26.403(b)(2) and (3) relate to the construction of SR SSCs.

10 CFR 26.405, “Drug and alcohol testing,” provides requirements related to drug and alcohol testing for individuals involved in activities related to safety-related SSCs.

10 CFR 26.406, “Fitness monitoring,” provides a requirement for monitoring the fitness for duty of individuals constructing or directing the construction of safety-related SSCs.

10 CFR 26.407, “Behavioral observations,” provides a requirement for individuals involved in construction of safety-related SSCs to be subject to behavioral observation.

10 CFR 26.409, “Sanctions,” requires licensees and other entities implementing an FFD program to establish sanctions for FFD policy violations that prohibit certain individuals from being involved in the construction of safety-related SSCs.

10 CFR 26.419, “Suitability and fitness evaluations,” provides a requirement for licensees and other entities implementing fitness-for-duty programs to develop, implement, and maintain procedures for evaluating whether to assign individuals to construct safety- and security-related SSCs.

B.5.2.1.2 Modifications

The alternate definition of safety-related SSCs for that in 10 CFR 26.5, will be as follows for the applicant:

...for the purposes of this part, those structures, systems, and components **classified as safety-related in accordance with NEI 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1, as endorsed by Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Revision 0.**

An alternate to the language in the remaining affected portions of 10 CFR part 26 is not necessary, but the effect of the alternate definition of safety-related SSCs is evaluated below in section B.5.3.1 of this appendix.

B.5.2.2 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”

B.5.2.2.1 Affected Regulations

10 CFR 50.2 defines safety-related SSCs as follows:

Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

10 CFR 50.10, “License required; limited work authorization,” paragraphs (a)(1)(i), (iii), and (iv) define construction activities in terms of safety-related SSCs, SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function, and SSCs whose failure could cause a reactor scram or actuation of a safety-related SSC.

10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems, and components,” provides a voluntary alternative to compliance with a number of NRC regulations (identified in 10 CFR 50.69(b)(1)) for certain safety-related and nonsafety-related SSCs that perform low safety significant functions.

10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” provides requirements related to quality assurance for “all activities affecting safety-related functions” of SSCs that “prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.”

10 CFR 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” III, “Definitions,” provides the following definition:

Structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation are those necessary to assure:

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe-shutdown condition; or
- (3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of § 50.34(a)(1)."

Appendix S, Section IV, "Application to Engineering Design," also states the following relative to SSC response to the safe-shutdown earthquake:

(a)(1) Safe Shutdown Earthquake Ground Motion.

...

(ii) The nuclear power plant must be designed so that, if the Safe Shutdown Earthquake Ground Motion occurs, certain structures, systems, and components will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, applicable concurrent normal operating, functional, and accident-induced loads must be taken into account in the design of these safety-related structures, systems, and components. The design of the nuclear power plant must also take into account the possible effects of the Safe Shutdown Earthquake Ground Motion on the facility foundations by ground disruption, such as fissuring, lateral spreads, differential settlement, liquefaction, and landsliding, as required in § 100.23 of this chapter.

...

(iv) The evaluation must take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these safety-related structures, systems, and components during the Safe Shutdown Earthquake Ground Motion and under the postulated concurrent loads, provided the necessary safety functions are maintained.

B.5.2.2.2 Modifications

The alternate definition of safety-related SSCs for that in 10 CFR 50.2 will be as follows for the applicant:

Safety-related structures, systems and components means those structures, systems and components **classified as safety-related in accordance with NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1, as endorsed by Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Revision 0.**

The alternate definition for that in 10 CFR 50, Appendix S definition of “Structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation” will be as follows for the applicant:

Structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation are those **classified as safety-related in accordance with NEI 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1, as endorsed by Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Revision 0, as well as those other SSCs required to perform a safety function during and after vibratory ground motion based on their risk- and safety-significance.**

The alternate definition for that in Appendix S IV, “Application to Engineering Design,” will be as follows for the applicant:

(1) Safe Shutdown Earthquake Ground Motion.

...

(ii) The nuclear power plant must be designed so that, if the Safe Shutdown Earthquake Ground Motion occurs, certain structures, systems, and components will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, applicable concurrent normal operating, functional, and accident-induced loads must be taken into account in the design of these certain structures, systems, and components. The design of the nuclear power plant must also take into account the possible effects of the Safe Shutdown Earthquake Ground Motion on the facility foundations by ground disruption, such as fissuring, lateral spreads, differential settlement, liquefaction, and landsliding, as required in § 100.23 of this chapter.

...

(iv) The evaluation must take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these certain structures, systems, and components during the Safe Shutdown Earthquake Ground Motion and under the postulated concurrent loads, provided the necessary safety functions are maintained.

An alternate to the language in the remaining affected portions of 10 CFR Part 50 is not necessary, but the effect of the alternate the definition of safety-related SSCs is evaluated below.

B.5.2.3 10 CFR Part 73, “Physical protection of plants and materials”

10 CFR 73.2, “Definitions,” states that terms defined in 10 CFR Part 50 have the same meaning when used in this part. The NRC staff therefore evaluated 10 CFR Part 73 to identify regulations that would be affected by an alternate definition for SR SSCs in 10 CFR Part 50, as discussed below.

B.5.2.3.1 Affected Regulations

10 CFR 73.22, "Protection of Safeguards Information: Specific requirements," provides requirements related to the protection of Safeguards Information (SGI). 10 CFR 73.22(a)(1)(vii) specifically requires physical protection for documents that contain lists or locations of certain safety-related equipment.

B.5.2.3.2 Modifications

No alternate text is needed for the regulations in 10 CFR Part 73 to implement the exemption. The effect of the alternate definition of SR SSCs under 10 CFR 50.2 on the regulations in 10 CFR Part 73 is evaluated below in section B.5.3.3 of this appendix.

B.5.3 Discussion

B.5.3.1 Exemptions from 10 CFR Part 26

Pursuant to 10 CFR 26.9, "Specific exemptions," the Commission may, on its own initiative, grant exemptions from the requirements of 10 CFR Part 26 as it determines are authorized by law and will not endanger life or property or the common defense and security, and are otherwise in the public interest.

B.5.3.1.1 Exemption is authorized by law

The scope of SSCs considered to be SR is not defined in the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, as amended. The identified exemption in this instance is authorized by law, because no other prohibition of law exists to preclude the activities which would be authorized by the exemption.

As stated above, 10 CFR 26.9 allows the Commission to, on its own initiative, grant exemptions from the requirements of 10 CFR Part 26 if the Commission makes certain findings. Nothing in the AEA or in any other act specifically mandates that the Commission requires the use of the specific definition of safety-related SSCs in 10 CFR 26.5 when identifying safety-related SSCs for the purpose of complying with the above stated regulations covered by this exemption in 10 CFR Part 26.

As such, the staff has determined that granting an exemption from the requirement to apply the substantive regulations to safety-related SSCs meeting the definition in 10 CFR Part 26 for the purpose of complying with the above stated regulations in 10 CFR Part 26 and instead, use the alternate definition provided above, when identifying safety-related SSCs in the sections of 10 CFR Part 26 identified above will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, the exemption is authorized by law.

B.5.3.1.2 Exemption will not endanger life or property

As stated in RG 1.233, the NEI 18-04 definition of safety-related SSCs differs from that provided in the regulations, but "the methods described in NEI 18-04 constitute one acceptable means to... classify SSCs." RG 1.233 further states that the purpose of the safety classification aspect of NEI 18-04 is to ensure "applicants provide the necessary capabilities to perform mitigation functions and ensure the reliability of SSCs to prevent LBEs with more severe consequences." and "determine if special treatments beyond normal industrial practices are needed to ensure SSC performance of safety functions in the prevention and mitigation of LBEs." As discussed in NEI 18-04, these special treatments include provisions for the design, construction, installation,

operation, maintenance, of SSCs that provide increased assurance beyond normal industrial practices that SSCs perform their design-basis functions.

Though the NEI 18-04 definition differs from the 10 CFR 26.5 definition of safety-related SSCs, it is consistent with the underlying purpose of identifying those SSCs that are most safety-significant to ensure they have adequate assurance that they can perform their design-basis functions. A more comprehensive evaluation of the NEI 18-04 definition of safety-related and how it relates to the regulatory evaluations is provided in section B.5.3.2.2 below, with respect to the definition provided in 10 CFR 50.2 (which is identical to that provided in 10 CFR 26.5). All of the above discussed 10 CFR Part 26 regulations to which this exemption would apply are related to the FFD program. This exemption would ensure that the FFD program applies to the most safety-significant SSCs, which is consistent with the underlying intent of the definition of safety-related SSCs in 10 CFR Part 26.

As such, the proposed exemption will not endanger life or property because safety-related SSCs will continue to be defined using a process the NRC staff has determined to be acceptable and will remain subject to appropriate fitness-for-duty programs to ensure they are constructed appropriately.

B.5.3.1.3 Exemption will not endanger the common defense and security

The exemption does not affect requirements in 10 CFR Part 26 related to nuclear power reactor security; therefore, this exemption will not endanger the common defense and security. Requirements in 10 CFR Part 73 that are affected by the exemption and which could relate to the common defense and security are evaluated in section B.5.3.3 of this appendix.

B.5.3.1.4 Exemption is otherwise in the public interest

The FFD program ensures that personnel are fit to safely and competently perform their duties. This exemption is in the public interest because it would ensure that the FFD program applies to the most safety-significant SSCs.

B.5.3.2 Exemptions from 10 CFR Part 50

Pursuant to 10 CFR 50.12, the NRC may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50 when: (1) the exemptions are authorized by law, will not present an undue risk to the public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Under 10 CFR 50.12(a)(2), special circumstances include, among other things, when application of the specific regulation in the particular circumstances would not serve, or is not necessary to achieve, the underlying purpose of the rule.

B.5.3.2.1 Exemption is authorized by law

The scope of SSCs considered to be SR is not defined in the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, as amended. The identified exemption in this instance is authorized by law, because no other prohibition of law exists to preclude the activities which would be authorized by the exemption.

As stated above, 10 CFR 50.12 allows the Commission to, on its own initiative, grant exemptions from the requirements of 10 CFR Part 50 if the Commission makes certain findings.

Nothing in the AEA or in any other act specifically mandates that the Commission requires the use of the specific definition of safety-related SSCs in 10 CFR 50.2 when identifying safety-related SSCs for the purpose of complying with the above stated regulations covered by this exemption in 10 CFR Part 50.

As such, the staff has determined that granting an exemption from the requirement to apply the substantive regulations to safety-related SSCs meeting the definition in 10 CFR Part 50 for the purpose of complying with the above stated regulations in 10 CFR Part 50 and instead, use the alternate definition provided above, when identifying safety-related SSCs in the sections of 10 CFR Part 50 identified above will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, the exemption is authorized by law.

B.5.3.2.2 Exemption would not present an undue risk to public health and safety

Though the NEI 18-04 definition differs from the 10 CFR 50.2 definition of SR SSCs, it is consistent with the underlying purpose of identifying SSCs that are the most safety-significant to ensure they have adequate assurance that they can perform their design-basis functions. The 10 CFR 50.2 definition of "safety-related" contains three elements which, as noted above, relate to the maintaining the integrity of the reactor coolant pressure boundary, achieving and maintaining safe shutdown, and preventing and mitigating radiological consequences of accidents. With respect to the first and second elements, the NEI 18-04 process includes the evaluation and classification of those SSCs necessary to support three fundamental safety functions (FSFs), including control of heat generation, control of heat removal, and radionuclide retention. The staff considered that, in relating the 10 CFR 50.2 definition of "safety-related" to the discussion in NEI 18-04, a light-water reactor needs protection of the reactor coolant pressure boundary to assure control of heat removal and radionuclide retention, because the reactor coolant pressure boundary is responsible for maintaining sufficient coolant inventory to keep the reactor cooled and retain any radionuclides released from the fuel within the primary coolant. Shutdown is similarly analogous to control of heat generation, and maintaining consequences within limits is analogous to radionuclide retention. The NEI 18-04 process evaluates the SSCs necessary to achieve these FSFs and classifies them as SR if they are necessary to maintain the frequencies and consequences of DBEs and certain BDBEs below a target derived from NRC regulations, including 10 CFR 50.34 and 10 CFR Part 20. With respect to the third element, the NEI 18-04 process explicitly includes an evaluation of DBAs to ensure that their radiological consequences are maintained below the 10 CFR 50.34(a)(1)(ii)(2) dose criteria; those SSCs identified as necessary to maintain consequences below the acceptance criterion are classified as SR. Based on this, the staff determined that the NEI 18-04 process includes appropriate consideration of the key elements of the 10 CFR 50.2 definition of safety-related, particularly including the consequences of accidents. As such the staff concluded that replacing this portion of the 10 CFR 50.2 definition with the NEI 18-04 definition of safety-related, as delineated above for the applicant, would not present an undue risk to public health and safety.

The purpose of 10 CFR 50.10 is to define construction (10 CFR 50.10(a)); to provide a requirement for a license for production and utilization facilities (10 CFR 50.10(b)); to specify that construction cannot begin until a construction permit, combined license, early site permit, or limited work authorization (LWA) is issued (10 CFR 50.10(c)); to provide requirements concerning the request for, issuance of, and effect of LWAs (10 CFR 50.10(d) through (f)); and to provide requirements for implementation of a redress plan (10 CFR 50.10(g)). The staff notes that the KU1 SE recommends inclusion of a permit condition that would require USO to establish and document within the PSAR a dedicated position responsible for overseeing the

execution of the QAPD prior to the start of construction as defined by 50.10. This exemption would allow USO to conduct certain activities defined in 10 CFR 50.10(a)(1) for certain SSCs identified as non-SR (i.e., NSRST or NST) following the NEI 18-04 process prior to complying with this condition. As discussed below, the scope of the QA program mandated by 10 CFR 50 Appendix B under this exemption would conform to the definition of SR SSCs using NEI 18-04. For the same reasons discussed below with respect to 10 CFR 50 Appendix B, the staff determined granting this exemption with regard to these non-SR SSCs for 10 CFR 50.10 would not present an undue risk to public health and safety. For any SSCs the construction of which would be prevented by the license condition, the alternate definition of safety-related SSCs would not affect when construction can begin due to the terms of the condition. Therefore, the staff determined that with regard to these SSCs the exemption would not present an undue risk to public health and safety as it relates to 10 CFR 50.10.

10 CFR 50.69 is a voluntary alternative to compliance with several regulations and is not identified in the KU1 application. Because of this, the staff did not consider it in the present evaluation.

Regarding Appendix B to 10 CFR Part 50, using an alternate definition of SR SSCs does not affect the need for quality assurance (QA) for SSCs that “prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.” Specifically, SR SSCs identified under the NEI 18-04 process include those SSCs that serve to prevent or mitigate the consequences of design basis accidents to within the 10 CFR 50.34 acceptance criteria, and other SSCs as necessary to maintain consequences from design basis events within more stringent acceptance criteria (depending on frequency) and beyond-design basis events that have the potential to exceed the 10 CFR 50.34 acceptance criteria within below an acceptable frequency. The staff also notes that the QA program, which is described in PSAR section 8.1 and was found acceptable in section 8.2 of this SE, is applied to both SR and non-SR SSCs as described in PSAR section 8.1. Under this program, QA for SR SSCs meets the requirements of 10 CFR 50 Appendix B, while non-SR SSCs receive a graded approach to QA that applies controls to those characteristics or critical attributes that render the SSC a significant contributor to plant safety. Based on these considerations, the staff determined that implementation of the modified definition of SR SSCs would not present an undue risk to public health and safety as it relates to 10 CFR 50 Appendix B.

Regarding Appendix S to 10 CFR 50, the modification would provide an alternate scope of SSCs required to withstand the effects of the safe shutdown earthquake (SSE) ground motion or surface deformation to SR SSCs identified following NEI 18-04, as well as those other SSCs required to perform a safety function during and after vibratory ground motion based on their risk- and safety-significance. The changes from “safety-related” to the alternate language of “certain” SSCs in the context of the requirements for section IV of Appendix S are needed to conform to this scope. The staff considers it important for certain non-SR SSCs to be included in the scope of Appendix S because, under the risk-informed and performance-based framework of NEI 18-04, NSRST SSCs may need to perform functions following an SSE that contribute to plant safety and all SSCs should be assessed for potential seismic interactions. USO’s process for seismic design and classification is described in PSAR section 6.4. Under that process, SR SSCs are designed to withstand the effects of the SSE and assigned to one of two safety-related seismic classification levels (SCS1 or SCS2) depending on their seismic risk significance. NSRST SSCs are designed using commercial standards with additional seismic special treatments applied to provide confidence of functional performance following the SSE, by considering earthquakes that meet or exceed the SSE. NSRST SSCs are assigned to one of three non-SR seismic classifications (SCN1, SCN2, or SCN3) depending on SSC function,

importance to life safety (consistent with the commercial building code) and potential contribution to events with dose consequences. Seismic interactions between SR, NSRST, and non-safety-related with no special treatment (NST) SSCs are considered to ensure these interactions will not adversely impact the ability of SR or NSRST SSCs to perform their safety functions. The staff evaluated this process and the preliminary seismic classifications described in PSAR chapter 7 and found them to be acceptable in SE section 6.4 and SE chapter 7. Based on this, the staff determined that the modifications to use an alternate to that in 10 CFR 50 Appendix S do not present an undue risk to public health and safety.

For the reasons discussed above relative to each affected regulation in 10 CFR Part 50, the staff determined overall that the proposed exemption will not present an undue risk to public health and safety.

B.5.3.2.3 Exemption would be consistent with common defense and security

The exemption does not affect requirements in 10 CFR Part 50 related to nuclear power reactor security; therefore, this exemption will not endanger the common defense and security. Requirements in 10 CFR Part 73 that are affected by the exemption and which could relate to the common defense and security are evaluated in section B.5.3.3 of this appendix.

B.5.3.2.4 Special circumstances are present

Special circumstances exist pursuant to 10 CFR 50.12(a)(2)(ii) because application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. Applying the original definition of SR SSCs in 10 CFR 50.2, 50.10, 50.69, Appendix B, and Appendix S is unnecessary to achieve the underlying purpose of these rules because SR SSCs will continue to be defined using a process found acceptable by the staff that appropriately ensures safety, as discussed above. The requirements for SR SSCs in each section will still apply, and the modifications to use an alternate scope of 10 CFR Appendix S appropriately address non-SR SSCs that perform or could affect the ability of other SSCs to perform safety-significant functions.

B.5.3.3 Exemptions from 10 CFR Part 73

Pursuant to 10 CFR 73.5, "Specific exemptions," the Commission may, upon application of any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 73 as it determines are authorized by law and will not endanger life or property or the common defense and security, and are otherwise in the public interest.

B.5.3.3.1 Exemption is authorized by law

The scope of SSCs considered to be SR is not defined in the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, as amended. The identified exemption in this instance is authorized by law, because no other prohibition of law exists to preclude the activities which would be authorized by the exemption.

As stated above, 10 CFR 73.5 allows the Commission to, on its own initiative, grant exemptions from the requirements of 10 CFR Part 73 if the Commission makes certain findings. Nothing in the AEA or in any other act specifically mandates that the Commission requires the use of the specific definition of safety-related SSCs in 10 CFR 50.2 when identifying safety-related SSCs

for the purpose of complying with the above stated regulation covered by this exemption in 10 CFR Part 73.

As such, the staff has determined that granting an exemption from the requirement to apply the substantive regulations to safety-related SSCs meeting the definition in 10 CFR Part 50 for the purpose of complying with the above stated regulation in 10 CFR Part 73 and instead, use the alternate definition discussed above in section B.5.2.2.2, when identifying safety-related SSCs in the section of 10 CFR Part 73 identified above will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, the exemption is authorized by law.

B.5.3.3.2 Exemption will not endanger life or property or the common defense and security

10 CFR 73.22(a)(1)(vii) requires the physical protection of documents and other matter that contain lists or locations of certain safety-related physical equipment explicitly identified in the documents or other matter as vital for purposes of physical protection, as contained in security plans, contingency measures, or plant specific safeguards analyses. Though SSCs identified as NSRST using the NEI 18-04 process may perform safety- or risk-significant functions, SSCs identified as SR using the NEI 18-04 process are those most important to ensure the safety of the facility. These SSCs would continue to be included in the scope of protected information under 10 CFR 73.22. As such, the staff concluded that the proposed exemption will not endanger life or property or the common defense and security as it relates to 10 CFR 73.22.

B.5.3.3.3 Exemption is in the public interest

Safeguards information concerns the physical protection of facilities, spent fuel shipments, and other radioactive material. This exemption is in the public interest because it would ensure that relevant information is protected as safeguards information.

B.5.3.4 Environmental Considerations

The staff considered the environmental effects of this exemption in its environmental review associated with the construction permit application. After issuing the final environmental impact statement (ML25287A017), the staff identified additional conforming changes needed to align regulations with this exemption. These include certain requirements in Parts 26, 50, and 73 that were not identified in the final environmental impact statement (EIS). Because identifying the safety classification of SSCs is a programmatic process without environmental impacts, the staff determined that these changes would not affect the conclusions of the final EIS.

B.5.4 Conclusions

As discussed above, the staff determined that in accordance with 10 CFR 26.9, the exemption is authorized by law, will not endanger life or property, will not endanger the common defense and security, and is otherwise in the public interest. The staff additionally determined that in accordance with 10 CFR 50.12, the exemptions from Part 50 are authorized by law, would not present an undue risk to public health and safety, are consistent with the common defense and security, and special circumstances are present pursuant to 10 CFR 50.12(a)(2)(ii). Finally, the staff also determined that, in accordance with 10 CFR 73.5, the exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest. Therefore, the staff hereby grants USO the exemption as described above.

APPENDIX C – Report by the Advisory Committee on Reactor Safeguards



**UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001**

November 16, 2025

The Honorable David A. Wright
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

**SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE CONSTRUCTION PERMIT
APPLICATION FOR A TERRAPOWER NATRIUM REACTOR AT THE
KEMMERER POWER STATION**

Dear Chairman Wright:

During the 730th meeting of the Advisory Committee on Reactor Safeguards, November 5 through 6, 2025, we completed our review of the safety aspects of the construction permit application (CPA) for Unit 1 of the Kemmerer Power Station (KU1) and selected portions of the U.S. Nuclear Regulatory Commission (NRC) staff's associated Safety Evaluation (SE). Our TerraPower Design Center Subcommittee reviewed this matter during subcommittee meetings on October 8 through 9 and October 21 through 23, 2025. During these meetings, we had the benefit of discussions with NRC staff and representatives from TerraPower (the applicant).¹ We also had the benefit of the referenced documents. This report fulfills the requirements of Section 182b of the Atomic Energy Act, as amended.²

CONCLUSIONS AND RECOMMENDATIONS

1. The reactor design for KU1 is a TerraPower Natrium pool-type, metal-fueled, sodium-cooled fast reactor (SFR). This design includes safety enhancements when compared to prior generation SFRs, including two means of passive heat removal, two diverse means of generating scrams, and significant separation between the sodium and steam systems. The design does not credit electrical power or operator intervention to achieve a safe shutdown.
2. The KU1 CPA is the first application for a power reactor to use the Licensing Modernization Project (LMP) methodology that was endorsed by the NRC staff in 2020. This methodology focuses the safety case on those items important to overall risk with increased use of the probabilistic risk assessment (PRA). We consider the applicant's implementation of this

¹ The construction permit application was submitted by TerraPower, LLC, on behalf of US SFR Owner, LLC (USO), a wholly owned subsidiary of TerraPower. For simplicity, this letter report refers to TerraPower as the applicant.

² Section 182b of the Atomic Energy Act (AEA) states, in part, "The Advisory Committee on Reactor Safeguards shall review each application under section 103 or section 104 b. for a construction permit or an operating license for a facility and shall submit a report thereon which shall be made part of the record of the application and available to the public except to the extent that security classification prevents disclosure." The Kemmerer CPA was submitted under section 103 of the AEA.

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methodology at this stage of licensing to be acceptable and consistent with Commission policy on risk-informed, performance-based regulation for advanced reactors.

3. As noted in their SE, the staff concludes that the facility can be constructed in accordance with relevant regulations and the design bases outlined in the preliminary safety analysis report (PSAR). Detailed design, analysis, and technology qualification will be completed prior to the operating license (OL) application review. We agree with the NRC staff's assessment.
4. This letter report identifies several areas that warrant special attention during review of the OL application, including implementation of the functional containment approach, the system response to reactivity accidents, validation of the passive cooling design, completion of design features to prevent or mitigate sodium fires, seismic design, integration of the completed PRA and defense-in-depth assessments, evaluation of uncertainties, and quantification of safety margins.
5. Our review supports issuance of the construction permit for KU1.

BACKGROUND

The Kemmerer Power Station consists of one TerraPower-designed Sodium SFR rated to provide 840 megawatts thermal, along with the required power conversion equipment to generate 336 megawatts electrical (MWe) steady-state and 500 MWe peak electrical power. Key features of this plant design include the following:

- A nuclear island that includes the reactor core within a sodium pool and an intermediate heat transfer system that uses liquid sodium to transport heat from the sodium pool to a sodium-salt heat exchanger. The nuclear island also includes structures, systems, and components (SSCs) required to provide the fundamental safety functions of control of heat generation (e.g., reactivity control), control of heat removal (e.g., decay heat removal), and retention of radionuclides (e.g., containment function).
- An energy island that uses molten salt to store and transfer heat for further use, such as steam generation to power a turbine-generator. Because normal operation and fault conditions in the energy island are not expected to affect reactor safety, the applicant has designated all SSCs in the energy island as non-safety.

The PSAR, provided to support the CPA, was prepared using the LMP methodology.³ The LMP approach to safety justification is based primarily on a PRA, supported by a hazards analysis and supplemented by explicit assessment of safety margins and defense-in-depth. This CPA is the first application to be submitted to the NRC using the LMP methodology.

³ The LMP approach is described in Nuclear Energy Institute (NEI) Report NEI 18-04, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1, and was endorsed by Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Revision 0.

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DISCUSSION

Approach to Review: Our review of the safety aspects of the Kemmerer CPA was focused on areas that we identified as potentially unique, novel, and/or noteworthy in the application. These are summarized as follows:

- Implementation of Fundamental Safety Functions:
 - Control of Heat Generation: How does KU1 address the unique reactivity characteristics of SFRs, such as the potential for an increase in reactivity on core geometry changes, and what design features are included to mitigate them?
 - Control of Heat Removal: How are the different passive means of decay heat removal and their assumed reliability validated?
 - Retention of Radionuclides: The KU1 reactor is the first SFR to use a functional containment approach. What departures are made relative to the historical approach to SFR containment, and how are they justified?
- Adequacy of the overall safety case: Does this first use of the LMP methodology provide sufficient justification to issue a Construction Permit?

Control of Heat Generation:

The KU1 design uses active, passive, and inherent means of reactivity control to limit heat generation. When required by upset conditions such as plant transients, control rods can be inserted into the reactor core. The control rod drive system uses passive and active means to insert two control rod banks of diverse geometrical design into the core:

- The scram function results in passive gravity insertion of control rods into the reactor core to control heat generation. A gravity insertion will occur when reactor trip circuit breakers (RTBs) open via either of two diverse systems: (1) the reactor protection system (RPS), which removes power to the RTB undervoltage trip circuits based on any of several plant parameter sensors; or (2) as described by the applicant, the "alternative shunt trip" system that is not detailed in the PSAR, which will use parameter sensors and interfacing circuitry that are diverse from the RPS to trip the RTBs using their active shunt trip circuits. In either case, the RTBs remove power from scram system solenoid valves that then open to vent scram pistons, de-latching the control rods and allowing them to drop due to gravity.
- The driveline scram follow function results in active motor-driven insertion of the control rods when the RPS or alternative shunt trip system generates a scram signal. This feature provides defense in depth in case control rods fail to de-latch.

Inherent reactivity feedback characteristics of the reactor core also substantially contribute to the control of heat generation. As fuel and coolant temperatures rise, the combined effects of various feedback mechanisms result in net negative reactivity feedback within the core that reduces power. Their combined effect is a safe and stable power level at which heat production and heat removal are in balance and, importantly, provides a strong inherent response to counteract any unanticipated reactivity excursions.

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Past SFR designs considered the possibility that common cause failures could result in loss of the active, passive, and inherent features that act to control heat generation. Specifically, they included consideration of a "hypothetical core disruptive accident (HCDA)," where an accident, caused by failure of the active and passive features, was postulated to relocate fissile material into a more reactive configuration and thereby perturb the inherent reactivity control features. An HCDA could potentially lead to an energetic transient with consequences such as pressure-driven leakage of sodium coolant from the reactor vessel into containment.⁴

TerraPower explained that HCDAs will be shown to be incredible events as part of the OL application, and their consequences will not be specifically considered as part of plant design and safety analysis. The safety case is expected to include: (1) design features that make accidents that lead to the potential for melting/relocation of fuel extremely unlikely; and (2) characteristics of metal fuel that make energetic reactivity transients unlikely even if fuel melting and relocation were to occur, particularly when compared to other SFR designs that use oxide fuel. The Appendix to this letter report summarizes the information provided by the applicant on this issue. We plan to further evaluate the justification when we review the OL application.

Safety analyses for reactivity addition and loss of flow transients assume certain bounds on the severity of the transients; specifically, only one control rod is assumed to withdraw, and the loss of flow is assumed to be caused by loss of power to primary sodium pumps whose rotors then coast down. More severe transients than those described above can be postulated, including: 1) withdrawal of more than one rod, which is precluded by a non-safety rod withdrawal interlock; and 2) pump failures that cause a pump locked rotor condition without a coastdown, which will be mitigated by a pump design that allows enough bypass flow to establish natural circulation and prevent fuel melt. TerraPower intends to review the safety classification of preventive controls such as the rod withdrawal interlock prior to submitting an OL application. We agree that this review is warranted and consider that the evaluation of a dual locked-rotor scenario, to provide defense-in-depth for such an unlikely transient, will be an important detail for the OL application.

Control of Heat Removal:

The KU1 design uses active and passive means to control heat removal. Two diverse residual heat removal systems, the intermediate air cooling (IAC) system and reactor air cooling (RAC) system, perform these functions. The IAC system provides means that are active or passive, while the RAC system provides passive means of removing decay heat.

Both the IAC and RAC systems rely on natural circulation of sodium coolant and air to passively remove decay heat. The OL application is expected to have more information to confirm effective natural circulation flow and heat transfer from sodium systems to air. For example, models of natural circulation flow will need to be validated, as well as heat transfer from the core to the ultimate heat sink through various interfaces such as the gap between the reactor vessel and guard vessel. Additionally, both the IAC and RAC systems rely on air flow through long passages that could be blocked as a consequence of external events. A means to reliably keep such air passages open will need to be demonstrated.

⁴ As an example, a 2022 study included an assumption of 350 kilograms of sodium forced into the containment volume due to a postulated HCDA; see International Atomic Energy Agency (IAEA), "Modelling and Simulation of the Source Term for a Sodium Cooled Fast Reactor Under Hypothetical Severe Accident Conditions," IAEA TECDOC-2006, <https://www-pub.iaea.org/MTCD/Publications/PDF/TE-2006web.pdf>.

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Retention of Radionuclides:

The KU1 design performs the fundamental safety function of retaining radionuclides using a functional containment strategy that employs diverse passive barriers to ensure regulatory dose criteria and Quantitative Health Objectives are met. These barriers begin at a radionuclide source and include all SSCs between that source and the environment. The functional containment system consists of: (1) the safety-related primary functional containment boundary, defined as the minimum set of barriers encompassing the core and primary system that prevent a release of radionuclides from exceeding regulatory limits; and (2) enveloping barriers, defined as either non-safety-related or non-safety-related with special treatment SSCs that each provide a backup radionuclide retention function to the primary functional containment boundary it envelopes. Under LMP, the safety classifications for the primary and enveloping boundaries are established through their relative safety significance.

In addition to the physical barriers, an important inherent plant design feature supporting the functional containment strategy is the high boiling point of sodium (relative to operational coolant and fuel temperatures). This ensures that the reactor core remains covered by subcooled sodium at near atmospheric pressure during licensing basis events (LBEs). The low operating pressure of the primary system, including the sodium cover gas, ensures there is no significant driving force to energetically transport radionuclides away from the reactor.

The KU1 reactor is the first SFR design to use a functional containment strategy. TerraPower stated that the functional containment barriers described in the PSAR were selected to meet radiological dose criteria for the mechanistic source term associated with the range of accident scenarios considered for the CPA under LMP. This includes an assessment of the most limiting design basis accident (DBA) to support regulatory requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.⁵ Accident analyses to be performed in support of the OL application are expected to confirm adequacy of the selected barriers.

The radionuclide barriers that were selected to meet the LMP frequency/consequence guidelines share many similarities with prior SFR containment designs, such as credited physical barriers with leak-rate limits and planned testing through plant life to confirm those limits continue to be met. The principal differences when using the KU1 functional containment approach are higher leak rate limits than prior designs and non-safety designations for certain components of the enveloping barriers. When compared to prior SFR designs, the KU1 reactor has several safety enhancements as described in the Appendix, resulting in what is expected to be a net reduction in risk despite these changes in containment design. Therefore, it is reasonable to expect that the functional containment approach can be fully justified during the OL phase, pending completion of design. We expect to further review the functional containment approach at the OL stage, with continued focus on the adequacy of the supporting

⁵ 10 CFR 50.34(a)(1) states in part, "In performing this assessment, an applicant shall assume a fission product release³ from the core into the containment assuming that the facility is operated at the ultimate power level contemplated." The cited footnote 3 states, "The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products." Per option 1 of Regulatory Position C.3.b in Regulatory Guide 1.253, the staff will accept the DBA dose consequence results from an LMP-based approach to meet this regulatory requirement.

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accident analyses and the design and testing approaches as they compare to traditional containment designs.

One of the primary goals of the containment system for an SFR is to contain the effects of a fire caused by chemical interaction of sodium with air. TerraPower stated they will optimize locations in the KU1 design where two barriers are provided between the sodium coolant and air to minimize risk. For example, a guard vessel surrounds the reactor vessel to provide two barriers for sodium in the core. However, given the difficulty in fighting sodium fires, more details about mitigating barriers, sodium fire progression models, and the fire protection program are needed to support the fire risk assessment that will be performed as part of the OL application review.

Radionuclide retention during accident conditions will also depend upon the sodium-salt heat exchanger being a reliable pressure boundary. The sodium-salt heat exchanger couples the nuclear and energy islands. Ongoing research and development programs by TerraPower are addressing materials compatibility, reaction energetics, and leak detection and isolation methods. The heat exchanger design concept is innovative but remains one of the least mature elements of the plant, and continued progress in these areas will be important to the safety case at the OL stage.

Adequacy of the overall safety case:

Application of Licensing Modernization Project

Implementation of the LMP approach is an important element of the safety case. The LMP methodology is centered around use of a PRA to select LBEs, determine appropriate safety classification of SSCs, assign associated risk-informed special treatments, and determine adequacy of defense in depth. As the PRA evolves from the CP stage, the above attributes may be revised accordingly. This iterative process, as conceived in Regulatory Guide (RG) 1.233 and RG 1.253, will provide valuable information when evaluating advantages and shortcomings of an application that is based on LMP. The NRC staff stated they are evaluating this CPA for lessons learned and potential clarifications to associated NRC guidance.

TerraPower's approach to implementing the LMP guidance for defense in depth was notable. They used a methodical "defense line" approach, where every group of LBEs is assessed against the LMP five-layer defense-in-depth model to assure appropriate independence and diversity in system design and operation. TerraPower stated that the linkage between these defense lines, principal design criteria, key safety functions, and quantifiable performance measures will continue to progress through final design and the OL application. We encourage the NRC staff to further develop guidance for such approaches as they have the potential to apply the principle of defense in depth in a straightforward manner.

The LMP methodology uses a frequency-consequence curve that specifies acceptable dose consequence limits that get larger as the event frequency gets smaller. For event sequences with frequency less than a 5×10^{-7} per year cutoff, LMP does not specify a dose limit but requires consideration of these very low frequency sequences to assess for cliff-edge effects and to assure adequacy of defense in depth. The Kemmerer CPA uses the term "other quantified events," or OQE, to refer to such events. However, the PSAR is not clear on cliff-edge effects or what acceptance criteria are used when assessing OQE. The intent of evaluating OQEs is to determine if any low frequency events are sufficiently consequential to consider additional mitigation. To address cliff-edge effects and defense-in-depth adequacy, TerraPower stated

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they will evaluate all scenarios that meet either of two conditions: (1) the scenario has an estimated frequency of occurrence of 1.0×10^{-7} per year or greater with a 95% confidence level; or (2) any scenario, regardless of frequency, that has a consequence of greater than 1000 rem Total Effective Dose Equivalent over 30 days at the exclusion area boundary. They also stated that they had not yet encountered a scenario with a dose level high enough to warrant this further evaluation, even considering event types such as unprotected reactivity addition and unprotected loss of flow. We expect to review these conclusions at the OL stage.

We consider the applicant's implementation of the LMP methodology at this stage of licensing to be acceptable and consistent with Commission policy on risk-informed, performance-based regulation for advanced reactors.

Seismic Isolation System

One specific design feature considered to be novel is the use of a seismic isolation system (SIS) for the reactor enclosure system. The design approach is to embed in the reactor building substructure an SIS to support the reactor and enhance protection against seismic events. TerraPower is following the general lead of the GE PRISM™ design of the 1980s, which envisioned using steel/rubber isolation devices to support the reactor and provide seismic protection. Special attention will be required to address the relative deflections between SSCs supported by the seismic isolation system and those outside the system, such as connections to intermediate heat transfer system piping. Additionally, phenomena such as reactivity response to seismic forces may be relatively new for an SFR design and warrant additional attention during the OL stage.

Treatment of Safety Analysis Uncertainties and Margins

Some of the calculations required to support the safety case are much more involved and complex than for other reactor technologies, and the PSAR does not include discussion of the uncertainties or available margins. These analyses will be refined as the design proceeds from the CP stage to the OL stage. It will be important to establish the margin to figures of merits that are used to assess whether principal design criteria are met. While in the end there may be sufficient margin, the uncertainties associated with those calculations need to be established and assessed for acceptability at the OL stage. Examples that should be addressed and explicitly documented in the Final Safety Analysis Report include:

- Reactivity feedback coefficients depend on complex calculations of the deformation of the fuel rods as restrained by the core restraint system, including the effect of the lower grid plate. Moreover, point kinetics may not capture the three-dimensional (3-D) nature of the response of the core and 3-D kinetics may be necessary. The uncertainties associated with those calculations and the margin to avoiding a net positive reactivity coefficient should be quantified.
- Calculations related to confirmation of acceptable fuel integrity and safety-related structural materials, in general, depend on detailed thermomechanical analysis and the uncertainty in the predicted response (stress, strain, creep rupture, etc.) that are not discussed in the PSAR or referenced topical reports. In particular, analysis of the margin to eutectic melt, a key metric for fuel failure, is needed. These uncertainties and margins are expected to be quantified.

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SUMMARY

TerraPower has sufficiently completed the early phases of a risk-informed safety case, using the LMP approach, to justify approval of their construction permit application for a Sodium SFR at a site in Kemmerer, Wyoming. This application is supported by safety enhancements associated with the KU1 reactor design that are significant when compared to prior SFR designs. These enhancements can be leveraged to support simplifications (such as use of a functional containment approach) in design and analysis. We look forward to reviewing the final safety case when it is submitted with an application for an operating license.

Our review supports issuance of the construction permit for KU1.

We are not requesting a formal response from the staff to this letter.

Sincerely,



Signed by Kirchner, Walter
on 11/16/25

Walter L. Kirchner
Chairman

Enclosures:

1. Appendix: Safety Enhancements of the Sodium Reactor Design Relative to Prior Sodium-Cooled Fast Reactors (SFRs)
2. List of Acronyms

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2. TerraPower, LLC, "Submittal of Revisions to the Construction Permit Application for the Natrium Reactor Plant, Kemmerer Power Station Unit 1," October 3, 2025 (ADAMS Package No. [ML25276A289](#))
3. TerraPower, LLC, "Preventative Measures Classification Methodology and Preliminary Results," September 10, 2025 (ADAMS Package No. [ML25253A385](#))
4. TerraPower, LLC, "Transmittal of TerraPower, LLC, 'Natrium Demonstration DID Evaluation Report,' NAT-4770 Revision 1," July 23, 2025 (ADAMS Package No. [ML25205A086](#) (public), Accession No. [ML25205A088](#) (non-public))
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November 16, 2025

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE REAUTHORIZATION OF
POWER OPERATIONS FOR THE PALISADES NUCLEAR PLANT

Accession No: ML25311A150 Publicly Available (Y/N): Y Sensitive (Y/N): N

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APPENDIX

**SAFETY ENHANCEMENTS OF THE NATRIUM REACTOR DESIGN RELATIVE TO PRIOR
SODIUM-COOLED FAST REACTORS (SFRs)**

Purpose: This appendix enumerates some of the safety enhancements in the Natrium design as described by TerraPower. Their implementation of the Licensing Modernization Project (LMP) methodology takes credit for such safety enhancements to address the potential for a hypothetical core disruptive accident (HCDA) and the adequacy of the functional containment design strategy in the safety basis for Kemmerer Unit 1 (KU1).

Background: As a risk-informed methodology, LMP focuses on accident scenarios determined via a probabilistic risk assessment (PRA) and mechanistic source terms instead of postulated events that are deterministically judged to be bounding. Use of a risk informed approach can lead to differences in safety design and analysis compared to the use of deterministic methods. For KU1, this approach resulted in two significant changes relative to earlier SFR designs:

1. Prior SFR designs considered the potential for an HCDA, which is a hypothetical severe event in a SFR characterized by a rapid, uncontrolled increase in reactor power and subsequent rearrangement of fuel into a more reactive configuration. This could lead to a power excursion and energy release that might challenge the reactor's containment. Such accidents may be triggered by a loss of coolant flow, or a transient overpower event, especially if the reactor fails to shut down automatically (known as "unprotected" events). For KU1, TerraPower does not include an HCDA in their safety basis since it is not deemed to be credible per the LMP risk-informed process.
2. Prior SFR designs included a containment structure (or set of structures) with characteristics specified by principal design criteria that define low leakage barriers and safety classifications. For KU1, TerraPower employs a functional containment design strategy that identifies the radionuclide retention barriers and their performance characteristics shown to be needed by a mechanistic source term analysis.

Discussion: The Natrium reactor design incorporates several important features that support the LMP-based design decisions discussed above:

- A large sodium pool with a high degree of thermal inertia. The primary sodium, except for a small amount that is sent to the sodium processing system, remains in the reactor vessel. All penetrations are above the top of the primary sodium pool.
- Metal fuel, which significantly reduces the amount of stored energy available for interaction between the fuel and the coolant when compared to oxide fuels used in other SFR designs.
- No addition of plutonium in fuel, which reduces the magnitude of the void reactivity worth compared to other SFR fuel systems.
- Two independent and diverse shutdown mechanisms (gravity scram and motor-driven driveline scram follow), each controlled by independent and diverse trip systems (reactor protection system and alternative shunt trip system). Each of these shutdown mechanisms inserts two diversely designed sets of control rods.

- Two independent, decay heat removal systems. The reactor air cooling (RAC) system is passive and is always on, and the intermediate air cooling (IAC) system can operate in active and passive modes.
- Two primary mechanical sodium pumps whose design enables the transition from forced to natural circulation of sodium.
- Functional containment as supported by improved understanding of fission product release from metallic fuel and its subsequent transport in sodium and the cover gas.
- Separation of the energy island from the nuclear island, which essentially eliminates the potential for sodium-water interaction. There is potential for sodium-salt interaction at the sodium-salt heat exchanger, but TerraPower plans to prevent or mitigate this interaction as described in Chapter 13 of the preliminary safety analysis report.
- Acceptable reactor performance for many unprotected scenarios. Preliminary analyses done by TerraPower indicated that while fuel rod failure (releasing the contents of the fuel rod plenum) could be expected in severe low frequency unprotected events such as unintended withdrawal of one control rod or a loss of flow caused by loss of power to the primary sodium pumps, inherent reactivity feedback features (doppler, fuel axial expansion, grid plate radial expansion) should preclude sodium boiling or fuel melt.
- Additional preventive features. Unprotected transients complicated by failures such as unintended withdrawal of multiple control rods or locked rotor failures of both primary sodium pumps could potentially lead to fuel melt or boiling of sodium. The applicant intends to include defense-in-depth features, such as rod withdrawal interlocks and bypass flow in the sodium pumps to mitigate locked rotor conditions, to further lower the likelihood of such outcomes.
- Reduced potential for a severe accident to lead to an energetic reactivity transient. Specifically, as discussed in referenced National Laboratory documents, certain thermal and physical properties of metal fuel make it unlikely that fuel melt would lead to a reactivity addition, particularly when compared to designs that use oxide fuel.^{1, 2}

Conclusion: These features result in a robust implementation of the key safety functions in the Natrium design. They demonstrably influence plant safety by either reducing the frequency of postulated events or reducing the associated consequences. Thus, a significant reduction in the overall risk profile of the plant is expected when compared to other SFR designs. It is reasonable to expect that these safety enhancements can be leveraged in an operating license application to fully justify the TerraPower conclusions that an HCDA is not credible and that a functional containment strategy driven by a robust mechanistic source term analysis is technically sound.

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LIST OF ACRONYMS

10 CFR	Title 10 of the <i>Code of Federal Regulations</i>
3-D	Three Dimensional
AEA	Atomic Energy Act
CPA	Construction Permit Application
DBA	Design Basis Accident
HCDA	Hypothetical Core Disruptive Accident
IAEA	International Atomic Energy Agency
IAC	Intermediate Air Cooling
KU1	Kemmerer Power Station, Unit 1
LBE	Licensing Basis Events
LMP	Licensing Modernization Project
MW _e	Megawatts Electric
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OL	Operating License
OQE	Other Quantified Events
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
RAC	Reactor Air Cooling
RG	Regulatory Guide
RPS	Reactor Protection System
RTBs	Reactor Trip Breakers
SE	Safety Evaluation
SFR	Sodium Fast Reactor
SIS	Seismic Isolation System
SSCs	Structures, Systems and Components
USO	US SFR Owner, LLC

Enclosure 2

APPENDIX D – PRINCIPAL CONTRIBUTORS

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