June 30, 2023

TP-LIC-LET-0087
Project Number 99902100

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

Subject: In-Vessel DBA without Release Methodologies Presentation Material

This letter provides the TerraPower, LLC presentation material for the upcoming Natrium™ advanced reactor[1] pre-application engagement meeting "In-Vessel DBA without Release Methodologies" (Enclosures 2 and 4).

The presentation material contains proprietary information and as such, it is requested that Enclosure 4 be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." An affidavit certifying the basis for the request to withhold Enclosure 4 from public disclosure is included as Enclosure 1. Proprietary materials have been redacted from the presentation provided in Enclosure 3; redacted information is identified using [[ ]][(a)(4)].

This letter and enclosures make no new or revised regulatory commitments.

If you have any questions regarding this submittal, please contact Ryan Sprengel at rsprengel@terrapower.com or (425) 324-2888.

---

[1] a TerraPower and GE-Hitachi technology.

15800 Northup Way, Bellevue, WA 98008 www.TerraPower.com P. +1 (425) 324-2888 F. +1 (425) 324-2889

Sincerely,

Ryan Sprengel
Director of Licensing, Natrium
TerraPower, LLC

Enclosure:
1. TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))
2. "In-Vessel DBA without Release Methodologies" Presentation Material – Open Meeting – Non-Proprietary (Public)
3. "In-Vessel DBA without Release Methodologies" Presentation Material – Closed Meeting – Non-Proprietary (Public)
4. "In-Vessel DBA without Release Methodologies" Presentation Material – Closed Meeting – Proprietary (Non- Public)

cc: Mallecia Sutton, NRC
William Jessup, NRC
Andrew Proffitt, NRC
Nathan Howard, DOE
Jeff Ciocco, DOE

ENCLOSURE 1

TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))

Enclosure 1
TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure
(10 CFR 2.390(a)(4))

I, George Wilson, hereby state:

1. I am the Vice President, Regulatory Affairs and I have been authorized by TerraPower, LLC (TerraPower) to review information sought to be withheld from public disclosure in connection with the development, testing, licensing, and deployment of the Natrium™ reactor and its associated fuel, structures, systems, and components, and to apply for its withholding from public disclosure on behalf of TerraPower.
2. The information sought to be withheld, in its entirety, is contained in Enclosure 4, which accompanies this Affidavit.
3. I am making this request for withholding, and executing this Affidavit as required by 10 CFR 2.390(b)(1).
4. I have personal knowledge of the criteria and procedures utilized by TerraPower in designating information as a trade secret, privileged, or as confidential commercial or financial information that would be protected from public disclosure under 10 CFR 2.390(a)(4).
5. The information contained in Enclosure 4 accompanying this Affidavit contains non-public details of the TerraPower regulatory and developmental strategies intended to support NRC staff review.
6. Pursuant to 10 CFR 2.390(b)(4), the following is furnished for consideration by the Commission in determining whether the information in Enclosure 4 should be withheld:
    a. The information has been held in confidence by TerraPower.
    b. The information is of a type customarily held in confidence by TerraPower and not customarily disclosed to the public. TerraPower has a rational basis for determining the types of information that it customarily holds in confidence and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application and substance of that system constitute TerraPower policy and provide the rational basis required.
    c. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR 2.390, it is received in confidence by the Commission.
    d. This information is not available in public sources.
    e. TerraPower asserts that public disclosure of this non-public information is likely to cause substantial harm to the competitive position of TerraPower, because it would enhance the ability of competitors to provide similar products and services by reducing their expenditure of resources using similar project methods, equipment, testing approach, contractors, or licensing approaches.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on: June 30, 2023

George Wilson

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 2

"In-Vessel DBA without Release Methodologies"
Presentation Material – Open Meeting

Non-Proprietary (Public)

In-Vessel DBA without Release Methodologies

a TerraPower & GE-Hitachi technology

TP-LIC-PRSNT-0008

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Agenda

- Natrium™ Reactor Overview
- Previous Meetings and Upcoming Submittals
- DBA Selection
- In-Vessel DBA EMDAP
- Closed Session Topics

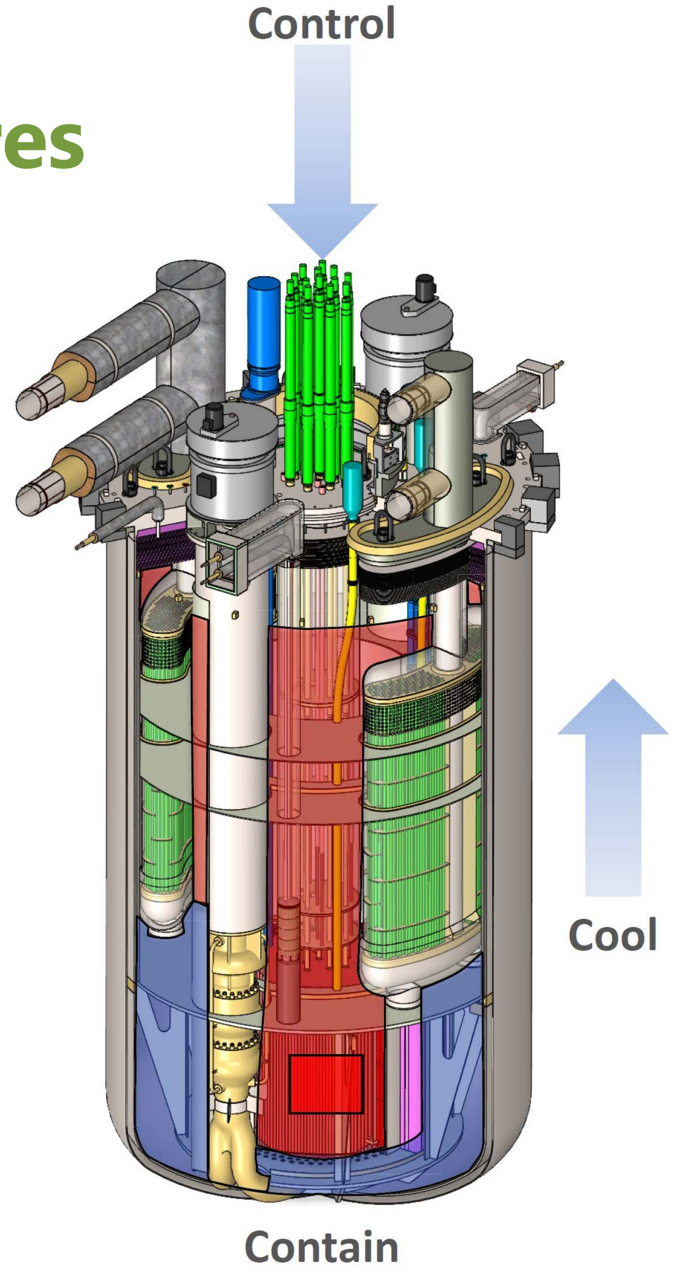
Natrium Reactor Overview

- The Natrium project is demonstrating the ability to design, license, construct, startup and operate a Natrium reactor.

- Pre-application interactions are intended to reduce regulatory uncertainty and facilitate the NRC's understanding of the Natrium design and its safety case.

Natrium Safety Features

- Pool-type Metal Fuel SFR with Molten Salt Energy Island
  - Metallic fuel and sodium have high compatibility
  - No sodium-water reaction in steam generator
  - Large thermal inertia enables simplified response to abnormal events
- Simplified Response to Abnormal Events
  - Reliable reactor shutdown
  - Transition to coolant natural circulation
  - Indefinite passive emergency decay heat removal
  - Low pressure functional containment
  - No reliance on Energy Island for safety functions
- No Safety-Related Operator Actions or AC power
- Technology Based on U.S. SFR Experience
  - EBR-I, EBR-II, FFTF, TREAT
  - SFR inherent safety characteristics demonstrated through testing in EBR-II and FFTF



Control

- Motor-driven control rod runback and scram follow
- Gravity-driven control rod scram
- Inherently stable with increased power or temperature

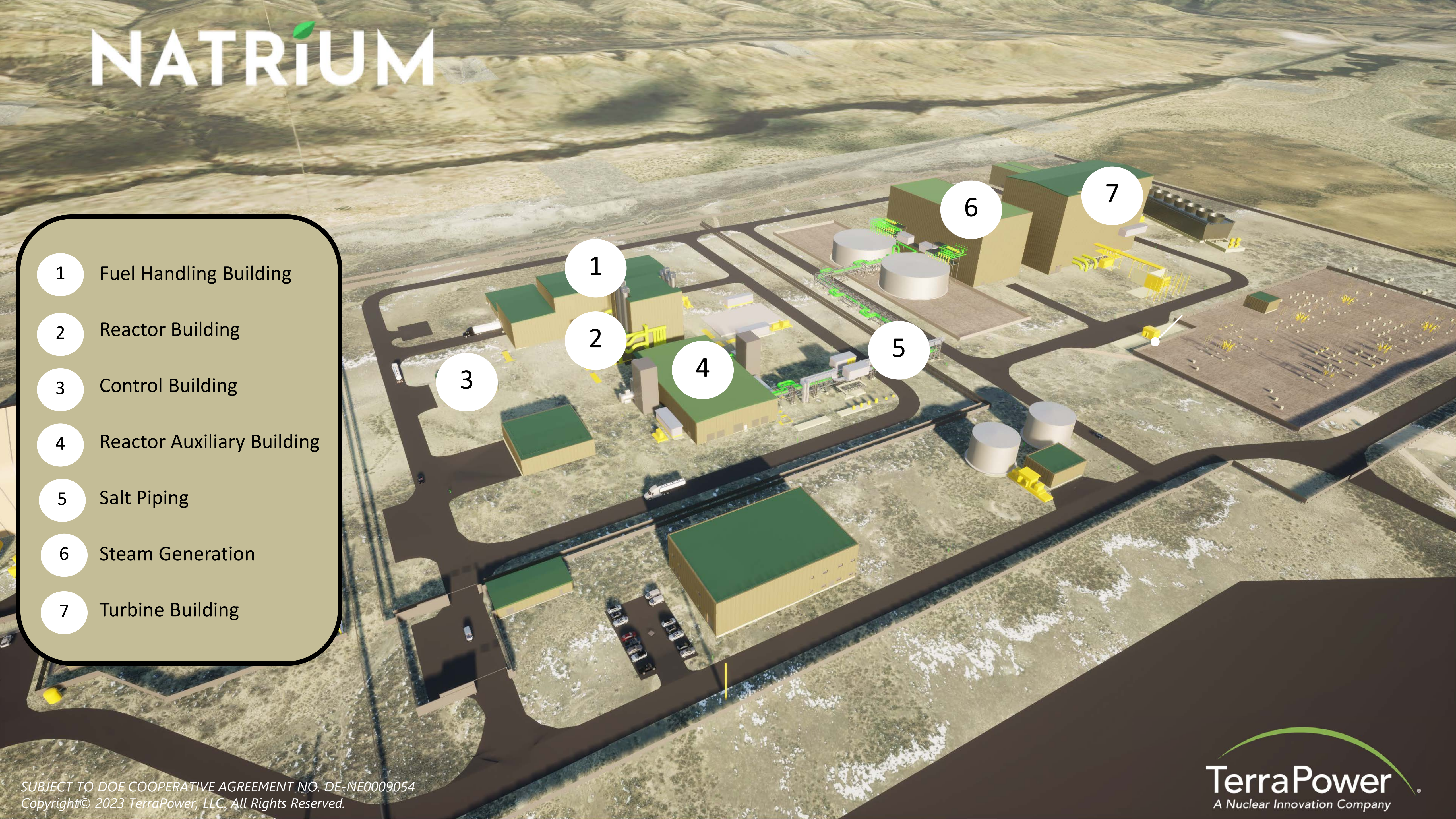
Cool

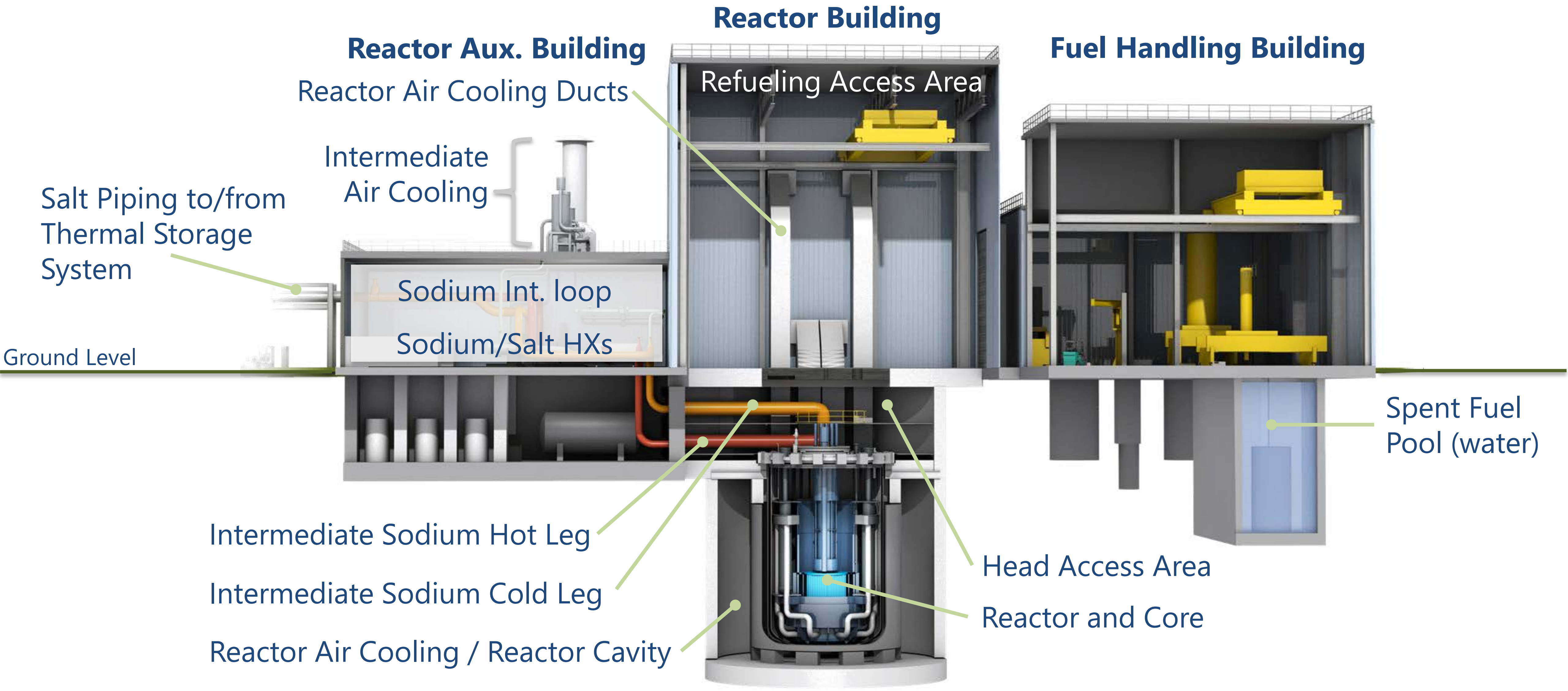
- In-vessel primary sodium heat transport (limited penetrations)
- Intermediate air cooling natural draft flow
- Reactor air cooling natural draft flow – always on

Contain

- Low primary and secondary pressure
- Sodium affinity for radionuclides
- Multiple radionuclides retention boundaries

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054


NATRIUM
1 Fuel Handling Building
2 Reactor Building
3 Control Building
4 Reactor Auxiliary Building
5 Salt Piping
6 Steam Generation
7 Turbine Building
1
2
3
4
5
6
7
TerraPower
A Nuclear Innovation Company



Reactor Building
Reactor Aux. Building
Fuel Handling Building
Refueling Access Area
Reactor Air Cooling Ducts
Intermediate Air Cooling
Salt Piping to/from Thermal Storage System
Sodium Int. loop
Sodium/Salt HXs
Ground Level
Spent Fuel Pool (water)
Intermediate Sodium Hot Leg
Intermediate Sodium Cold Leg
Reactor Air Cooling / Reactor Cavity
Head Access Area
Reactor and Core

Previous Meetings and Upcoming Submittals

- Previous meetings
  - ML23031A098 - 03/01/2023, In-Vessel Events without Release Methodology for the Natrium Design
  - ML23109A339 - 04/25/2023, SAS4A/SASSYS-1 (SAS) as Evaluation Model for In-Vessel DBA and LBEs without Radiological Release
  - ML23142A293 - 06/06/2023, Thermal-Hydraulic Methods for the Natrium Design
- Upcoming Submittals
  - In-Vessel DBA without Release Methodology Report
  - Construction Permit Application

DBA Selection

- DBAs are derived from the list of DBEs with the prescriptive assumption that only SR SSCs are available
- The required safety functions are defined
  - If a DBA does not meet event consequences within the 10 CFR 50.34 dose limits, new SR SSCs are selected, and their required safety functions are defined
- Once the DBA is identified, the dose release is compared to the 10 CFR 50.34 dose limits to ensure they are not exceeded. For the events in consideration in this presentation a release is precluded so the dose limits are met
- This process is completed for all DBEs and may result in the combination of multiple DBEs into a single DBA if the safety related systems credited result in the same event progression

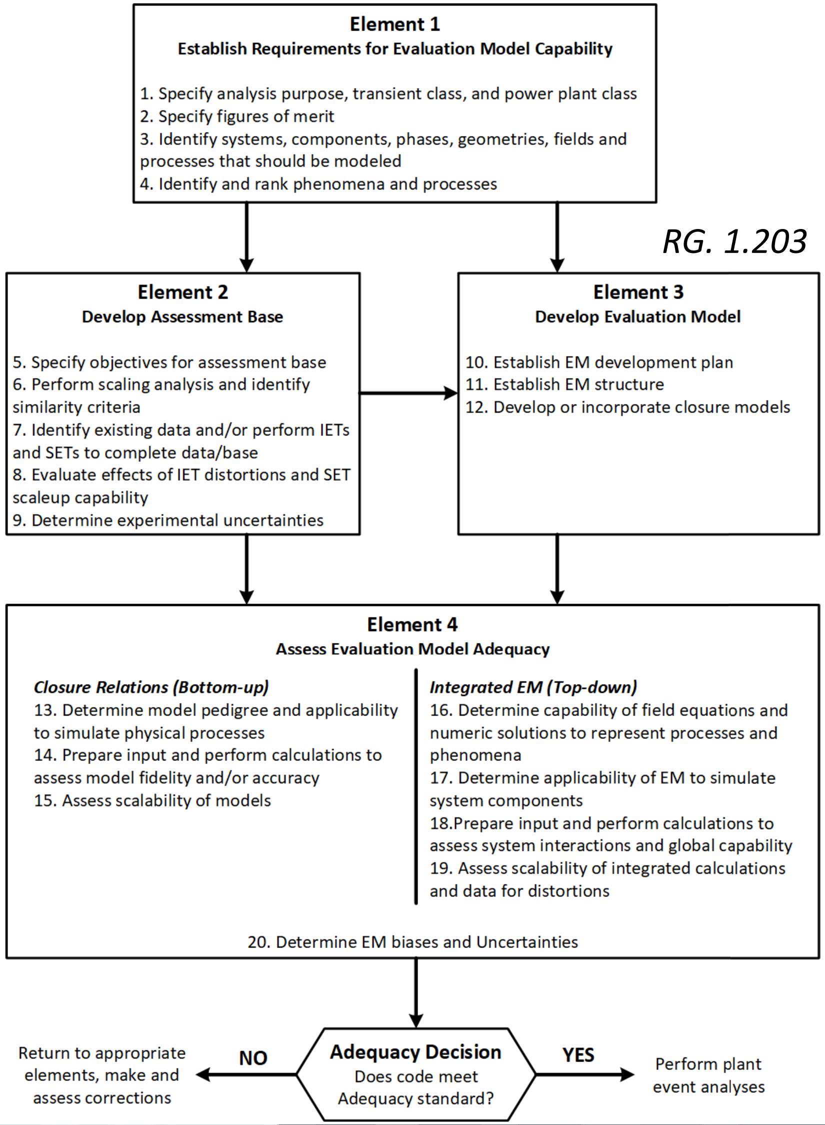
DBA Selection

In DBA analyses

- Any **SR SSCs** not failed as part of the parent DBE(s)
- **Continued existence of other passive SSCs** not failed as part of the parent DBE
  - Passive SSCs are not assumed to experience a gross failure but cannot change state
- **Examples of Design Basis Events that have associated DBAs**
  - Loss Of Offsite Power with failure of IAC, Rod Withdrawal At Power, etc.

In-Vessel DBA without Release EM Development

- The In-Vessel DBA w/o Radiological Release Methodology is intended to comply with EMDAP in RG 1.203
- EM development is similar to that presented in ML23031A098 - 03/01/2023, “Methodology for In-vessel Events without Release for the Natrium Design”



Element 1 - Evaluation Model Capability Requirements

- Step 1: Analysis Purpose

  To demonstrate that the Natrium Reactor has enough safety margin for In-vessel Design Basis Accidents without Radiological Release and will meet construction permit and operating license expectations

Element 1
Establish Requirements for Evaluation Model Capability

1. Specify analysis purpose, transient class, and power plant class
2. Specify figures of merit
3. Identify systems, components, phases, geometries, fields and processes that should be modeled
4. Identify and rank phenomena and processes

- Step 2: Figures of Merit (PIRT)
  - Cladding temperature history, Thermal creep strain
  - Fuel temperature
  - Coolant temperature
- Steps 3 and 4:
  - Consistent with previous pre-engagement discussion

Element 2 – Assessment Base Development

- Step 5: Objectives for assessment base
  - To present experimental data to demonstrate the EM capabilities of predicting Natrium reactor responses to postulated DBAs without fuel failure
- Steps 6, 7, 8, and 9:
  - Consistent with previous pre-engagement discussion, additional tests and data are under consideration

Element 2
Develop Assessment Base

5. Specify objectives for assessment base
6. Perform scaling analysis and identify similarity criteria
7. Identify existing data and/or perform IETs and SETs to complete data/base
8. Evaluate effects of IET distortions and SET scaleup capability
9. Determine experimental uncertainties

Element 3 – Evaluation Model Development

- Steps 10, 11, 12:
  - Consistent with previous pre-engagement discussion, closure models selected for DBA

Element 3

Develop Evaluation Model

10. Establish EM development plan
11. Establish EM structure
12. Develop or incorporate closure models

Element 4 – Evaluation Model Adequacy

- Step 13: Model pedigree and applicability
  - In accordance with RG 1.203
- Step 14 & Step 18: Input preparation
  - Safety-related SSCs
  - Input values developed with conservative approach and accumulative uncertainties of input parameters
- Steps 15, 16, 17, 19, & 20
  - Numerical techniques, scheme and options are under evaluation
  - Will be conducted in accordance with RG 1.203

Element 4
Assess Evaluation Model Adequacy

Closure Relations (Bottom-up)
13. Determine model pedigree and applicability to simulate physical processes
14. Prepare input and perform calculations to assess model fidelity and/or accuracy
15. Assess scalability of models

Integrated EM (Top-down)
16. Determine capability of field equations and numeric solutions to represent processes and phenomena
17. Determine applicability of EM to simulate system components
18. Prepare input and perform calculations to assess system interactions and global capability
19. Assess scalability of integrated calculations and data for distortions
20. Determine EM biases and uncertainties

EM Summary

- EM development principles and regulatory positions are followed
- EMDAP Element 1
  - Evaluation model capability requirements developed
  - PIRTs developed, future revision planned to reflect final Natrium design
- EMDAP Element 2
  - Legacy test data identified
  - TerraPower tests identified to fill the gap between phenomena ranked “High” in the PIRTs and legacy test data
  - Assessment matrix for validation developed
  - Scaling analyses
    - EBR-II
    - TerraPower Test Facility

EM Summary

- EMDAP Element 3
  - Primary safety analysis computer code selected for in-vessel DBAs without release
    - SAS4A/SASSYS-1
  - EM structure developed
  - EM development plan established
  - Three new closure models implemented following the required SQA procedure
- EMDAP Element 4
  - Evaluate model adequacy (top-down and bottom-up) planned
  - Benchmark analyses
    - EBR-II, FFTF, Phenix
    - New tests (IET and SETs) are planned

Additional Closed Session Topics

Preliminary Overview of In-Vessel DBA without Release:

- Plant Model
- Reactor Core Model
- Reactor Protection System Model
- UQ/Conservative Modeling

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Acronym List

AOO - Anticipated Operational Occurrence
ANL - Argonne National Laboratory
BDBE - Beyond Design Basis Event
CGD - Commercial Grade Dedication
DBA - Design Basis Accident
DBE - Design Basis Event
DID - Defense in Depth
EBR - Experimental Breeder Reactor
EM - Evaluation Model
EMDAP - Evaluation Model Development and Application Process
FFTF - Fast Flux Test Facility
IAC - Intermediate Air Cooling
IET - Integral Effect Test
IFR - Integral Fast Reactor
IHT - Intermediate Heat Transport System
IHX - Intermediate Heat Exchanger
LBE - Licensing Basis Event
NQA - ASME Nuclear Quality Assurance
PHT - Primary Heat Transport System
PIRT - Phenomena Identification and Ranking Table
PRA - Probabilistic Risk Assessment
PRISM - Power Reactor Innovative Small Module
RAC - Reactor Air Cooling System
RCC - Reatcor Core and Core Components
RES - Reactor Enclosure System
SET - Separate Effect Test
SFR - Sodium-Cooled Fast Reactor
SHRT - Shutdown Heat Removal Tests
SQA - Software Quality Assurance
SSC - Structures, Systems, and Components
TREAT - Transient Reactor Test Facility
UQ – Uncertainty Quantification
VTR - Versatile Test Reactor

ENCLOSURE 3

“In-Vessel DBA without Release Methodologies” Presentation Material – Closed Meeting

Non-Proprietary (Public)

TerraPower

NATRIUM

In-Vessel DBA without Release Methodologies

a TerraPower & GE-Hitachi technology

TP-LIC-PRSNT-0009

Portions of this presentation are considered proprietary and TerraPower, LLC requests it be withheld from public disclosure under the provisions of 10 CFR 2.390(a)(4).

Nonproprietary versions of this presentation indicate the redaction of such information using [[ ]]$^{(a)(4)}$.

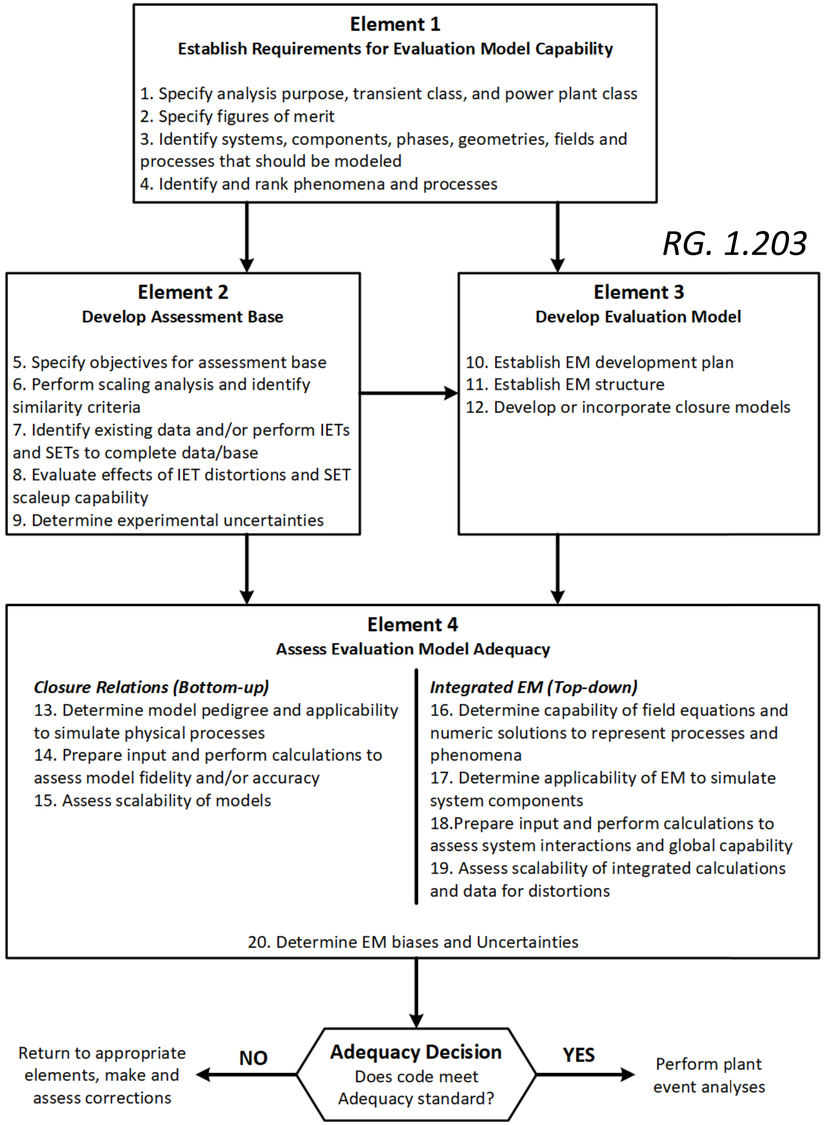
Agenda

- Additional Detail on EMDAP Updates and Differences
- Preliminary Overview of Natrium™ Input Model Preparation for In-Vessel DBA without Release:
  - Plant Model
  - Reactor Core Model
  - Decay Heat Model
  - SCRAM Model
  - Reactor Protection System and Engineered Safety Features Model
- Uncertainty Quantification / Conservative Modeling Approach

In-Vessel DBA without Release EM Development

- The In-Vessel DBA w/o Radiological Release Methodology is intended to comply with EMDAP in RG 1.203
- EM development is similar to that presented in ML23031A098 on 03/01/2023, “Methodology for In-Vessel Events without Release for the Natrium Design”

[[



]](a)(4)

Element 2 – Assessment Base Development – DBAs

- Step 6: Scaling analysis and similarity criteria

[[

Element 2
Develop Assessment Base

5. Specify objectives for assessment base
6. Perform scaling analysis and identify similarity criteria
7. Identify existing data and/or perform IETs and SETs to complete data/base
8. Evaluate effects of IET distortions and SET scaleup capability
9. Determine experimental uncertainties

]](a)(4)

Element 2 – Assessment Base Development – DBAs

[[

]](a)(4)

Element 2 – Assessment Base Development – DBAs

- Step 7: Legacy test data
  - Investigation of the legacy data driven by the PIRT
  - Potentially obtainable legacy test data

[[

]](a)(4)

Element 2 – Assessment Base Development – DBAs

- Step 7: TerraPower test needs (cont.)

[[

]](a)(4)

Element 2 – Assessment Base Development – DBAs

- Step 7: TerraPower test needs (cont.)

[[

]](a)(4)

Element 3 – Evaluation Model Development – DBAs

- Step 10: EM development plan
  - Selected code
    - **SAS4A/SASSYS-1**

[[

]](a)(4)

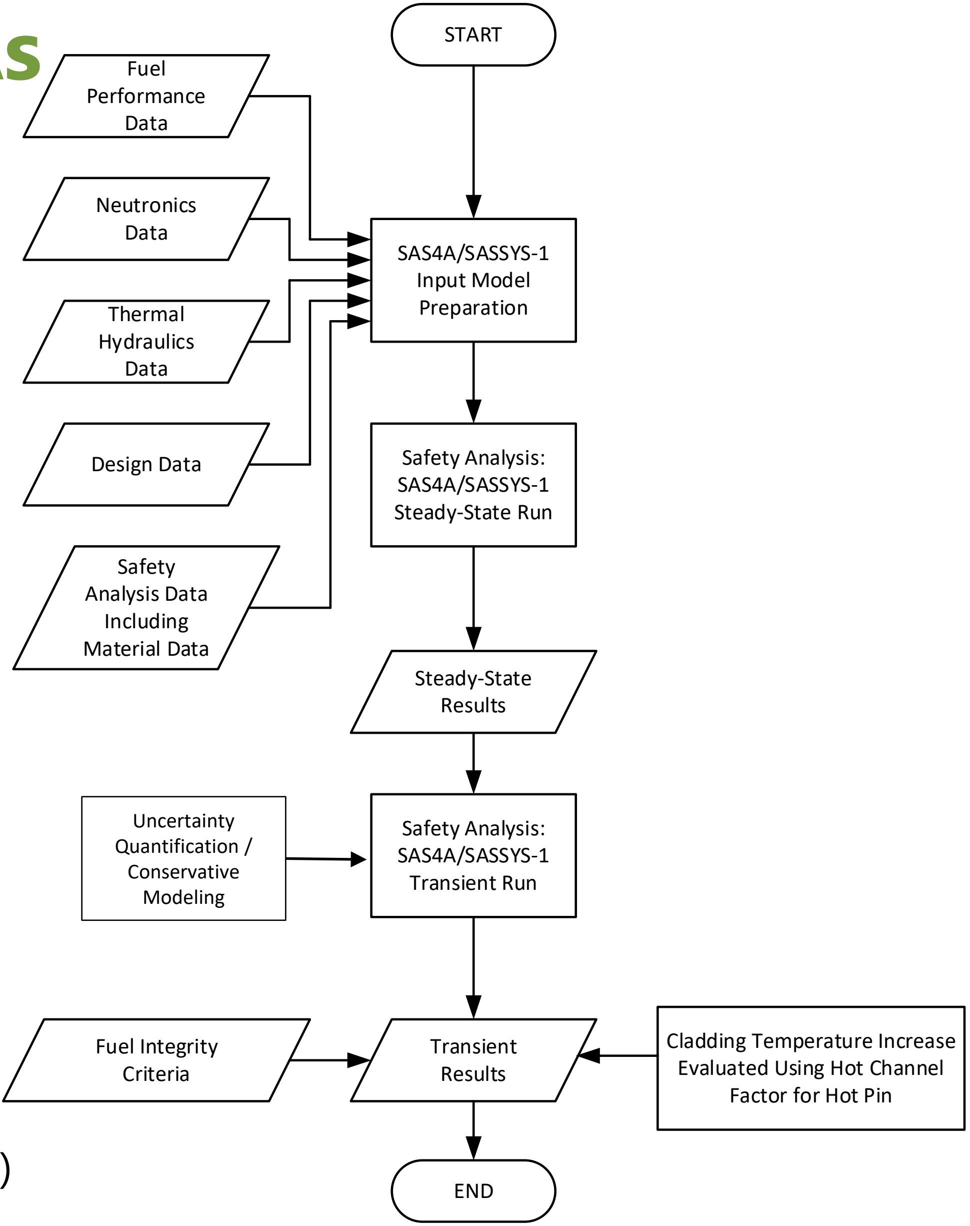
Element 3
Develop Evaluation Model

10. Establish EM development plan
11. Establish EM structure
12. Develop or incorporate closure models

Element 3 – EM Development – DBAs

- Step 11: EM Structure
- Step 12: Closure Models

[[



]](a)(4)

Element 4 – Evaluation Model Adequacy

- Step 13: Model pedigree and applicability
  - In accordance with RG 1.203
- Step 14 & Step 18: Input preparation
  - Safety-related SSCs
  - Input values
    - **Conservative approach with accumulative uncertainties of input parameters**
  - **Natrium input model will be discussed in later slides**
- Steps 15, 16, 17, 19, & 20
  - Numerical techniques, scheme and options are under evaluation
  - Will be conducted in accordance with RG 1.203

Element 4
Assess Evaluation Model Adequacy

Closure Relations (Bottom-up)
13. Determine model pedigree and applicability to simulate physical processes
14. Prepare input and perform calculations to assess model fidelity and/or accuracy
15. Assess scalability of models

Integrated EM (Top-down)
16. Determine capability of field equations and numeric solutions to represent processes and phenomena
17. Determine applicability of EM to simulate system components
18. Prepare input and perform calculations to assess system interactions and global capability
19. Assess scalability of integrated calculations and data for distortions
20. Determine EM biases and uncertainties

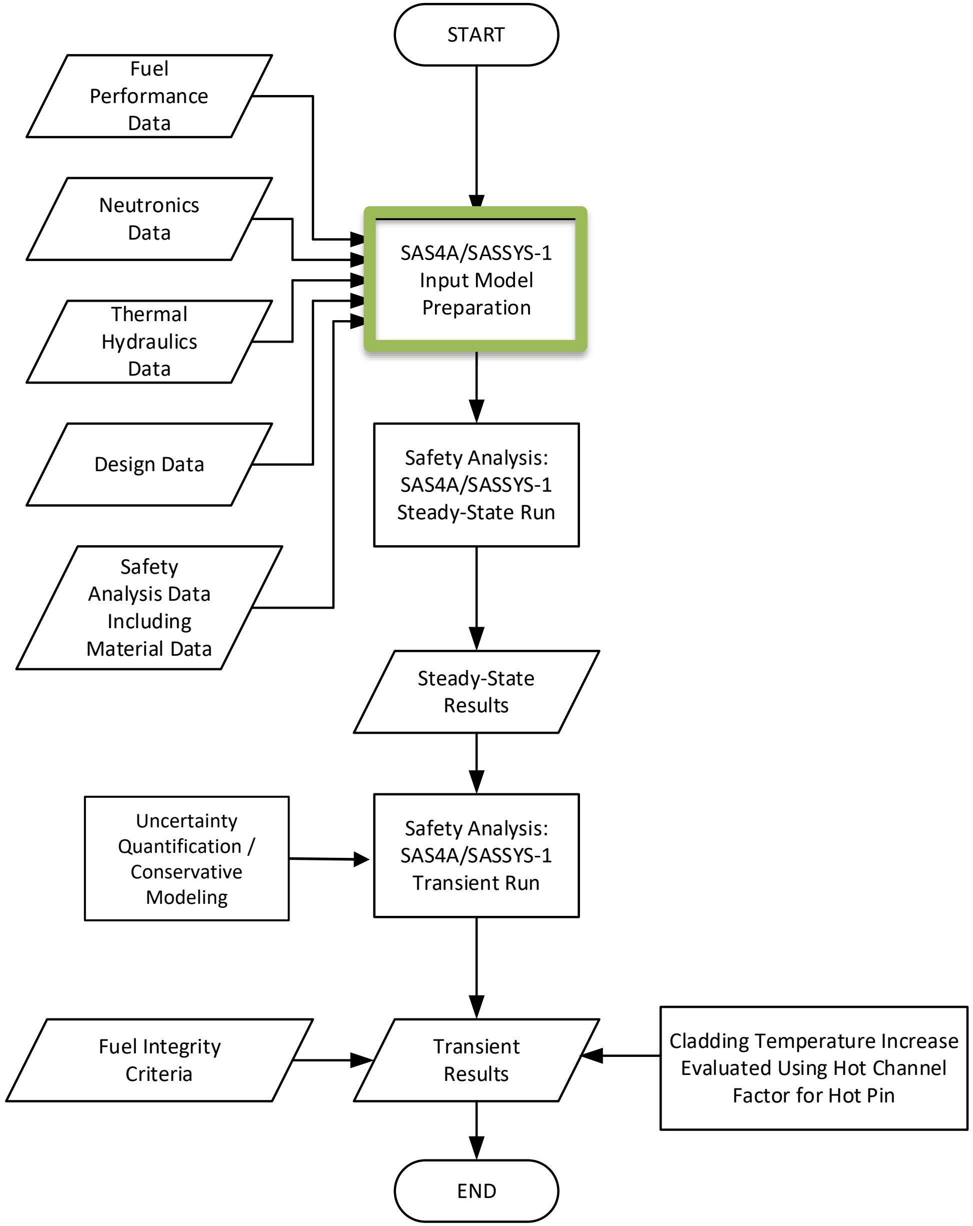
EM Summary

- EM development principles and regulatory positions are followed
- EMDAP Element 1
  - Evaluation model capability requirements developed
  - PIRTs developed, future revision planned to reflect final Natrium design
- EMDAP Element 2
  - Legacy test data identified
  - TerraPower tests identified to fill the gap between phenomena ranked "High" in the PIRTs and legacy test data
  - Assessment matrix for validation developed
  - Scaling analyses
    - EBR-II
    - TerraPower Test Facility

EM Summary

- EMDAP Element 3
  - Primary safety analysis computer code selected for in-vessel DBAs without release
    - SAS4A/SASSYS-1
  - EM structure developed
  - EM development plan established
  - Three new closure models implemented following the required SQA procedure
- EMDAP Element 4
  - Evaluate the model adequacy (top-down and bottom-up) planned
  - Benchmark analyses
    - EBR-II, FFTF, Phenix
    - New tests (IET and SETs) are planned

Input Model Preparation for In-Vessel DBA without Release Analysis



DBA Plant Model

Topics

- Plant Model Overview
  - Primary Heat Transport System
  - Intermediate Heat Transport System
  - Reactor Air Cooling System
- Future Development Areas

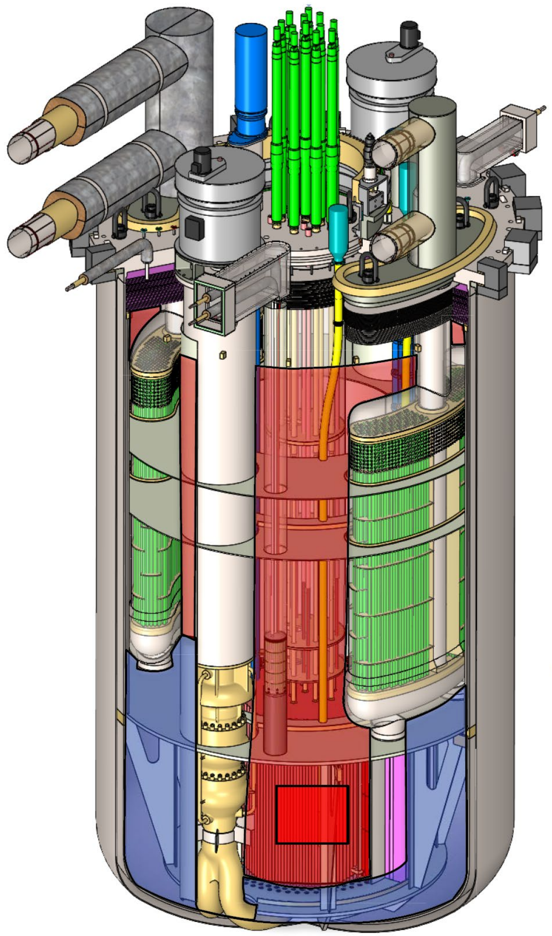
Plant Model – System Wide

[[

]](a)(4)

Plant Model – Primary Heat Transport System

[[



[[

]](a)(4)

]](a)(4)

Plant Model – Intermediate Heat Transport System

[[

]](a)(4)

Plant Model – Reactor Air Cooling System

[[

]](a)(4)

Future Development Areas

[[

]](a)(4)

DBA Reactor Core Model

Topics

- Channelization Strategy
- Axial Nodalization
- Radial Nodalization
- Geometry Calculations
- Reactivity Feedback
- Assembly Material Properties
- Convection Heat Transfer Parameters
- Friction Factor Correlations
- Reynolds-Dependent Core Inlet/Outlet Loss Coefficients
- Core Statepoints
- Decay Heat Model
- SCRAM Model
- Future Development Areas

Core Model – Channelization Strategy - Natrium Core Map Description

[[

]](a)(4)

Core Model – Channelization Strategy

[[

]](a)(4)

Core Model – Channelization Strategy - Results

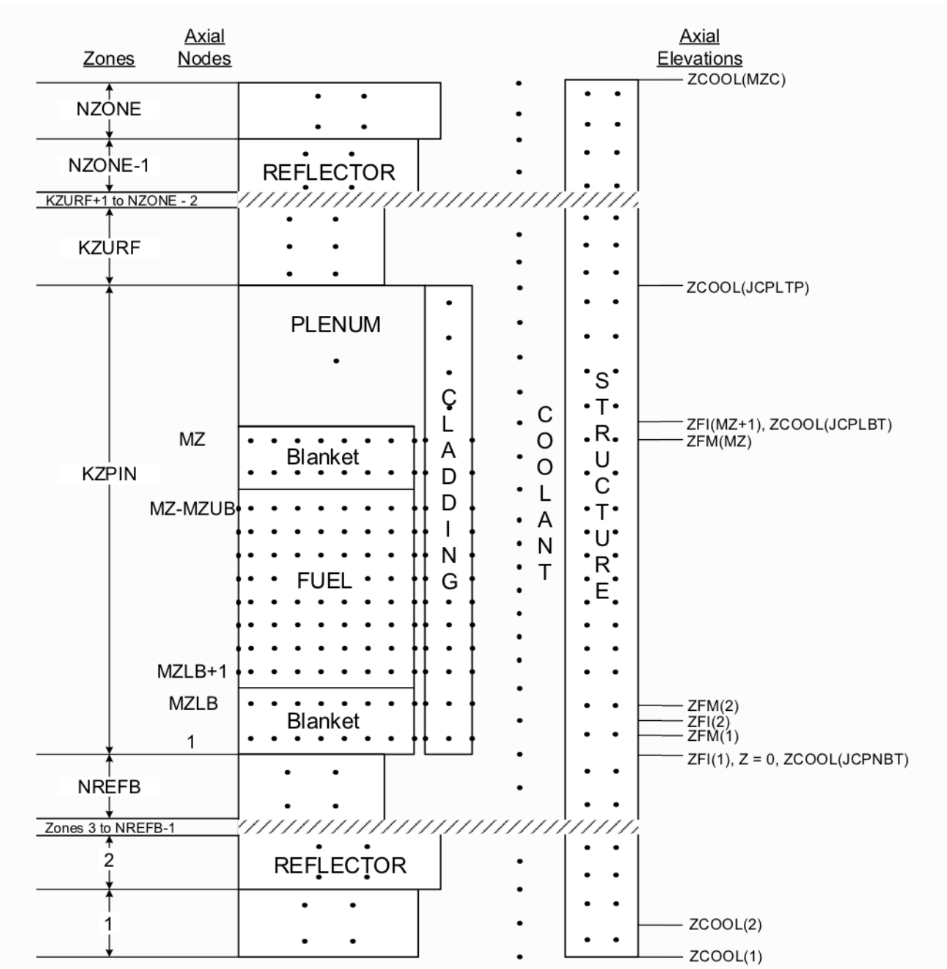
[[

]](a)(4)

Core Model – Axial Nodalization

[[

]](a)(4)



Source: SAS4A/SASSYS-1 Documentation 5.6 (6296), Figure 3.2.3

Core Model – Axial Nodalization

[[

]](a)(4)

Core Model – Axial Nodalization

[[

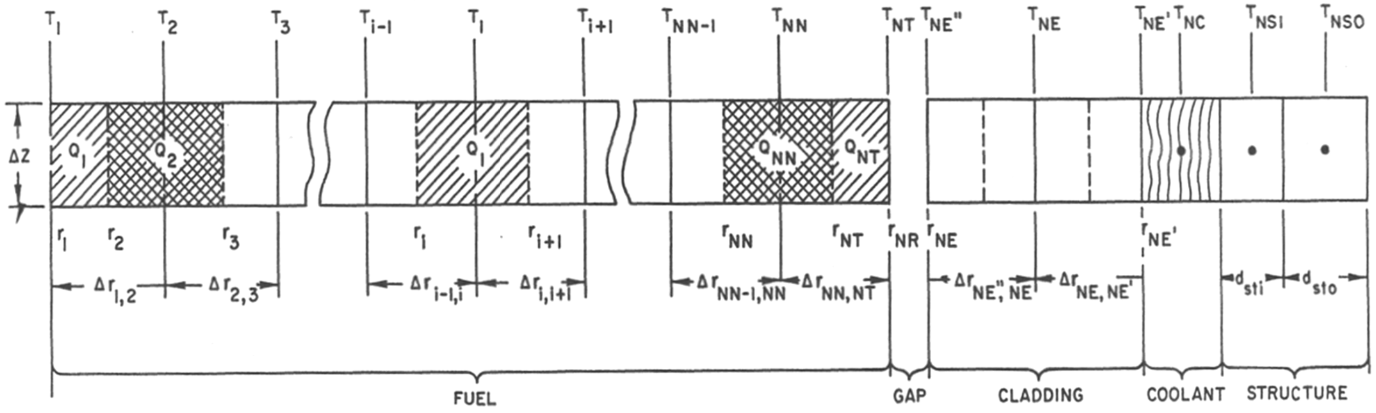
]](a)(4)

TerraPower NATRIUM

Core Model – Radial Nodalization

[[

]](a)(4)

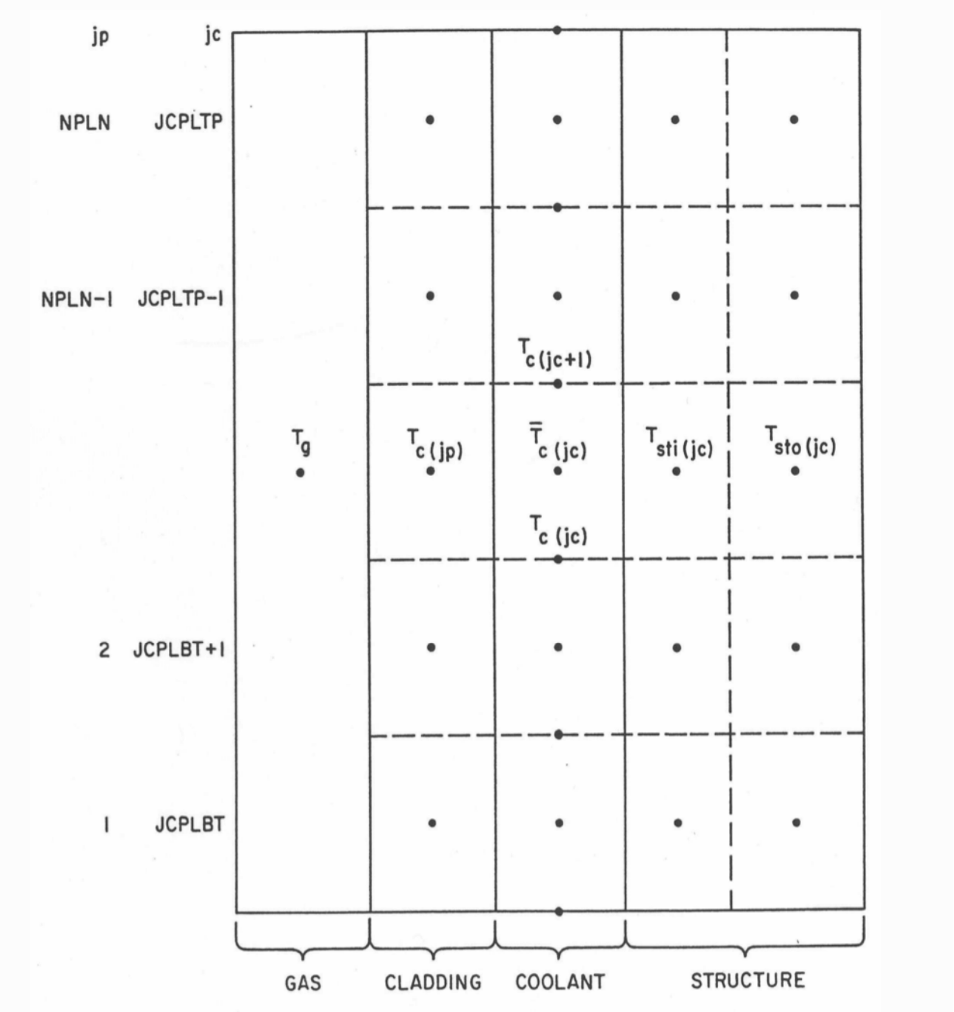


Source: SAS4A/SASSYS-1 Documentation 5.6 (6296), Figure 3.2.4

Core Model – Radial Nodalization

Fission Gas Plenum:

- Per the SAS code conventions,
  - A single temperature and pressure describe the fission gas plenum
  - A single radial node is used in each cladding segment defined in the axial mesh

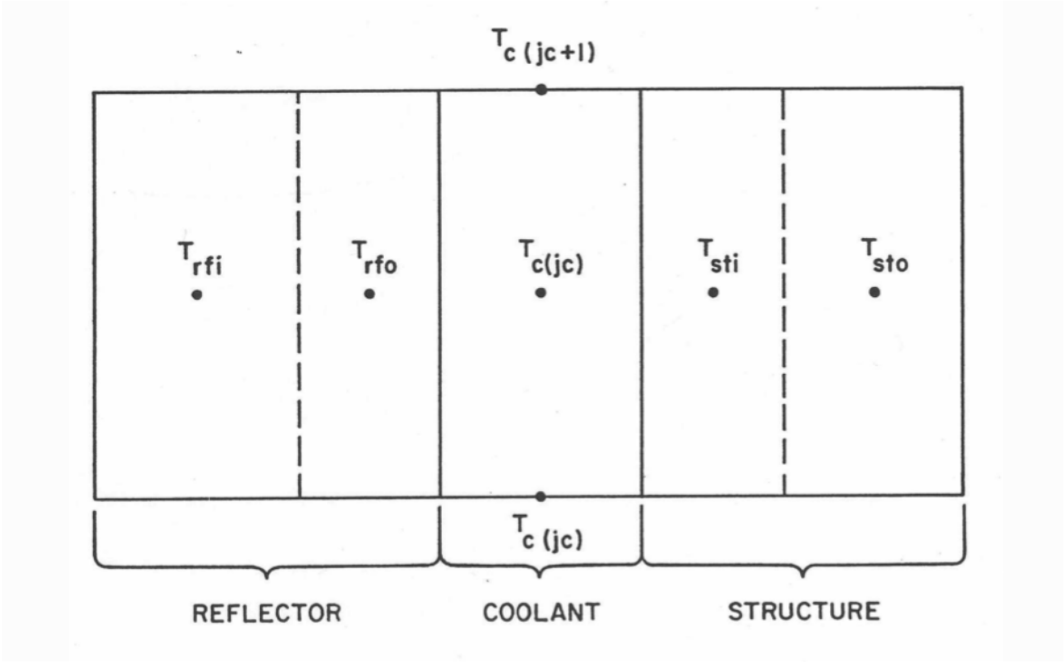


Source: SAS4A/SASSYS-1 Documentation 5.6 (6296), Figure 3.2.5

Core Model – Radial Nodalization

Upper and Lower Reflectors:

- Per the SAS code conventions,
  - Upper and lower reflectors (or shields) are modeled using a two-node slab geometry
  - Thickness of the reflector is defined for each node
- Reflector properties are defined for:
  - Density x Heat Capacity
  - Thermal Conductivity



Source: SAS4A/SASSYS-1 Documentation 5.6 (6296), Figure 3.2.6

Core Model – Radial Nodalization

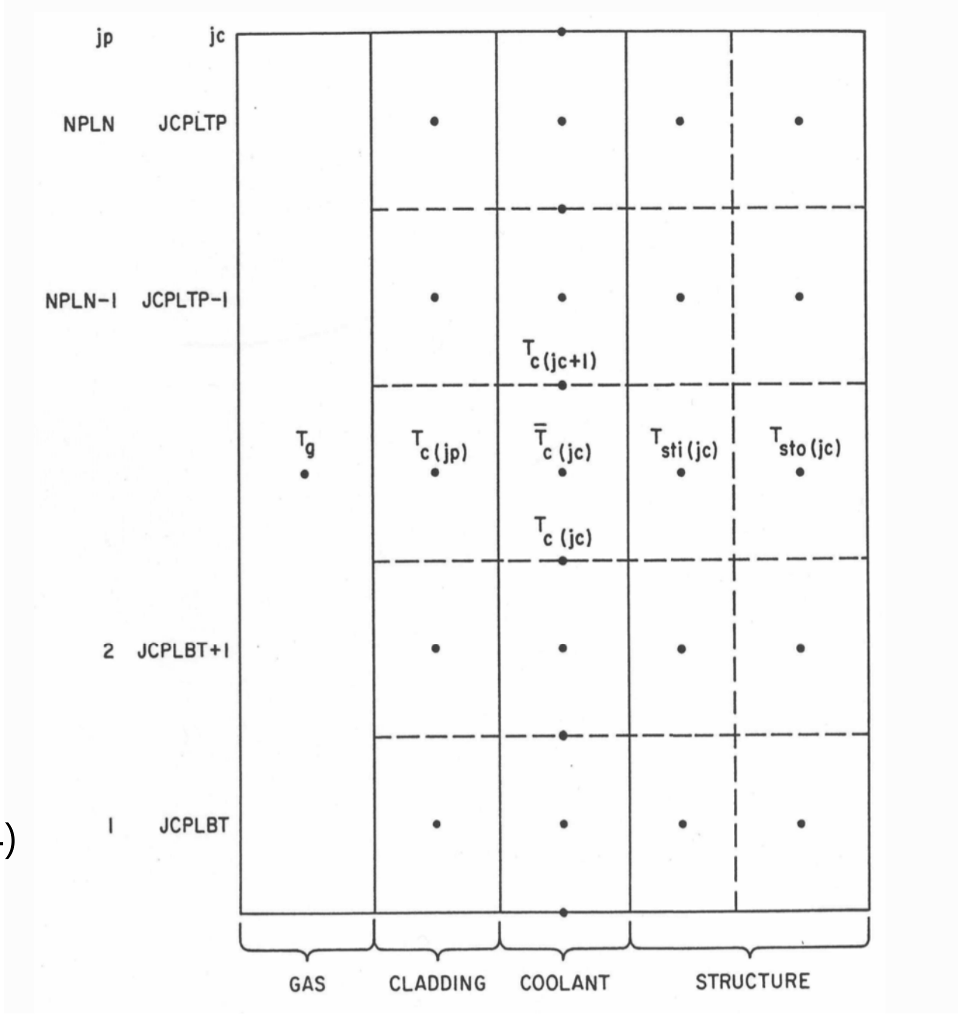
Structure:

- Per the SAS code conventions,
  - Structure region is used to account for materials not modeled explicitly by the fuel pin or reflector regions (e.g., assembly duct wall)
  - Structure is modeled as a two-node slab geometry
- Structure Thickness:

[[

]](a)(4)

- Structure properties are defined for:
  - Density x Heat Capacity
  - Thermal Conductivity



Source: SAS4A/SASSYS-1 Documentation 5.6 (6296), Figure 3.2.5

TerraPower NATRIUM

Core Model – Geometry Calculations

- Geometric dimensions are calculated using data provided from a database containing design data provided by upstream groups
- During initialization of the upstream database, all assemblies are axially expanded to T_{hot} conditions

Core Model – Reactivity Feedback

- The point kinetics calculation of fission power is influenced by reactivity feedback within the SAS model
- Reactivity Feedback Sources:
  - Core Radial Expansion
  - Fuel, Cladding, and Structure Axial Expansion
  - Fuel Doppler
  - Coolant Density
  - Control Rod Driveline Expansion
  - Fuel and Cladding Relocation

Core Model - Assembly material properties

[[

]](a)(4)

Core Model – Convection Heat Transfer Parameters

- The built-in SAS convection heat transfer coefficient equation coefficients (C1, C2, and C3) support the Lyon-Martinelli correlation.

$$H_c = \frac{K_c \cdot (C1 \cdot (R_e \cdot P_r)^{C2} + C3)}{D_h}$$

where:

K_c = coolant thermal conductivity
R_e = Reynolds number
P_r = Prandtl number
D_h = channel hydraulic diameter

$$N_u = C1 \cdot (P_e)^{C2} + C3$$

where:

N_u = Nusselt number
P_e = Peclet number = Reynolds number · Prandtl number

[[

]](a)(4)

Core Model – Friction Factor Correlations

[[

]](a)(4)

Core Model – Reynolds-Dependent Core Inlet/Outlet Loss Coefficients

[[

]](a)(4)

Core Model – Statepoints

- Distinct core models are generated to represent:
  - Specific cycles and times in life:
    - Beginning of Life (Beginning of initial plant cycle, all fuel assemblies at 0 EFPD)
    - Beginning of Equilibrium Cycle [[ ]]$^{(a)(4)}$
    - End of Equilibrium Cycle [[ ]]$^{(a)(4)}$
  - An array of core power levels and flowrates from Hot Full Power / Full Flow to Low Power / Low Flow conditions

Decay Heat Model

SAS decay heat models for BOL, BOEC, and EOEC:

[[

]](a)(4)

SCRAM Model

SAS SCRAM model for BOL, BOEC, and EOEC:

[[

]](a)(4)

Future Development Areas

[[

]](a)(4)

Reactor Protection System and Engineered Safety Features Model

Topics

- Reactor Protection System Trips
- Engineered Safety Features
- Post-SCRAM Functions
- Future Development Areas

Reactor Trips

The following functions will trigger a reactor SCRAM:

[[

]](a)(4)

Reactor Trips (continued)

The following functions will trigger a reactor SCRAM:

[[

]](a)(4)

Engineered Safety Features

[[

]](a)(4)

Post-SCRAM Functions

[[

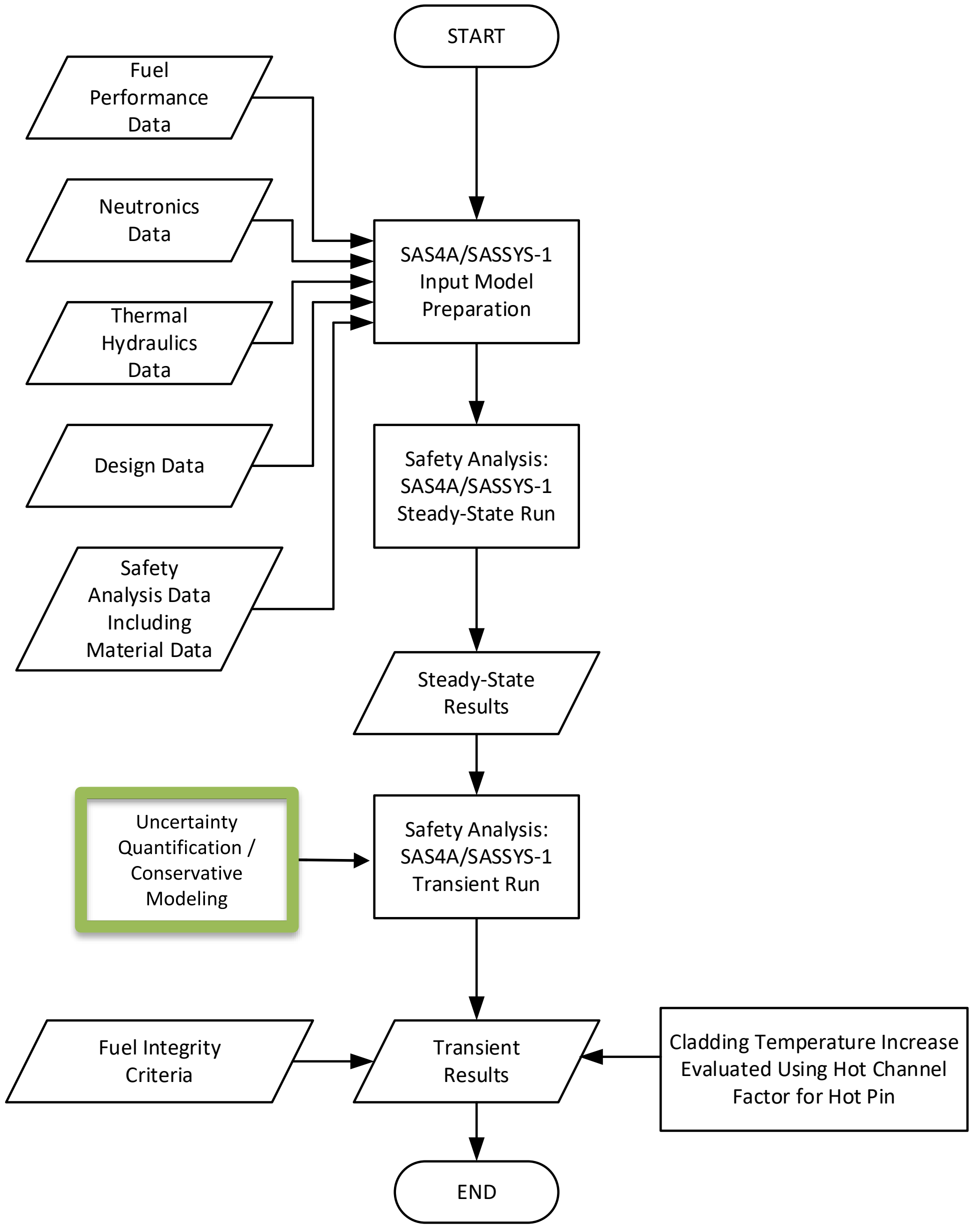
- Functions are **Non-Safety Related**
- Perform sensitivities for DBA analyses
  - Component maintains its current state
  - Component operates as designed
  - Component trips at the beginning of the transient
- Evaluate limiting scenario in DBA analyses

]](a)(4)

Future Development Areas

- Ongoing analysis sensitivity studies and iteration with the I&C design team to be performed to support determination of RPS setpoints
- Additional sensitivities performed to ensure conservative treatment of all systems

Uncertainty Quantification and Conservative Modeling



Topics

- UQ / Conservative and BEPU strategy
- Biases and Uncertainty Applications
- Safety HCF/HPR Application

UQ / Conservative and BEPU Strategies

[[

]](a)(4)

SAS Direct Biases/Uncertainties:

[[

]](a)(4)

Safety HCF/HPR

Safety Hot Channel Factor

[[

]](a)(4)

Safety HCF/HPR

Hot Pin Ratio

[[

]](a)(4)

Future Development Areas

- Improve bias, uncertainty and range accuracy
- Improve Safety HCF and HPR definition and application for transients

Review

- Overview of Input Model Preparation for In-Vessel DBA without Release:
  - Plant Model
  - Reactor Core Model
  - Decay Heat Model
  - SCRAM Model
  - Reactor Protection System and Engineered Safety Features Model
- Uncertainty Quantification / Conservative Modeling Approach

Questions?

Acronym List

AHX - Sodium-Air Heat Exchanger
AOO - Anticipated Operational Occurrence
ANL - Argonne National Laboratory
BDBE - Beyond Design Basis Event
BEPU - Best-Estimate Plus Uncertainty
BOEC - Beginning of Equilibrium Cycle
CRD - Control Rod Drive
CRDL - Control Rod Drive Line
CV - Compressible Volumes
DBA - Design Basis Accident
DBE - Design Basis Event
EBR - Experimental Breeder Reactor
EFPD - Effective Full Power Days
EM - Evaluation Model
EMDAP - Evaluation Model Development and Application Process
FFTF - Fast Flux Test Facility
GV - Guard Vessel
HCF - Hot Channel Factor
HPR - Hot Pin Ratio
HT - Heat Transport
IAC - Intermediate Air Cooling
IET - Integral Effect Test
IFR - Integral Fast Reactor
IHT - Intermediate Heat Transport System
IHX - Intermediate Heat Exchanger
ISP - Intermediate Sodium Pump
IVS - In-Vessel Storage
LBE - Licensing Basis Event
NQA - ASME Nuclear Quality Assurance
NSTF - Natural Convection Shutdown Heat Removal Test Facility
PCT - Peak Cladding Temperature
PHT - Primary Heat Transport System
PIRT - Phenomena Identification and Ranking Table
PLOF - Potected Loss of Flow
PRA - Probabilistic Risk Assessment
PRISM - Power Reactor Innovative Small Module
PSP - Primary Sodium Pump
RAC - Reactor Air Cooling System
RCC - Reactor Core and Core Components
RES - Reactor Enclosure System
SET - Separate Effect Test
SFR - Sodium-Cooled Fast Reactor
SHRT - Shutdown Heat Removal Tests
SHX - Sodium-Salt Heat Exchanger
SQA - Software Quality Assurance
SSC - Structures, Systems, and Components
TREAT - Transient Reactor Test Facility
TSTF - Thermal Stratification Test Facility
UIUC - University of Illinois at Urbana-Champaign
ULOF - Unprotected Loss of Flow
UQ - Uncertainty Quantification
VTR - Versatile Test Reactor