TerraPower

May 25, 2023

TP-LIC-LET-0080
Project Number 99902100

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

Subject: Core Thermal Hydraulic Methods Presentation Material

This letter provides the TerraPower, LLC presentation material for the upcoming Natrium™ advanced reactor[1] pre-application engagement meeting "Core Thermal Hydraulic Methods" (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. Enclosure 4 also contains ECI which can be disclosed to Foreign Nationals only in accordance with the requirements of 15 CFR 730 and 10 CFR 810, as applicable. Proprietary and ECI materials have been redacted from the presentation provided in Enclosure 3; redacted information is identified using [[ ]]$^{(a)(4)}$, [[ ]]^{ECI}, or [[ ]]$^{(a)(4), ECI}$.

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.

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. "Core Thermal Hydraulic Methods" Presentation Material – Open Meeting – Non-Proprietary (Public)
3. "Core Thermal Hydraulic Methods" Presentation Material – Closed Meeting – Non-Proprietary (Public)
4. "Core Thermal Hydraulic Methods" 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: May 25, 2023

George Wilson

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 2

“Core Thermal Hydraulic Methods”
Presentation Material – Open Meeting

Non-Proprietary (Public)

TerraPower

NATRIUM

Core Thermal Hydraulic Methods

a TerraPower & GE-Hitachi technology

NAT-4652

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Objectives

- Provide an Overview of the Natrium^{TM} Reactor
- Introduce Core Thermal Hydraulics Group
- Review Codes Selected and Developed To-Date
- Share Validation Plans for Codes and Methods

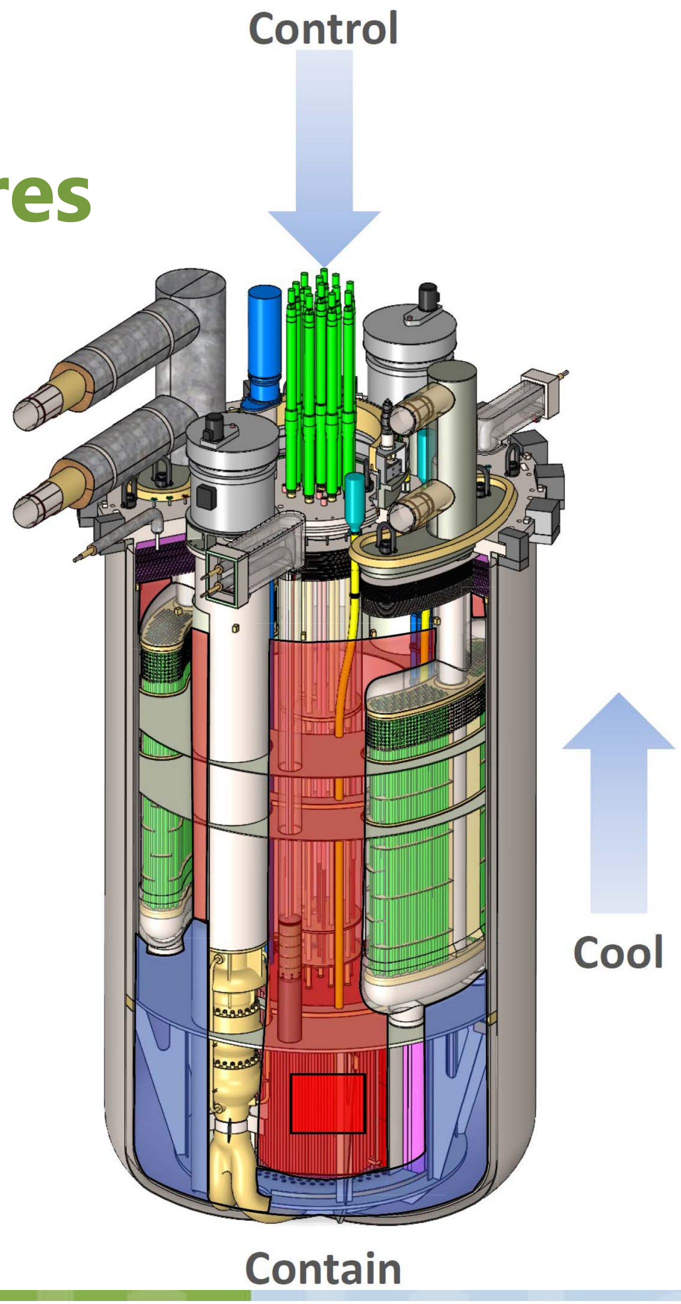
Natrium Reactor Overview

- 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

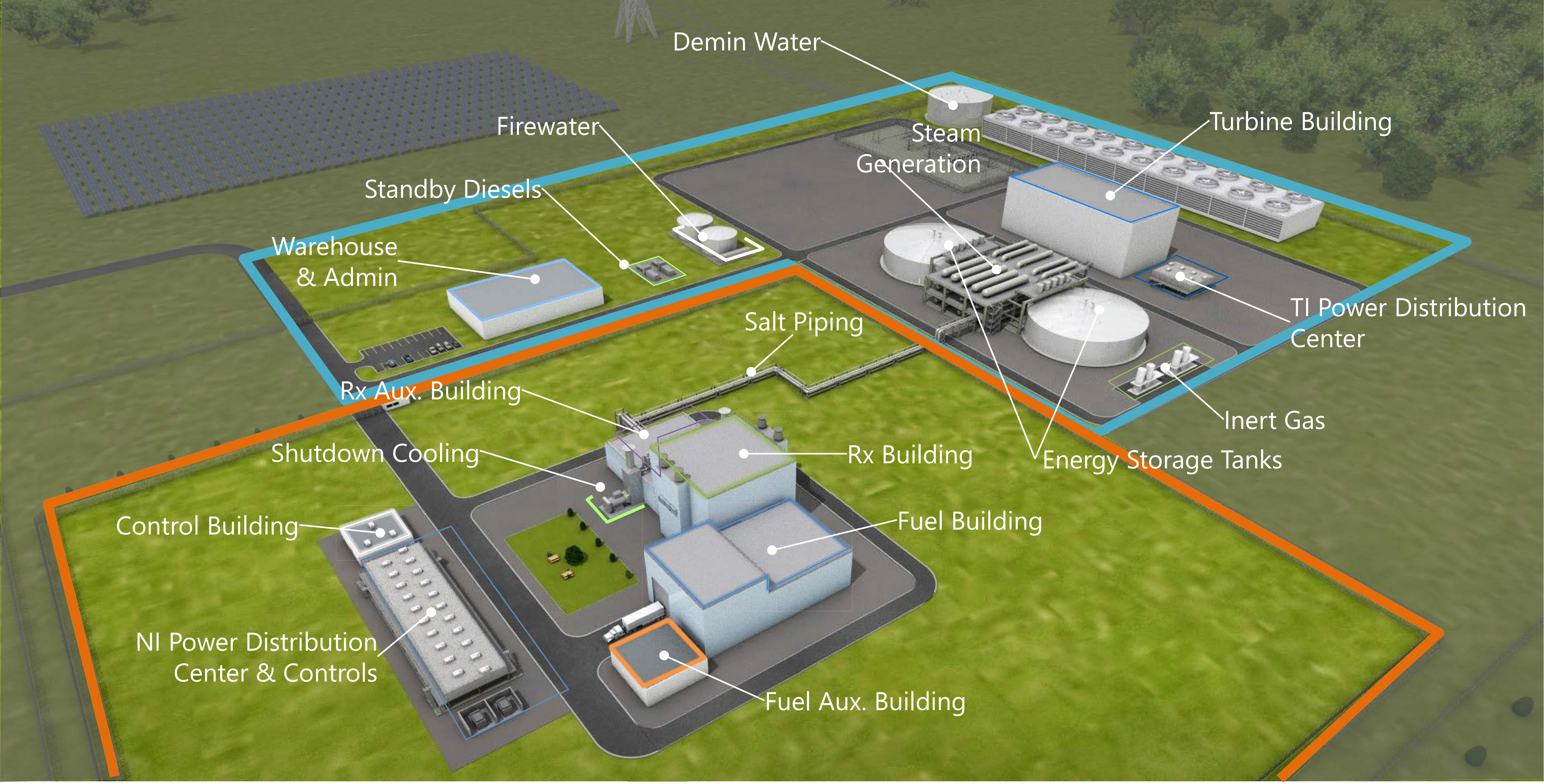
- Motor-driven control rod runback and scram follow
- Gravity-driven control rod scram
- Inherently stable with increased power or temperature

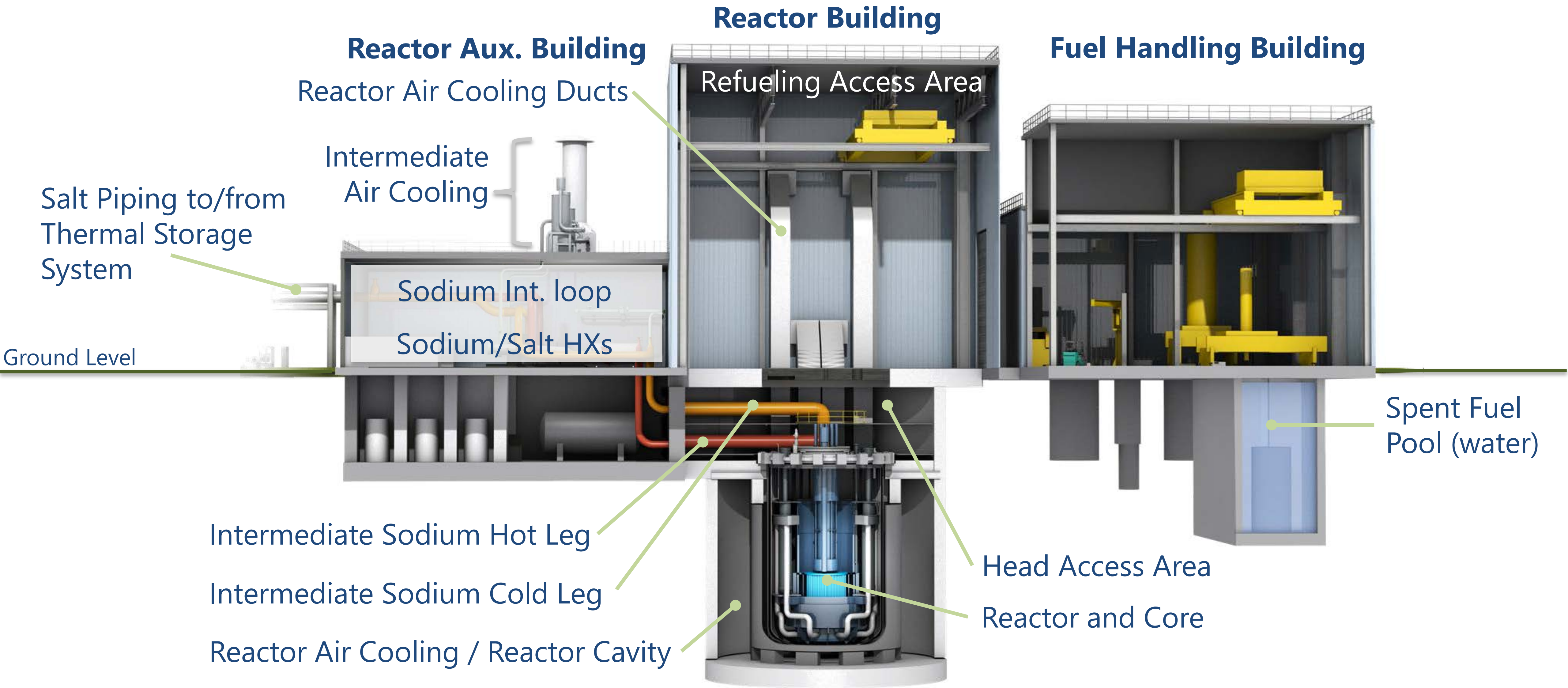
Cool

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

Contain

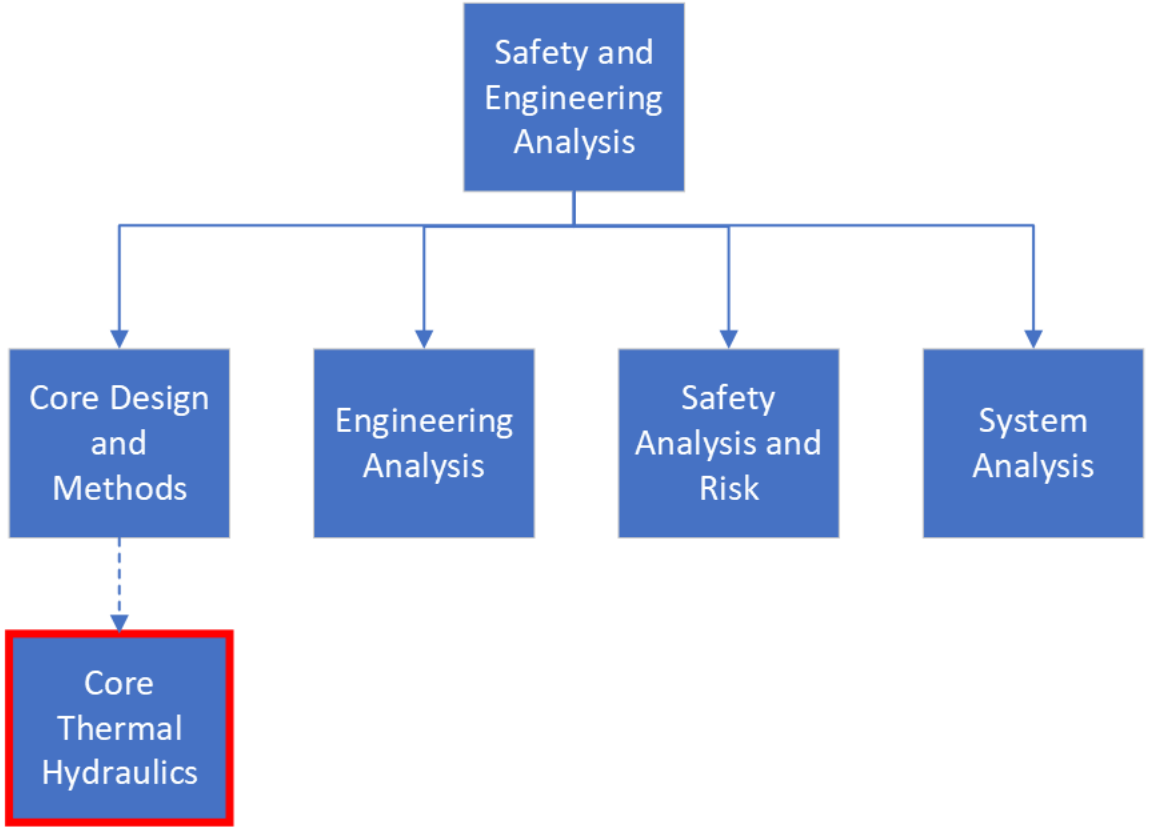
- Low primary and secondary pressure
- Sodium affinity for radionuclides
- Multiple radionuclides retention boundaries


Demin Water
Firewater
Steam Generation
Turbine Building
Standby Diesels
Warehouse & Admin
Salt Piping
TI Power Distribution Center
Rx Aux. Building
Inert Gas
Shutdown Cooling
Rx Building
Energy Storage Tanks
Control Building
Fuel Building
NI Power Distribution Center & Controls
Fuel Aux. Building

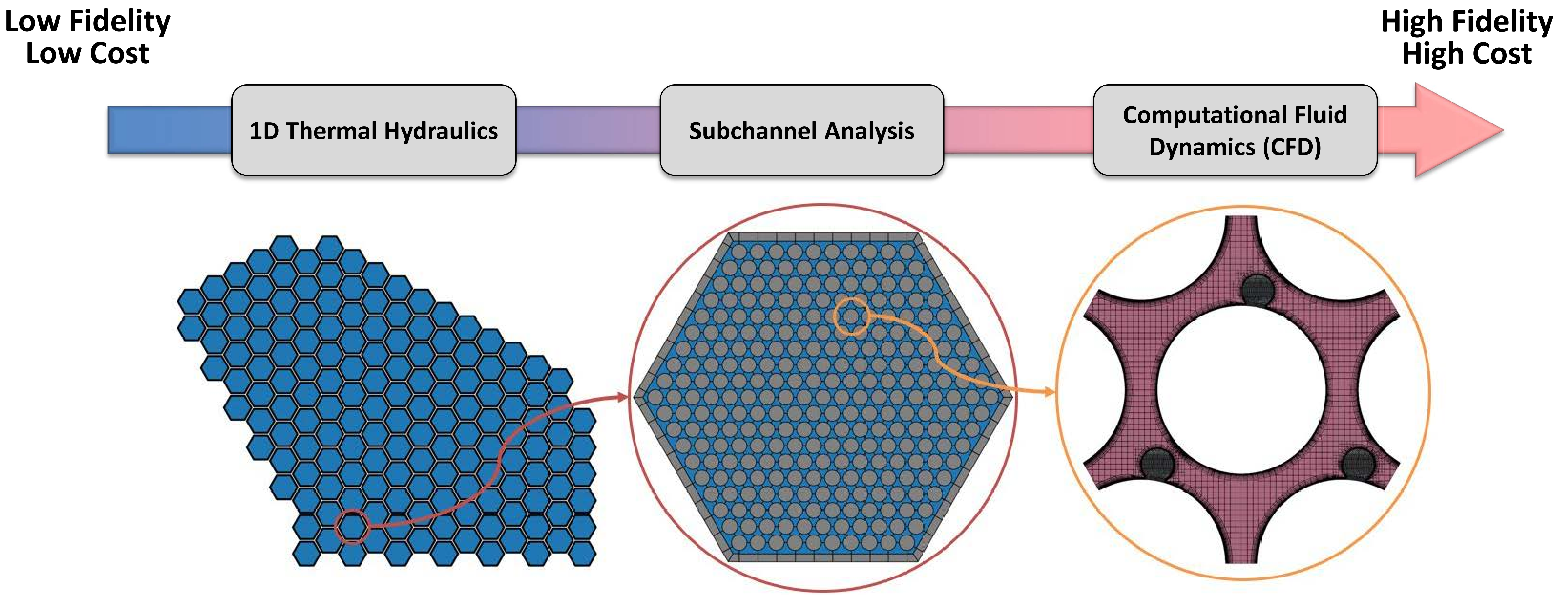

Reactor Building
Reactor Aux. Building
Fuel Handling Building
Reactor Air Cooling Ducts
Refueling Access Area
Intermediate
Air Cooling
Salt Piping to/from
Thermal Storage
System
Sodium Int. loop
Sodium/Salt HXs
Ground Level
Spent Fuel
Pool (water)
Intermediate Sodium Hot Leg
Intermediate Sodium Cold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

Introduction

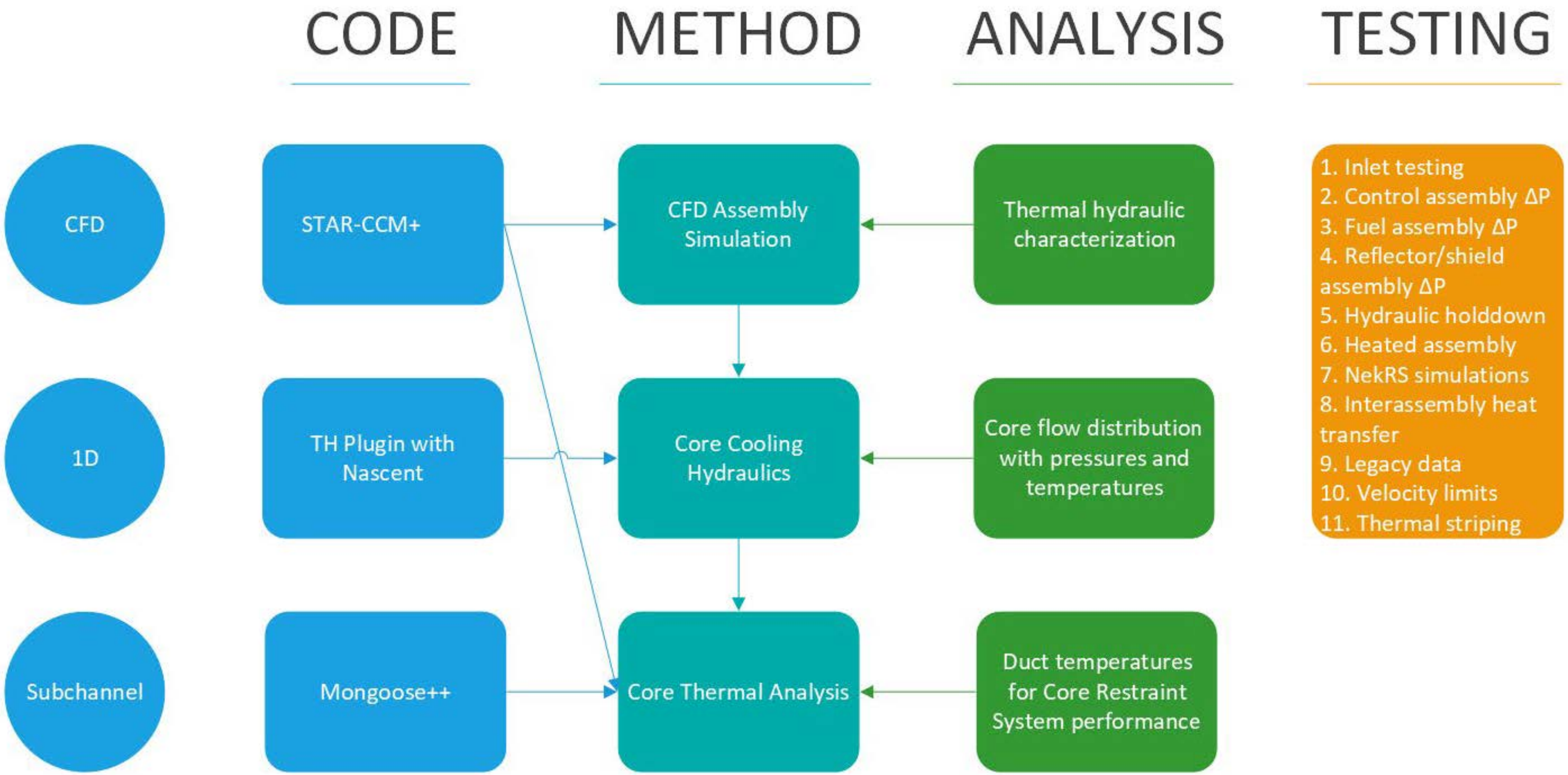
- Core Thermal Hydraulics group
  - Support core design at steady state conditions
- Interfaces with other groups include
  - Safety Analysis and Risk
  - Equipment Qualification and Testing
  - Fuel Development and Qualification
- Key analyses
  - Design orificing to achieve optimal core flow distribution
  - Enforce peak clad temperature limits
  - Manage and apply uncertainties through Hot Channel Factors
  - Generates core wide temperature distribution in support of core restraint system performance



Thermal Hydraulics Analysis Tools



Overview of Core Thermal Hydraulics



Methodology Development and Assessment

There are four fundamental elements associated with the methodology development and assessment (MD&A) process:

1. Establish the purpose and associated requirements for the methodology and/or evaluation model
2. Develop the assessment base
3. Develop the evaluation model
4. Assess the evaluation model adequacy

MD&A Element 1

- Identification and specification of the analysis purpose
  - Analysis purpose is specified with sufficient detail that an appropriate methodology and/or evaluation model can be identified
  - The purpose may envelope multiple analyses provided the requirements, dominant phenomena, etc. are common between the various analyses
- Identification of the requirements and figure(s) of merit (FOM)
  - The process of specifying the analysis purpose, requirements, and FOMs may necessitate generation of a Phenomena Importance Ranking Table

	Core Cooling Hydraulics	Core Thermal Analysis	CFD Assembly Simulation
High-level purpose	Manage operating limit/margin based on fuel temperature limits	Support core restraint system design	Thermal hydraulic characterization of an assembly
Figure(s) of merit	• Peak clad temperature • Assembly average outlet temperature • Assembly pressure drop • Core heat balance	• Duct temperatures	• Assembly pressure drop • Subchannel temperature distribution

Licensing Plan

The following pre-CPA licensing activities for the core thermal hydraulics group are planned:

- Nascent white paper submitted (March 2023)
- Core Design Methodology report (TBD)
  - Support the thermal hydraulic aspects of the core design
- Additional engagements related to safety analysis are planned
  - Partial Flow blockage

Closed Session Overview

- Natrium Core Thermal Hydraulic Design Summary
- Core Cooling Hydraulics Methodology
  - Thermal Hydraulic Plugin
  - Assembly Hydraulic Characterization Plan
  - Peak Clad Temperature with Nascent Model
  - Hot Channel Factors
- CFD Assembly Simulation
- Core Thermal Analysis
  - Subchannel Code Mongoose++
- Core Thermal Hydraulic Testing Plan

Questions?

Acronym List

ARCAP – Advanced Reactor Content of Application Project
ARDC – Advanced Reactor Design Criteria
ARDP – Advanced Reactor Demonstration Program
CFD – Computational Fluid Dynamics
CFR – Code of Federal Regulations
EBR – Experimental Breeder Reactor
FFTF – Fast Flux Test Facility
FOM – Figure of Merit
LMP – Licensing Modernization Project
MD&A – Methodology Development and Assessment
PDC – Principal Design Criteria
PSAR – Preliminary Safety Analysis Report
SFR – Sodium Fast Reactor
SSC – Structures, systems, and components
TICAP – Technology Inclusive Content of Application Project
TREAT – Transient Reactor Test

ENCLOSURE 3

“Core Thermal Hydraulic Methods”
Presentation Material – Closed Meeting

Non-Proprietary (Public)

TerraPower

NATRIUM

Core Thermal Hydraulic Methods

a TerraPower & GE-Hitachi technology

NAT-4653

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Portions of this presentation are considered export controlled information (ECI). ECI can be disclosed to Foreign Nationals only in accordance with the requirements of 15 CFR 730 and 10 CFR 810, as applicable.

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 [[ ]]$^{(a)(4)}$, [[ ]]^{ECI}, or [[ ]]$^{(a)(4), ECI}$.

Table of Contents

Natrium Core Thermal Hydraulic Design Summary

Natrium Core Description

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Natrium Core Map Description

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Type 1 Fuel Assembly Description

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Assembly Inlet Design

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Typical Axial Temperature Profile

- Cladding temperature is an important Figure of Merit
  - Impacts creep and Fuel-Cladding Chemical Interaction limits
  - Cladding temperature constraints are typically more limiting than fuel or coolant temperature constraints
- In SFRs the steady-state peak cladding temperature occurs near the top of the heated length
  - This differs from LWRs

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Typical Radial Temperature Profile

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- Assemblies can have large radial temperature variations
- Peripheral subchannel temperatures are cooler due to their relatively large flow area
- The profile in this example is skewed due to a flux gradient across the assembly

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Core Cooling Hydraulics Methodology

Core Cooling Hydraulics

Core Cooling Hydraulics evaluates the core flow and pressure loss, margin to sodium boiling, and the peak clad and fuel temperatures with a 1D thermal hydraulic model of the core.

The peak fluid temperature is computed using the Nascent model.

Analysis is performed at steady-state rated/off-rated conditions to meet temperature limits accounting for uncertainties with Hot Channel Factors.

Following sections describe:

- Thermal Hydraulic plugin
- Assembly Hydraulic Characterization Plan
- Peak Clad Temperature with Nascent Model
- Hot Channel Factors

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Thermal Hydraulics Plugin

Thermal Hydraulics Plugin

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Assembly Hydraulic Characterization Plan

Assembly Thermal Hydraulic Characterization

- Assembly pressure is characterized across three regions:
  - Frictional losses for pinned region
  - Pressure losses at the inlet including orificing
  - Form losses at the outlet
- STAR-CCM+ is used within the conceptual/preliminary design phase as part of the CFD Assembly Simulation methodology

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Pressure Drop Calculation

- Pin assembly pressure drop is calculated using a simplified version of the Pacio-Chen Todreas Detailed correlation [1]. Coefficients are computed for assembly type specific geometry,

$$f = \begin{cases} f_L & Re_b < 700 \\ f_T & Re_b > 10^4 \\ f_L(1-\psi)^\gamma(1-\psi^\lambda) + f_T\psi^\gamma & \text{otherwise} \end{cases} \quad \text{with} \quad \begin{aligned} f_L &= \frac{C_{fbL}}{Re_b} \\ f_T &= \frac{C_{fbT}}{Re_b^{0.18}} \end{aligned}$$

- Inlet/outlet pressure drops are based on the following form, which incorporate flow rate dependency,

$$\Delta P_{inlet} = \frac{1}{2} K_{inlet} v^2 \rho_{in}$$
$$K_{inlet} = A + BRe^C$$

- The correlation and loss coefficients will be updated with the Natrium full-scale assembly testing to reduce uncertainties

Peak Clad Temperature with Nascent model

Peak Clad Temperature Overview

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Nascent Model Overview

- Nascent = Natrium simplified coolant energy transport
- Scope: peak fluid temperature in fuel assemblies
- Focus on very fast execution with simple models
  - Contrasts with Mongoose++ which focuses on general applicability
- Uses subchannel radial discretization for flexibility
- Uses constitutive models to account for
  - Intra-assembly flow distribution
  - Coolant mixing
  - Lateral conduction
- The primary calculation is a small linear system of equations
- See submittal: "Nascent Thermal Hydraulic Model White Paper" NAT-3049

Nascent Predictions Approximate Subchannel Solver Predictions at Nominal Fuel Conditions

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Nascent Predictions Approximate Subchannel Solver Predictions at Nominal Fuel Conditions

- Buoyancy causes flattening of temperature distribution at lower flow rates
- Nascent neglects buoyancy and conservatively over-predicts peak temperature at low flow rate

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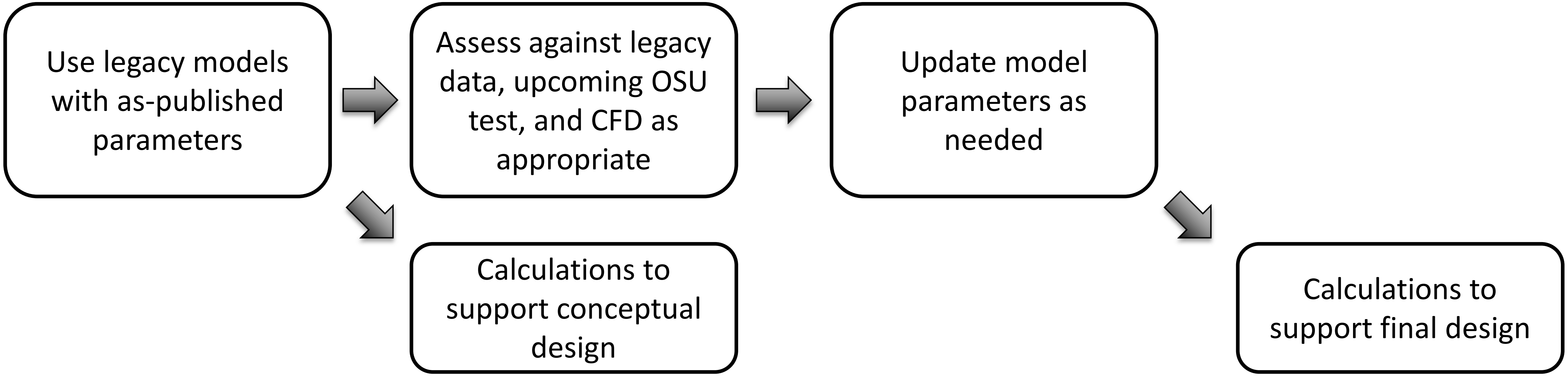
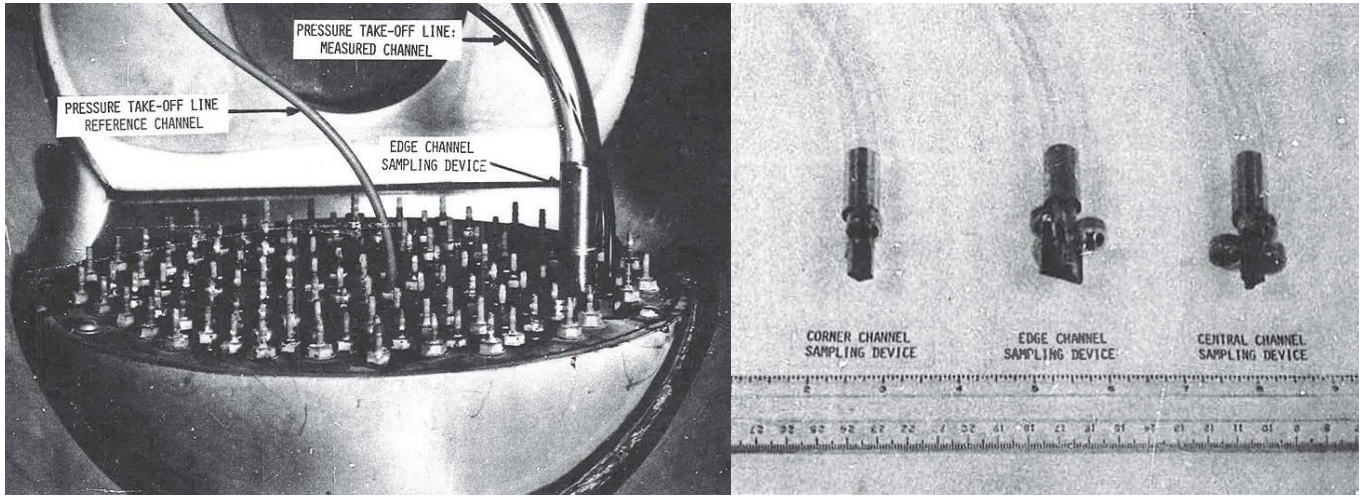
Development Plan for Nascent Constitutive Models



- Models include: subchannel friction factors, coolant mixing, and conduction shape factor
- Level of effort applied to each model will be guided by scrutable, quantitative ranking of phenomena

Leveraging Legacy Datasets for Model Development

- TerraPower is exploring legacy datasets in detail for strong understanding of the basis of existing constitutive models
- Where appropriate, we will use legacy data alongside new data to update models
- Legacy data includes a wide range of experiments
  - Heated pins in sodium
  - Salt injection and mixing in water
  - Isokinetic extraction
  - Laser Doppler anemometry
  - Etc.



Isokinetic extraction experiment setup
Source: Lorenz 1974 [6]

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Update Models for Velocity Distribution using RANS CFD Methodology

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- The intra-assembly axial velocity distribution (referred to as *flow split* in literature) is expected to depend on the number of pins in the assembly
- Consequently, models (subchannel friction factors) will be tuned using RANS CFD predictions of velocity distribution [[
  ]]$^{(a)(4),ECI}$
- The predictive capability of RANS CFD for this application will be assessed against legacy experimental data and LES CFD

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High Fidelity CFD: NekRS Overview

- NekRS is a high-fidelity tool that provides a step-change in predictive capability over typical CFD tools
  - Spectral Element Method makes high resolution achievable
  - Offers *consistent* LES filtering methods where error decreases as resolution is increased
  - Potential for Direct Numerical Simulation at low Reynolds numbers
- Argonne National Laboratory is the primary developer of NekRS and is executing the Natrium project simulations
- We plan to compare the NekRS results to measurements on an electrically-heated test

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Plan for Qualification of Nek Output

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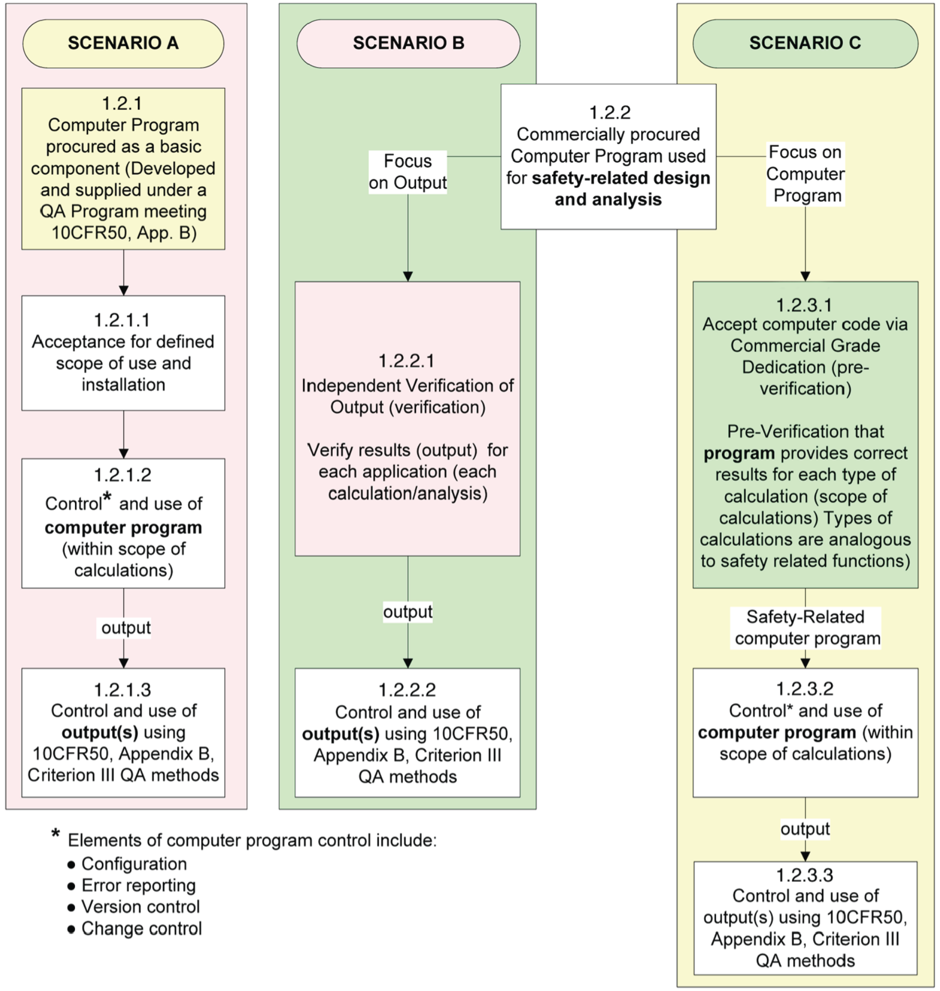

Figure 1-1
Three Basic Scenarios for Procurement of Commercially Procured Computer Programs for Use in Safety-Related Design and Analysis Applications

Nascent Model Uncertainties

- Uncertainties in the Nascent model predictions will be applied using Hot Channel Factors
- The statistical HCF will account for:
  - Sensitivity in peak temperature due to uncertainty in constitutive models
- The direct HCF will account for:
  - Localized temperature peaking due to wire azimuthal position
  - Bias that yields conservative predictions relative to OSU [[ ]](a)(4) experiment

Hot Channel Factors

Hot Channel Factor Overview

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Detailed HCFs

[[

HCFs fall into five different categories:

- Control/measurement uncertainties
- Manufacturing uncertainties
- Material property uncertainties
- Nuclear uncertainties
- Thermal hydraulic uncertainties

For conceptual and preliminary design either Natrium specific values are determined, or historical values are shown to be conservative with a development plan. For final design Natrium specific values will be determined.

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Coolant HCF Comparison

Currently the Natrium Type 1 HCFs are similar to historical designs

Comparison of coolant HCF combinations

	Coolant				
	Natrium Type 1	EBR-II	FFTF-Metal	FFTF-SAR	FFTF-SAR*
Direct combination	[[ ]](a)(4)	1.020	1.071	1.221	[[ ]](a)(4)
Statistical combination --2σ	[[ ]](a)(4)	1.089	1.087	1.067	[[ ]](a)(4)
Overall combined -- 2σ	[[ ]](a)(4)	1.111	1.164	1.302	[[ ]](a)(4)

*The statistical uncertainty replaces the direct bias for flow distribution, and the inlet flow distribution uncertainty treated statistically

Application of Hot Channel Factors

To compute the "2-sigma" inner cladding temperature:

$$\Delta T_{cladID}^{0\sigma} = F_{direct}^{coolant}\Delta T_{coolant} + F_{direct}^{film}\Delta T_{film} + F_{direct}^{clad}\Delta T_{clad}$$

$$\Delta T_{j,stat}^{3\sigma} = \begin{pmatrix} F_{j,stat}^{coolant} - 1 \\ F_{j,stat}^{film} - 1 \\ F_{j,stat}^{clad} - 1 \end{pmatrix}^{T} \begin{bmatrix} F_{direct}^{coolant} & 0 & 0 \\ 0 & F_{direct}^{film} & 0 \\ 0 & 0 & F_{direct}^{clad} \end{bmatrix} \begin{pmatrix} \Delta T_{coolant} \\ \Delta T_{film} \\ \Delta T_{clad} \end{pmatrix}$$

$$T_{cladID}^{2\sigma} = T_{inlet} + \Delta T_{cladID}^{0\sigma} + \frac{2}{3}\sqrt{\sum_{j}\left(\Delta T_{j,stat}^{3\sigma}\right)^{2}}$$

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CFD Assembly Simulation

CFD Assembly Simulation

CFD Assembly Simulation provides a method for modeling fuel and non-fuel assemblies pressure and temperature distributions. Methodology development provides confidence and validation of the thermal hydraulic predictions.

The methodology is used for pressure drop and thermal mixing predictions of assembly designs for the conceptual and preliminary design phases. It may also be used in Hot Channel Factor development within the preliminary and final design phases.

[[

Simulation Development

- Steady-state RANS simulations in STAR-CCM+
- Strategy for pin region
  - Parameters developed based on 37 pin simulations
    - Mesh parameters were selected after completing a grid convergence study

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- Flow sweeps were completed at discrete Reynolds numbers covering the expected operating range

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Validation Plans

- Pressure drop
  - preliminary comparisons to NekRS/PCTD show agreement in pressure drop within conceptual/preliminary design
  - prototypical full-scale pressure drop test data for all assembly types planned for preliminary to final design
- Flow split
  - legacy experiments with pin and wire geometry close to that of Natrium fuel assemblies
- Temperature distribution

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STAR-CCM+/NekRS Comparison

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Uncertainty Quantification

- [[ ]](a)(4)

- **Final Objectives**
  - Assemble a *robust and computationally affordable* methodology to quantify uncertainty in CFD simulations in support of licensing related calculation for the Natrium reactor
  - Demonstrate the consistent performance of the UQ methodology on selected representative validation cases
  - Assemble high-quality documentation to support the UQ methodology in a regulatory environment

- **Approach**

  [[ ]](a)(4)

  - Select representative applications and perform PIRTs for CFD-UQ to drive the formulation of the UQ methodology

Core Thermal Analysis Methodology

Core Thermal Analysis

Core Thermal Analysis evaluates the duct wall temperatures, inter-wrapper flow and temperature distribution, and interaction between the outlet plenum and the inter-wrapper flow in support of Core Restraint System performance.

Following sections describe:

- Subchannel code Mongoose++

Subchannel Code Mongoose++

Subchannel Model

- Each assembly is divided into a collection of subchannels, which are axially subdivided into small control volumes.
- Equations of mass, momentum, and energy conservation are solved in each control volume.
- Two subchannels may be connected via transverse *gaps* that permit lateral flow and mixing of enthalpy and energy.
- Empirical models are used for turbulent mixing of energy and momentum between channels.

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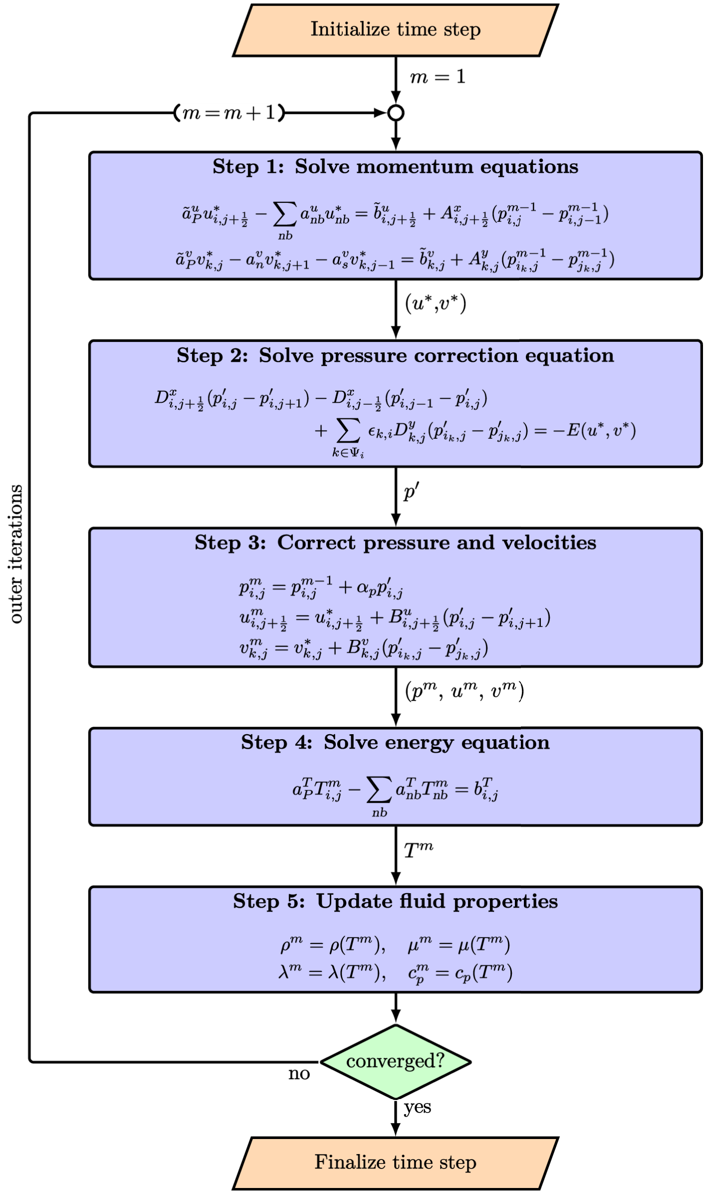
Mongoose++

- Heavily inspired by earlier COBRA-IV-I and COBRA-WC codes but takes advantage of many modern computing developments
  - Written in C++17
  - Parallel simulation of multiple assemblies using OpenMP
  - Concise, human-readable YAML input syntax
  - Binary HDF5 output format simplifies post-processing of results using python
  - Robust solver based on SIMPLE algorithm capable of handling local flow reversals
- Intended for SFR applications
  - Single phase
  - Wire-wrapped pins in hex bundles
  - Assemblies arranged in a hex grid

SIMPLE Algorithm

- Momentum equations are solved for *tentative* velocities, u^* and v^*, using the pressure computed during previous iteration
  - u^*, v^* **do not** (in general) satisfy mass conservation
- Pressure correction equation is solved using mass conservation error as source term
- Pressures and velocities are updated according to calculated pressure correction, p'
  - u^m, v^m now satisfy continuity but may not satisfy momentum
- Energy equation is solved given updated flow field, fluid properties are updated, and the loop continues until convergence



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Subchannel Equations

Mass Conservation

$$V_{i,j}\frac{\mathrm{d}\rho_{i,j}}{\mathrm{d}t} + F_n^x - F_s^x + \sum_{k\in\Psi_i}\epsilon_{k,i}F_k^y = 0$$

Energy Conservation

$$V_{i,j}\frac{\mathrm{d}}{\mathrm{d}t}(\rho h)_{i,j} + \left(F_n^x h_n^* + A_n^x\left\langle -\lambda\frac{\partial T}{\partial x}\right\rangle_n\right) - \left(F_s^x h_s^* + A_s^x\left\langle -\lambda\frac{\partial T}{\partial x}\right\rangle_s\right)$$
$$+\sum_{k\in\Psi_i}\epsilon_{k,i}\left(F_k^y h_k^* + G_k A_k^y\bar{\lambda}_k\frac{T_{i_k}-T_{j_k}}{\ell_k} + \bar{\rho}_k A_k^y \varepsilon_k^H\frac{h_{i_k}-h_{j_k}}{\ell_k}\right) = Q_{i,j}^{\mathrm{wall}}$$

Axial Momentum Conservation

$$V_{i,j+\frac{1}{2}}\frac{\mathrm{d}}{\mathrm{d}t}(\bar{\rho}u)_{i,j+\frac{1}{2}} + F_n^x u_n^* - F_s^x u_s^* + \sum_{k\in\Psi_i}\epsilon_{k,i}\left(F_k^y u_k^* + \bar{\rho}_k A_k^y \varepsilon_k^M\frac{u_{i_k}-u_{j_k}}{\ell_k}\right)$$
$$= -(\bar{\rho}Vg)_{i,j+\frac{1}{2}} - \left(\bar{\rho}V\frac{f_D}{D_h}\frac{|u|u}{2}\right)_{i,j+\frac{1}{2}} - \bar{A}_{i,j+\frac{1}{2}}(p_n - p_s)$$

Lateral Momentum Conservation

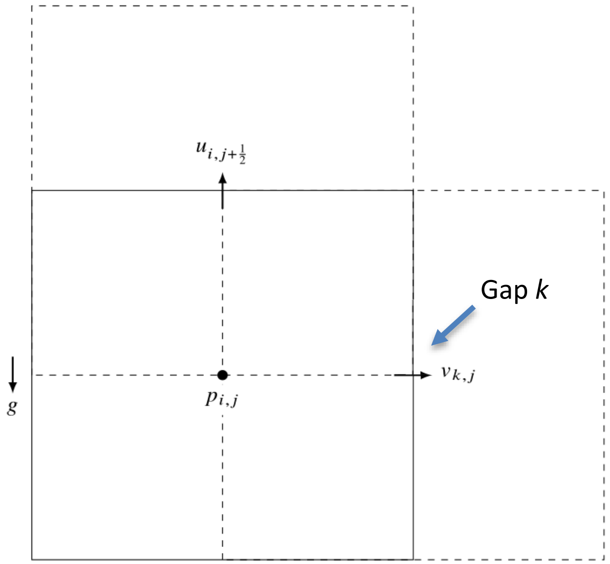
$$V_{k,j}\frac{\mathrm{d}}{\mathrm{d}t}(\bar{\rho}v)_{k,j} + F_n^x v_n^* - F_s^x v_s^* = A_{k,j}^y(p_{i_k,j} - p_{j_k,j}) - K_g A_{k,j}^y\left(\frac{\bar{\rho}|v|v}{2}\right)_{k,j}$$

$F_{n|s}^x$ = subchannel axial mass flow rate through north/south face

F_k^y = lateral mass flow rate through gap k

$\epsilon_{k,i} = +1$ if gap k is directed *out* of channel i

$\epsilon_{k,i} = -1$ if gap k is directed *into* channel i



Closure Models

G_k = Conduction Shape Factor

ε_k^H = Eddy Thermal Diffusivity

ε_k^M = Eddy Momentum Diffusivity

f_D = Darcy Friction Factor

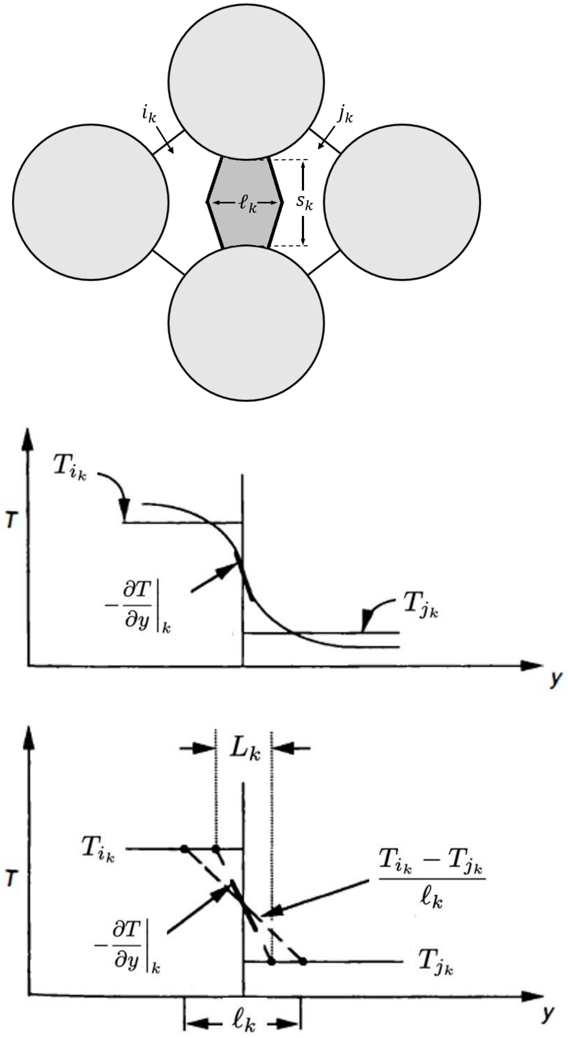
K_g = Gap Resistance Factor

Conduction Shape Factor

$$q''_k = -\bar{\lambda}_k \left.\frac{\partial T}{\partial y}\right|_k = G_k \bar{\lambda}_k \frac{T_{i_k} - T_{j_k}}{\ell_k}$$

$$\therefore \quad G_k \equiv \frac{\left(-\left.\frac{\partial T}{\partial y}\right|_k\right)}{\left(\frac{T_{i_k}-T_{j_k}}{\ell_k}\right)} = \frac{\ell_k}{L_k}$$

- q''_k : conduction heat flux across gap k
- ℓ_k : channel centroid-to-centroid distance
- L_k : effective conduction length



Turbulent Mixing and Eddy Diffusivity

- Turbulent eddies are postulated to induce an equal-mass exchange of coolant between two channels across a gap
  - No net mass transfer
  - Only energy and momentum are exchanged

- v'_k denotes the lateral turbulent velocity fluctuation
- Codes like COBRA typically assume v'_k to be proportional to the mean axial velocity in the neighboring channels

$$v'_k = \beta|\bar{u}_k|$$

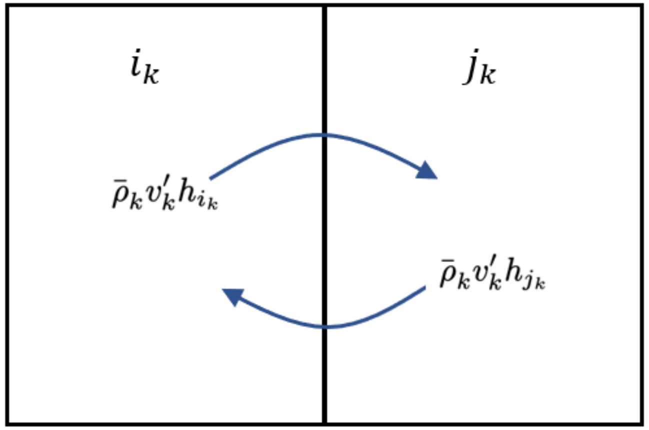
- β must be specified by the user as either a constant or correlated with Re

- Turbulent exchange of axial momentum is modeled analogously with

$$\varepsilon_k^M = \mathrm{Pr}_t \varepsilon_k^H$$

Pr_t : user-specified (constant) turbulent Prandtl number

$$q''^{\mathrm{turb}}_{i_k \to j_k} = \bar{\rho}_k v'_k (h_{i_k} - h_{j_k})$$

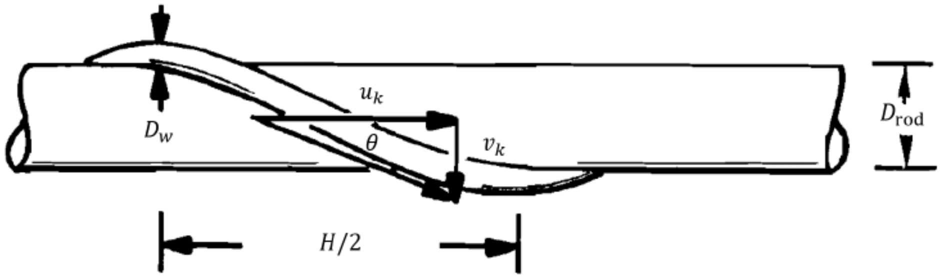


$$\varepsilon_k^H = v'_k \ell_k$$

$$q''^{\mathrm{turb}}_{i_k \to j_k} = \bar{\rho}_k \varepsilon_k^H \frac{h_{i_k} - h_{j_k}}{\ell_k}$$

Wire-wrap Models

- Option 1: Forced Diversion Crossflow model from COBRA
  - When a wire crosses a gap, a transverse velocity is imposed at that location—the lateral momentum equation is not solved for that node
  - The magnitude of the imposed velocity is proportional to the product of the local axial velocity and the tangent of the angle θ of the wire relative to the axial direction
- Option 2: Wire-enhanced diffusion with transverse forcing in periphery
  - Inspired by ENERGY model – interior wire wraps are approximated by an enhanced eddy diffusivity computed using the Cheng-Todreas correlation
  - Transverse force applied in edge channels to mimic sweeping effect



$$|v_k| = C \underbrace{\left[\frac{\pi(D_{\text{rod}} + D_w)}{H}\right]}_{\tan\theta} u_k$$

Mongoose++ Validation: ORNL FFM 2A [10]

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- 93 total tests (including duplicate runs) covering a range of flow rates and radial power distributions to simulate FFTF fuel bundles

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Mongoose++ Validation: Toshiba [2]

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- 9 experiments over a range of heat generation, flow and radial power peaking

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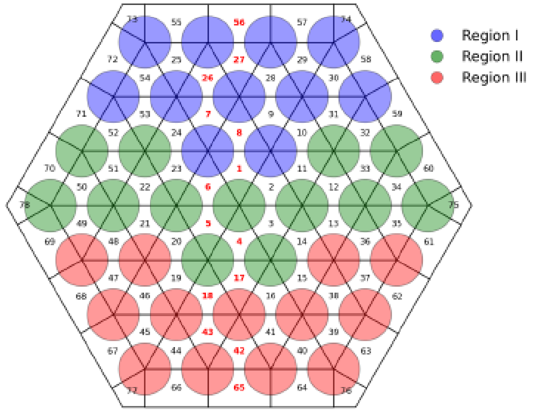

Toshiba 37 pin assembly

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Mongoose++ Validation: WARD [3]

- Westinghouse Advanced Reactor Division (WARD) experiment on a sodium cooled, electrically heated 61 pin bundle

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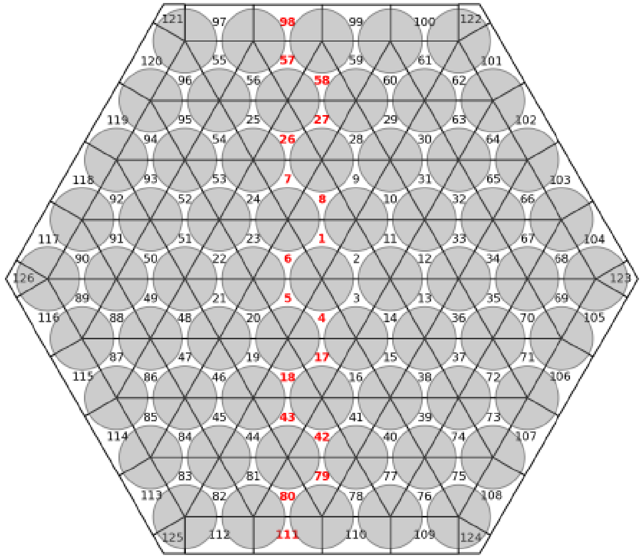

WARD 61 pin assembly

Re = 7,900, linear skew

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NekRS [[ ]](a)(4) Simulations

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- Single heated pin configurations with center pin and edge pin heated are performed similar to planned testing
  - Temperature difference between subchannels and inlet temperature is tracked to characterize the mixing/sweeping between subchannels
  - Models for Mongoose++ and Nascent will be tuned from this data in conjunction with the heated bundle test data
- Simulations with all rods heated in uniform or skewed power distribution will be used for validation of the Nascent model and Mongoose++ (along with comparable experimental data)

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Pressure Comparison

- Mongoose++ predicts a slightly higher axial pressure drop than either NekRS or STAR-CCM+
  - Friction factor correlation may require some tuning
- Both STAR-CCM+ and NekRS predict larger radial pressure gradient than Mongoose++ [[
  - Tuning wire-wrap model (more lateral forcing) could improve this

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Heated Central Pin Case

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Inter-Wrapper Flow

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Heat transfer to colder assemblies in core periphery

Some coolant deflected radially outward by load pads

Inter-wrapper coolant is heated by hotter central fuel assemblies

IWF facilitates more effective radial transport of heat compared to a purely conductive model

Likely results in reduced radial temperature gradients in ducts

Leakage from nozzles

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NOTE: diagram is of postulated IWF flow in FFTF and is not representative of Natrium design

Mongoose++ Inter-Assembly Heat Transfer Model

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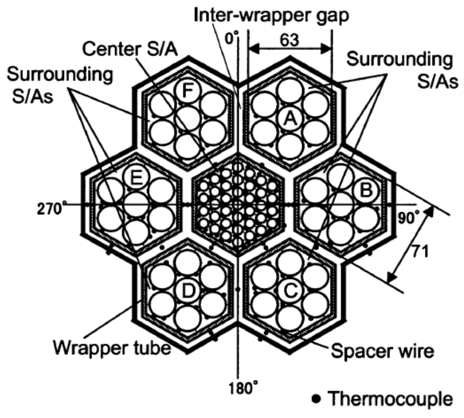
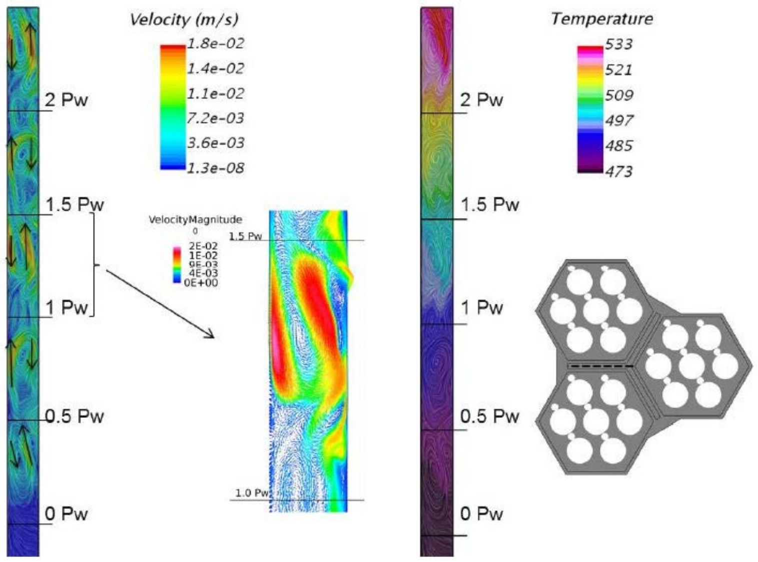
Future IAHT/IWF Model Validation

- PLANDTL-DHX (JAEA)
  - Sodium test with 7 assemblies cooled by natural convection
  - NC IWF driven by dipped DRACS in hot pool
    - Difficult conditions to replicate with Mongoose++
- KALLA (KIT)
  - LBE (not sodium) test with three 7-pin bundles
  - Used for several recent CFD validation studies
- Planning on performing our own testing if legacy data are not sufficient



PLANDTL experimental setup
Source: Nishimura et al. 2000 [8]



KALLA experiment simulation
Source: Doolaard et al. 2019 [9]

Core Thermal Analysis at Rated Conditions

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Core Thermal Analysis at Startup

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Mongoose++ Next Steps

- Additional model development
  - Pressure drop correlation updates from assembly hydraulic testing
  - Wire wrap model tuning and improvements
  - IAHT/IWF modelling
- Ongoing validation work
  - Data qualification and additional legacy tests
    - MIT 61 pin Salt Injection Tests [7]
      [[ ]](a)(4)
    - CRBR 217 pin 11:1 scale air flow tests [12]
      [[ ]](a)(4)

  [[ ]](a)(4)
- Uncertainty quantification will follow MD&A process:
  - Peak clad temperature
  - Duct temperatures

Core Thermal Hydraulic Testing Plan

Assembly Hydraulic Testing

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- Fuel assembly hydraulic characterization (inlet, bundle and exit)
  - First test ongoing using the conceptual design
  - Supporting hydraulic representation for PSAR application
- Natrium specific friction factor
- Further testing and characterization will use the final design
  - Fuel assembly
  - Primary and secondary control assemblies
  - Shield and reflector assemblies

Thermal hydraulic flow loop

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Fuel assembly instrumentation

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Nozzle Hydraulic Testing

- Inlet test section
- Test section was designed to measure the hydraulic resistance of various plates/plates stacks
- Used to calibrate design tools
- Confirm the hydraulic performance of specific stacks of plates
- Ultimately inlet losses will be characterized as part of the assembly hydraulic testing

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Velocity Limit Testing

- Orificing features in the inlet nozzle is performed with orifice plates sized to achieve the required pressure drop
- Historical testing (FFTF) specifies rather larger values. Testing need was identified to confirm those values and applicability to specific design.
- At least 5000 hours test to evaluate the effect of various velocities on hydraulic performance of orifice plates
  - Continuous pressure drop measurements,
  - Periodic optical measurements (every 500 hours) to evaluate impact on the geometry

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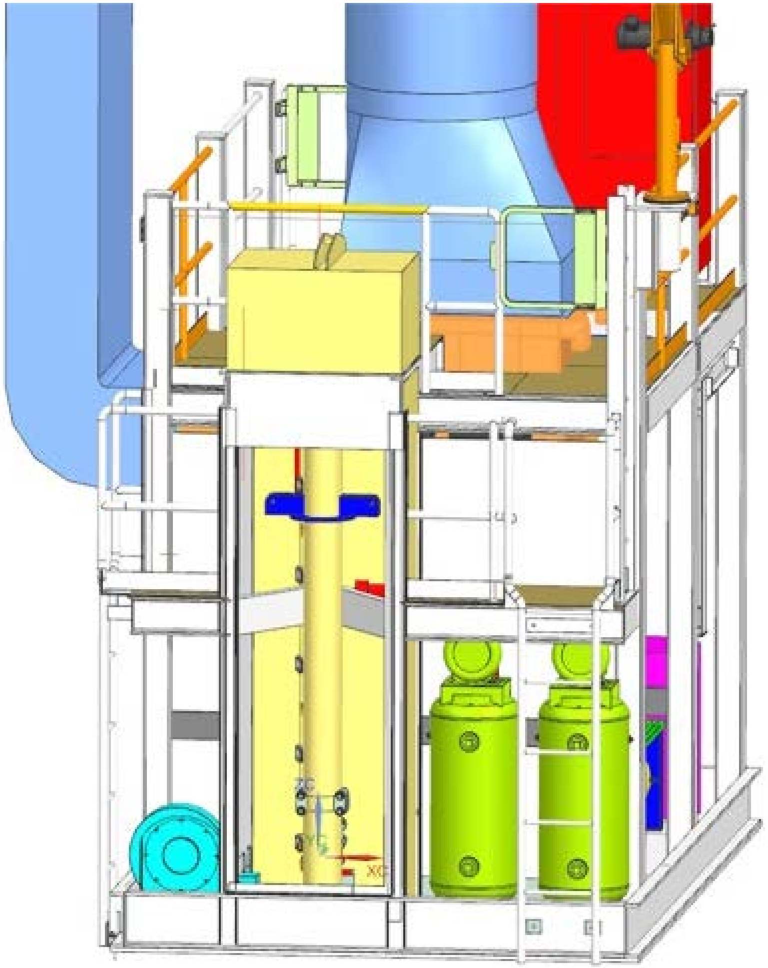
Sodium Testing Loop (SoFIE)

- Designing, building and commissioning a sodium loop at Oregon State University to support sodium testing for Natrium design
- Flexible sodium facility designed to accommodate various test sections
- Specifications
    - Oxygen content control
    - Up to 550 °C
    - 100 psid test section
    - 150 psig max operation
    - 150 gpm
    - 1.4 MW electric power
    - About 40 gallons of sodium (without the test section)



Sodium flow loop rendering

TerraPower NATRIUM

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Heated Assembly Test

- Need for specific heated bundle test to reduce uncertainty and validate models specifically for Natrium conditions
- Measure sodium temperature distribution under prototypic Natrium conditions (mass flux, power density, subchannel geometry)
- Fluid temperature distribution [[ ]](a)(4)
  - Prototypic subchannel behaviors (center, edge and corners)
  - Reynolds number (ranging from 500 to 60,000)
  - Axial power shape: chopped cosine
  - Various rod-to-rod peaking (uniform to skewed profiles) to create radial gradient (eg, proximity to control assembly)
- Packed geometry configuration to minimize geometrical uncertainty
- Pressure drop measurements
- Temperature measurements using fiber optic inserted inside the wire
  - Measurement technique being developed at University of Wisconsin to obtain high data resolution
  - Provides detailed temperature maps of the subchannels needed to validate models
- Steady-state and transient conditions

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Additional Tests in Support of Core Thermal Hydraulics

- Thermal striping at the core exit
  - Temperature differences between adjacent assemblies are limited to avoid interactions between fuel assembly exit flows
  - Thermal oscillations could impair the upper internal structure
  - CFD based methods to evaluate the design and the applicability of those legacy limits to Natrium design
- Seal testing
  - Leakage from the inlet plenum into the inter-wrapper region relies on seals between the nozzle and the receptacles
  - Piston rings, labyrinth seals will be characterized through hydraulic testing
- Hydraulic holddown
  - Hydraulic holddown of the assemblies is based on force balance
  - Testing on a dummy geometry with prototypic interface between parts will provide validation

Summary

- Core thermal hydraulic group is responsible for:
  - Core flow distribution
  - Duct temperatures in support of core restraint system
  - Core pressure drop
- Methodologies include:
  - Core Cooling Hydraulic
    - Peak clad temperature
  - Core Thermal Analysis
    - Duct temperature
  - CFD Assembly Simulation
    - Assembly thermal hydraulic characterization
- Validation plans:
  - Leverage legacy test data
  - Completed by in-house testing

Methodology Validation Plan with Design Phase

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Questions?

Acronym List

ARDP – Advanced Reactor Demonstration Program

ARMI® – Advanced Reactor Modeling Interface

CCH – Core Cooling Hydraulics

CFD – Computational Fluid Dynamics

CTA – Core Thermal Analysis

DOE – Department of Energy

EBR – Experimental Breeder Reactor

EPRI – Electric Power Research Institute

FCCI – Fuel-Cladding Chemical Interactions

FFTF – Fast Flux Test Facility

H/D - Wire lead length/rod diameter

IAHT – Inter-assembly heat transfer

IWF – Inter-wrapper flow

HALEU – High-Assay Low-Enriched Uranium

HCF – Hot Channel Factor

LES – Large Eddy Simulation

LWR – Light-Water Reactor

PCTD – Pacio-Chen Todreas Detailed

P/D – Rod Pitch to Diameter ratio

QA – Quality Assurance

QCR – Quadratic Constitutive Relation

RANS – Reynolds-Averaged Navier-Stokes

SFR – Sodium Fast Reactor

SST – Shear Stress Transport

TH – Thermal Hydraulics

THP – Thermal hydraulics Plugin

UQ – Uncertainty Quantification

USN – Universal Stack Nozzle

1D – One dimensional